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IRIS
Preliminary Safety Assessment

Westinghouse

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IRIS Preliminary Safety Assessment
-- Volume II --

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Foreword

This report presents a preliminary safety assessment of the IRIS reactor. First, an overview of IRIS approach to safety is presented, and the main engineered safeguards features of the design are discussed. Second, a preliminary study that addresses the main design basis events for the IRIS reactor is presented.

This preliminary report is developed in parallel with and in support of the development and assessment of appropriate evaluation models for IRIS. The main purpose of this study is to assist in the definition of requirements for IRIS evaluation models, and in particular to assist in the development of a complete set of phenomena identification and ranking tables (PIRTs). Once this PIRT activity, supported by a scaling and similitude analysis and by the identification of required testing, is completed, preliminary evaluation models adopted in this study will be reviewed for applicability, along with other potential models.

For the analyses presented in this report the RELAP5 Mod 3.3 computer code has been adopted. This does not necessarily represent a final selection of the computer code that will be used for IRIS safety analyses during the design certification phase. The worldwide used RELAP code was selected in this phase to simplify the collaboration among member organizations of the IRIS consortium. Analyses will be updated during the development phase of the IRIS Evaluation Models, and a final selection of computer programs to be used in the analyses shall be completed. Also, activities are in progress to define "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.

The safety assessment presented in this report cannot of course be as complete as required for SAR or DCD Chapter 15: the results should be considered preliminary and indicative of the IRIS performance. These results support the main purposes of this study, which is to assist in the identification of 1) the important phenomena, 2) sequences that IRIS Evaluation Models will have to address and 3) component test requirements. These results also support the final design of the IRIS protection and monitoring system.

The events analyzed in this study are a subset of those studied for AP1000 and AP600, and have been selected (1) to address those events where IRIS response is different from AP1000, and (2) to provide an initial overview of the IRIS response to different anticipated operational occurrences and design basis events. For each category of events, the rationale used in selecting the most representative sequences is discussed.

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2.4 Reactivity and Power Distribution Anomalies

A number of faults are postulated that results in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system. Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by the mispositioning a fuel assembly in the core. These events are discussed in this section. Detailed analyses are typically provided for the most limiting of the following events.

A. Uncontrolled rod cluster control rod assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition

B. Uncontrolled RCCA bank withdrawal at power

C. RCCA misalignment

D. Startup of an inactive reactor coolant pump at an incorrect temperature

E. A malfunction or failure of the flow controller in a boiling water reactor recirculation loop that result in an increase reactor coolant flow rate (not applicable to PWRs)

F. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

G. Inadvertent loading and operation of a fuel assembly in an improper position

H. Spectrum of RCCA ejection accidents

Items A, B, D, and F above are Condition II events, item G is a condition III event, and item H is a condition IV event. Item C includes both Conditions II and III events.

Because the IRIS core is not significantly different than other Westinghouse PWR cores, this document contains no detailed analyses for any of these events since no major phenomenological differences between IRIS and other PWRs are expected in the analyses of these events, with the exception of the RCCA ejection accident that is eliminated by design with the use of internal control rod drive mechanisms. Westinghouse evaluation models will be applicable without major modifications, and therefore it was considered not necessary to perform a complete quantitative assessment of these events at this time. Also, from a protection and mitigation point of view, the IRIS response will have the same characteristics as current Westinghouse PWRs.

The following sections will provide a brief overview of these events, with particular emphasis on any differences between IRIS and other Westinghouse PWRs. The focus of the discussion will be on the features of the protection and safety monitoring system that provide a complete and adequate protection for this set of events, and ample reference is made to the AP600/AP1000 design, since IRIS protection and mitigation for these events builds on the AP600/AP1000 design certification.

2.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Conditions

An RCCA withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCAs that result in a power excursion. Such a transient can be caused by a malfunction of the reactor control or rod control systems. This can occur with the reactor subcritical, at zero power, or at power. The at-power case is discussed in Section 2.4.2

.

Although the reactor is normally brought to power from a subcritical condition by RCCA withdrawal, initial startup procedures with a clean core use boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that typically assumed in the analysis of this event (see Section 2.4.6).

The RCCA drive mechanisms are grouped into pre-selected bank configurations. These groups prevent the RCCAs from being automatically withdrawn in other than in their respective banks. Power supplied to the banks is controlled such that no more that two banks are withdrawn at the same time and that the banks are withdrawn in their proper sequence. The RCCA drive mechanisms are the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed is that occurring with the simultaneous withdrawal of the combination of two sequential RCCA banks having the maximum combined worth at maximum speed.

This event is a Condition II event as defined in section 2.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a fast rise in flux terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion limits the power increase during the time required for detection and protective action. Should a continuous RCCA withdrawal accident occur, the transient is terminated by the following automatic features of the protection and safety monitoring system:

- Source range high neutron flux reactor trip – this trip function is actuated when two out of four independent source range channels indicate a neutron flux above a pre-selected, manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when the coincident two out of four intermediate range channels indicate a flux level below a specified level.
- Intermediate range high neutron flux reactor trip – this trip function is actuated when two out of four independent, intermediate range channels indicate a flux level above a pre-selected, manually adjustable setpoint. It may be manually bypassed only after two out of four power range channels are reading above approximately 10 percent full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.
- Power range high neutron flux reactor trip (low setting) - this trip function is actuated when two out of four power range channels indicates power level above approximately 25 percent of full power. It may be manually bypassed when two out of four power range channels are reading above approximately 10 percent full power.

It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (high setting) – this trip function is actuated when two out of four power range channels indicate a power level above a preset setpoint. It is always active.
- High nuclear flux rate reactor trip – This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above a preset setpoint. The trip may be manually bypassed after the coincident two out of four nuclear power range channels are manually reset.

In addition, control rod stops on high intermediate range flux level (one out of four) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

The analysis of the RCCA bank withdrawal from a subcritical core will be performed using evaluation models and computer codes commonly used to analyze other Westinghouse PWRs, and therefore is not discussed here. Analyses will be presented to show that in the event of an RCCA withdrawal accident from subcritical conditions, the core and the reactor coolant system are not adversely affected because the combination of thermal power and the reactor coolant system temperature do not result in a DNBR less than the safety analysis limit value. Thus, no fuel or cladding damage is predicted as a result of DNB.

It should be noted that while these functions are essentially identical to other Westinghouse PWRs, in IRIS, the design of the source, intermediate and power range instrumentation is still under development. This new design is required because the ex-vessel instrumentation currently used would not provide adequate performance in IRIS, due to the shielding effect of the wide reactor vessel downcomer. Thus, alternative solutions are under evaluation. This will not impact the analyses, since the IRIS instrumentation will meet or exceed the functional characteristics of current Westinghouse plants.

2.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

Uncontrolled RCCA bank withdrawal at power results in an increase in core heat flux. Because the heat extraction from the steam generator lags behind the core power generation, there is a net increase in the reactor coolant system temperature. Unless terminated by manual or automatic action, the power mismatch and resultant temperature rise could eventually result in DNB. Therefore, to avert damage to the fuel cladding, the protection and safety monitoring system is designed to terminate any such transient before the DNBR falls below the design limit.

The event is a Condition II incident as defined in Section 2.0.1.

Should a continuous RCCA withdrawal at power accident occur, the transient is terminated by the following automatic features of the protection and safety monitoring system:

- Reactor Trip on high power range neutron flux (high setpoint) – see Section 2.4.1 for a discussion of this protection
- Reactor Trip on high power range positive neutron flux rate – see Section 2.4.1 for a discussion of this protection
- Reactor Trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.
- Reactor Trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to prevent the allowable linear heat generation rate (kW/ft) from being exceeded
- Reactor Trip on high pressurizer pressure is actuated if any two out of four pressure channels exceed a high pressure setpoint. This setpoint is less than the set pressure for the pressurizer safety valves
- A high pressurizer water level reactor trip is actuated if any two out of four level channels exceed the setpoint when the reactor power is above approximately 10 percent (Permissive P-10)

Note that in IRIS, due to the large thermal inertia of the RCS and the increased steam space in the pressurizer, the high pressurizer pressure and high pressurizer level trip setpoint are not expected to be actuated during this event.

In addition to the preceding reactor trips, the withdrawal of the RCCAs is also stopped/blocked by the following signals:

- High neutron flux (two out of four power range)
- Overpower ΔT (two out of four)
- Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ?T trips provide protection over the full range of RCS conditions is common with other Westinghouse PWRs and is not discussed here in detail. First, allowable reactor coolant average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure are defined in a diagram. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, the trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point. The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); and overpower and overtemperature ΔT (variable setpoints).

An uncontrolled RCCA bank withdrawal at power will not present any significant phenomenological difference from current Westinghouse PWRs, and the analyses will show that the high neutron flux (high setting, low setting, and high positive rate) and overtemperature ΔT channels provide adequate protection over the entire range of possible reactivity insertion rates.

2.4.3 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

Rod cluster control assembly (RCCA) misoperation accidents include:

a. One or more dropped RCCAs within the same group
b. A dropped RCCA bank
c. Statically misaligned RCCA
d. Withdrawal of a single RCCA

Each RCCA has a position indicator channel which displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod-at-bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs are always moved in pre-selected banks, and the banks are always moved in the same pre-selected sequence. Each bank of RCCAs is divided into one or two groups. The rods comprising a group operate in parallel. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures can cause either RCCA insertion or immobility, but not RCCA withdrawal.

The dropped RCCA, dropped RCCA bank, and statically misaligned RCCA events are classified as ANS Condition II incidents (incidents of moderate frequency), as defined in Section 15.0.1. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed results from multiple wiring failures or multiple serious operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered low such that the limiting consequences may include slight fuel damage.

Thus, consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition "Condition III occurrences include incidents, any one

of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged . . .".
This selection of criterion is in accordance with GDC-25 which states, "The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods." (emphasis added). It has been shown that single failures resulting in RCCA bank withdrawals do not violate specified fuel design limits. Moreover, no single malfunction can result in the withdrawal of a single RCCA. Thus, it is concluded that the criterion established for the single rod withdrawal at power is appropriate and in accordance with GDC-25.

A dropped RCCA or RCCA bank is detected by one or more of the following:

- Sudden drop in the core power level as seen by the nuclear instrumentation system
- Asymmetric power distribution as seen on neutron detectors or core exit thermocouples
- Rod at bottom signal
- Rod deviation alarm (control rods only)
- Rod position indication

Misaligned RCCAs are detected by one or more of the following:

- Asymmetric power distribution as seen on neutron detectors or core exit thermocouples
- Rod deviation alarm (control rods only)
- Rod position indication

The resolution of the rod position indicator channel is ±5 percent of span (±7.5 inches). A deviation of any RCCA from its group by twice this distance (10 percent of span or 15.0 inches) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications.

If one or more of the rod position indicator channels is out of service, operating instructions shall be followed to assure the alignment of the non-indicated RCCA. The operator also takes action as required by the Technical Specifications.

In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control failure would both be displayed to the operator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip. The Condition III Standard Review Plan Section 15.4.3 evaluation criteria will be met; however, due to the increase in local power density, it is not possible in all cases to provide assurance that the core safety limits will not be violated.

For all these events IRIS will not present significant phenomenological differences from current Westinghouse PWRs, and the same evaluation models used in the analysis of AP600/AP1000 will be applied to IRIS. The analyses will verify that the Standard Review Plan Section 15.4.3 evaluation criteria will be met for all the events in this category.

2.4.4 Startup of an inactive Reactor Coolant Pump at an incorrect Temperature

The IRIS Technical Specifications will require all RCPs to be operating while in Modes 1 (Power Operation) and 2 (Hot Standby). The maximum initial core power level for the startup of an inactive loop transient is approximately zero MWt. Furthermore, the reactor will initially be subcritical by the Technical Specification requirement. There will be no increase in core power, and no automatic or manual protective action is required.

2.4.5 A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate

This is a BWR events, and as such not applicable to IRIS.

2.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant System

Other than control rod withdrawal, the principal means of positive reactivity insertion to the core is the addition of unborated, primary-grade water from the demineralized water transfer and storage system into the reactor coolant system through the reactor makeup portion of the chemical and volume control system. Normal boron dilution with these systems is a manually initiated operation under strict administrative controls requiring close operator surveillance. Procedures limit the rate and duration of the dilution. A boric acid blend system is available to allow the operator to match the makeup's boron concentration to that of the reactor coolant system during normal charging.

An inadvertent boron dilution is caused by the failure of the demineralized water transfer and storage system or chemical and volume control system, either by controller, operator or mechanical failure. The chemical and volume control system and demineralized water transfer and storage system are designed to limit, even under various postulated failure modes, the potential rate of dilution to values which, with indication by alarms and instrumentation, will allow sufficient time for automatic or operator response to terminate the dilution. It should also be noted that the large IRIS reactor coolant system inventory will have an important function in providing an inherent mitigation to this event.

An inadvertent dilution from the demineralized water transfer and storage system through the chemical and volume control system may be terminated by isolating the makeup pump suction line to the demineralized water transfer and storage tank. Flow from the demineralized water transfer and storage system, which is the source of unborated water, may be terminated by closing isolation valves in the chemical and volume control system. The lost shutdown margin may be regained by opening the isolation valves to the boric acid tank and thus allowing the addition of borated water to the reactor coolant system.

Generally, to dilute, the operator must perform two distinct actions:

- Switch control of the makeup from the automatic makeup mode to the dilute mode
- Start the chemical and volume control system makeup pumps.

Failure to carry out either of the above actions prevents initiation of dilution. As in the AP1000, IRIS chemical and volume control system makeup pumps do not run continuously (they are expected to be operated once per day to make up for reactor coolant system leakage), a makeup pump is started when the volume control system is placed into dilute mode.

The status of the reactor coolant system makeup is continuously available to the operator by the following:

- Indication of the boric acid and blended flow rates
- Chemical and volume control system makeup pumps status
- Deviation alarms, if the boric acid or blended flow rates, or total flow, deviate by more than the specified tolerance from the preset value
- When reactor is subcritical
  - High flux at shutdown alarm
  - Indicated source range neutron flux count rates
  - Audible source range neutron flux count rate
  - Source range neutron flux - multiplication alarm
- When the reactor is critical
  - Axial flux difference alarm (reactor power ≥ 50 percent rated thermal power)
  - Control rod insertion limit low and low-low alarms
  - Overtemperature ΔT alarm (at power)
  - Overtemperature ΔT reactor trip
  - Power range neutron flux - high, both high and low setpoint reactor trips.

This event is a condition II incident (a fault of moderate frequency), as defined in Section 2.0.1.

Boron dilution events during refueling (Mode 6), cold shutdown (Mode 5), safe shutdown (Mode 4), hot standby (Mode 3), startup (Mode 2) and power operation (Mode 1) are

considered in the analysis. Conservative values of relevant parameters are used (high reactor coolant system critical boron concentration, high boron worths, minimum shutdown margins, and lower than actual reactor coolant system volumes). These assumptions result in conservative determinations of the time available for operator or automatic system response after detection of a dilution transient in progress.
No detailed analysis or discussion of the event in different operation modes is presented here since IRIS does not present any phenomenological difference from other Westinghouse PWRs, so that existing evaluation model and procedures can be used without requiring additional qualifications.

A summary of the accident evolution for different operation modes is:

- Inadvertent boron dilution events are prevented by administrative controls during refueling;

- Inadvertent boron dilution events during cold shutdown, safe shutdown, and hot standby modes are automatically terminated by the protection system.

- Inadvertent boron dilution events during startup and power operation modes, if not detected and terminated by the operator, result in an automatic reactor trip. Upon any reactor trip signal, a safety-related function automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Following isolation, the unborated water that may remain in the purge volume of the chemical and volume control system is not sufficient to return the reactor to criticality. In power operation mode with the reactor in automatic rod control mode, a boron dilution results in a power and temperature increase that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller, if un-noticed and terminated by the operator, will result in several alarms to the operator. Given the many alarms, indications, and the inherent slow process of dilution at power, the operator will have several hours for action following the rod insertion limit low-low alarm before the shutdown margin is lost.

2.4.7 Inadvertent Loading and Operation of a Fuel Assembly in Improper Position

Fuel and core loading errors can inadvertently occur, such as those arising from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod with one or more pellets of the wrong enrichment, or having a full fuel assembly loaded with pellets of the wrong enrichment. This leads to increased heat fluxes if the error results in placing fuel in core regions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes more peaked than those calculated with the correct enrichments. A 5-percent uncertainty margin is included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The online core monitoring system is used to verify power shapes at the start of life and is capable of revealing any assembly enrichment or loading errors that cause power shapes to be peaked in excess of the design value. Power-distribution-related measurements are incorporated in the evaluation of calculated power distribution information using the incore instrumentation processing algorithms contained within the online monitoring system. The processing algorithms contained within the online monitoring system will be functionally identical to those historically used for the evaluation of power distribution measurements in Westinghouse pressurizer water reactors.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification number of each assembly will be checked before it is moved into the core. Serial numbers read during or after fuel movement are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with in-core flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one-half of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. In-core flux measurements are taken during the startup subsequent to every refueling operation.

This event is classified as an ANS Condition III incident (an infrequent fault), as defined in Section 15.0.1.

Several different fuel loading errors will be considered in detailed analyses, and in the unlikely case that a loading error occurs, the analyses in this section will confirm that the resulting power distribution effects are either easily detected by the online core monitoring system or cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes. Evaluation Models and computer codes used for this analyses are identical to those currently licensed and used for other Westinghouse plants.

2.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

The accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Since IRIS features internal Control Rod Drive Mechanisms, no driving force that can eject an RCCA from the core is present. Therefore, this event is eliminated by design in IRIS.

2.5 Increase in Reactor Coolant Inventory

Two events are postulated that may result in an increase in reactor coolant system inventory:

- Inadvertent actuation of the Emergency Boration Tanks during power operation
- Chemical and volume control system malfunction that results in an increase in reactor coolant system inventory

For IRIS, these events don't have any significant effect due to specific features of the design, as discussed in the following sections.

2.5.1 Inadvertent actuation of the Emergency Boration Tanks during power operation

Spurious emergency boration tank (EBT) operation at power could be caused by an operator error, a false electrical actuation signal, or a valve malfunction. A spurious signal may originate from any of the safeguard actuation channels. The IRIS protection logic is such that a single failure cannot actuate both EBTs without also actuating the EHRS. A scenario such as this is the spurious "S" signal event. However, if one of the EBTs is inadvertently actuated by a single failure, the event may progress with the plant at power until a reactor trip is reached. For a plant under automatic rod control, a reactor trip on high pressurizer water level would eventually be reached if the injection of coolant from the EBTs was sufficient to reach the trip setpoint.

The inadvertent opening of one of the EBTs discharge valves, due to operator error or valve failure, results in emergency boration tank injection flow leading to a boration similar to that resulting from a chemical and volume control system malfunction event. If the automatic rod control system is operable, it will begin to withdraw rods from the core to counteract the reactivity effects of the boration. As a result the EBT will continue injection and slowly raise the pressurizer level. If the EBT injection is sufficiently large, the high pressurizer level trip setpoint would be finally reached. However, in IRIS each EBT has a limited inventory, with a volume of 450 ft^3 (12.74 m^3).

A simple analysis is provided to show that injection from a single EBT will not result in a significant increase in reactor coolant system (RCS) volume. Assuming that the EBT initially is at cold conditions (40°F / 4.4°C) and that the injected water will be heated to the hot full power temperature (623°F / 328.3°C), the total volume increase in the RCS is due to the EBT fluid expansion from cold to hot conditions. For the conservative temperature excursion assumed, the coolant density changes from 62.9 to 41.0 lbm/ft^3 (1007 kg/m^3 to 656.8 kg/m^3). Therefore, the volume of the RCS would increase by 240 ft^3 (6.8 m^3). This injection will not be sufficient to reach the high level trip setpoint, and thus no protection measure will be required. After detecting the fault, the operator will proceed to a plant shutdown to re-establish boration concentration and EBT inventory. Even if injection from both EBTs is assumed, the total volume increase of the RCS of 480 ft^3 (13.6 m^3) does not have the potential for overfilling the pressurizer. Thus, the event is inherently mitigated by the design features of the plant, and does not present any safety concern.

This inherent IRIS mitigation is due to the novel approach in responding to postulated loss of coolant accidents (elimination of Large Break LOCAs by design, and mitigation of small break LOCAs by maintaining inventory rather than by safety injection at high pressure). Since the EBT has only the function of providing boration to the RCS, a small volume is required and therefore actuation of the EBT will only influence the boron concentration in the reactor coolant system, and have only a minimal impact on other process variables.

2.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

An increase of the reactor coolant inventory, which results from addition of cold unborated water to the reactor coolant system, has been discussed in Section 2.4.6.

In this Section, the increase of reactor coolant system inventory due to the addition of borated water at the same concentration of the reactor coolant system or higher, is analyzed.

The increases in reactor coolant inventory may be due to the spurious actuation of one or both of the chemical and volume control system makeup pumps or by the closure of the letdown path. If the chemical and volume control system is injecting highly borated water into the reactor coolant system, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power and subsequent coolant shrinkage. The load decreases due to the effect of reduced steam pressure after the turbine throttle valve fully opens. At high chemical and volume control system boron concentration, low reactivity feedback and reactor in manual rod control, an “S” signal will be generated by either the low Tcold or low steamline pressure setpoints before the chemical and volume control system can inject a significant amount of water into the reactor coolant system. In this case, the chemical and volume control system malfunction will proceed similarly to a spurious “S” signal. Several actions are taken upon reaching an “S” signal setpoint: in particular, the injection of water is terminated by the automatic isolation of the chemical and volume system. Also, since the EHRS is also actuated on an “S” signal, and due to the limited injection capacity of the emergency boration tank, no further level swelling occurs that has the potential for filling the pressurizer volume.

If the boron concentration of the water injected by the chemical and volume system is at the same concentration of the reactor coolant system, the pressurizer level will increase while the reactor continues to operate at power. If left unmitigated, the water injected by the chemical and volume system will lead to a continuous increase in the pressurizer level until a high-2 pressurizer level setpoint is reached. In this case no “S” signal is generated and the event is terminated by the chemical and volume control system automatic isolation upon reaching the safety-related high-2 pressurizer level setpoint.

Due to the limited injection that follows actuation of the emergency boration tank, this event is terminated at most with a reactor trip or a spurious “S” signal. In either case, no other consequence follows the event.

2.6 Decrease in Reactor Coolant System Inventory

This section discusses the following events that result in a decrease in reactor coolant inventory:

- An inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system (ADS)
- A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment
- A steam generator tube failure
- A loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

All these events are discussed in the following sections, with particular emphasis on the loss of coolant accidents. It is important to note that the LOCA analyses presented in this report do not represent the final safety analysis but are presented to provide an initial assessment of the important phenomena during a LOCA for IRIS.

2.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS

An inadvertent depressurization of the reactor coolant system can occur as a result of an inadvertent opening of a pressurizer safety valve or ADS valves. Initially, the event results in a rapidly decreasing reactor coolant system pressure. The average coolant temperature decreases slowly, but the pressurizer level increases until the reactor is tripped.

The reactor is tripped by the following reactor protection system signals:
- Overtemperature ΔT
- Pressurizer low pressure

The inadvertent opening of a pressurizer safety valve can only be postulated due to a mechanical failure.

The ADS system consists of a single stage of depressurization valves. The ADS includes two redundant parallel valve paths such that no single failure prevents operation of the ADS when it is called upon to actuate and the spurious opening (due for example to a mechanical failure of a valve) of a single ADS valve does not initiate ADS flow. This configuration also allows testing of the ADS during operation without initiating ADS flow. To actuate the ADS manually from the main control room, the operators actuate two separate controls positioned at some distance apart on the main control board. Therefore, one unintended operator action does not cause ADS actuation.

The IRIS ADS is therefore designed such that inadvertent operation of the ADS is classified as a Condition III event, an infrequent fault. However, since ADS testing is allowed during normal operation, this event is conservatively analyzed as a Condition II event. An inadvertent opening of a pressurizer safety valve is a Condition II event, a fault of moderate frequency.

For IRIS, this analysis is performed for the most limiting event, based on which flow path has the largest flow area and the shortest opening time. For IRIS, because of the larger flow area and shorter opening time of a safety valve, the inadvertent opening of a safety valve is the event considered in this analysis.

These analyses are performed to demonstrate that the departure from nucleate boiling ratio (DNBR) does not decrease below the design limit values while the reactor is at power.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, analyses are performed to evaluate the effects produced by a possible consequential loss of offsite power during inadvertent reactor coolant system depressurization events. As discussed in subsection 2.0.14, the loss of offsite power is considered as a direct consequence of a turbine trip occurring while the plant is operating at power. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

Detailed analyses for this event are not provided here. Evaluation Models similar to the ones used for other Westinghouse PWR analyses will be used with only limited modifications. Because of the IRIS large thermal margin, it is expected that the safety limit DNBR will not be approached during this event, and the reactor protection system will provide adequate protection against the reactor coolant system depressurization events.

The long-term plant responses due to a stuck-open ADS valve, or pressurizer safety valve, which cannot be isolated, is bounded by the small-break LOCA analysis discussed in Section 2.6.4.

2.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

The small lines carrying primary coolant outside containment are the reactor coolant system sample lines and the letdown line from the chemical and volume control system to the liquid radwaste system. These lines are used only periodically. No instrument lines carry primary coolant outside the containment.

When primary coolant must be removed from the RCS due to the addition of water for boron dilution or the addition of boric acid solution for boration, the chemical and volume control system purification flow is diverted out of containment to the liquid radwaste system. Before passing outside containment, the flow stream passes through the chemical and volume control system heat exchangers and mixed bed demineralizer, and then passes through the letdown orifice. The flow leaving the containment is at low pressure, at a temperature of less than 140°F and has been cleaned by the demineralizer. The flow out of a postulated break in this line is limited to the chemical and volume control system makeup pump capability to add water to the RCS, i.e.; 100 gpm. Considering the low temperature of the flow and the reduced iodine activity because of demineralization, this event is not analyzed, since the postulated sample line break is more limiting.

The sample line isolation valves inside and outside containment are open only when sampling. The failure of the sample line is postulated to occur between the isolation valve outside the containment and the sample panel. Because the isolation valves are open only when sampling, the loss of sample flow provides indication of the break to plant personnel. In addition, a break in a sample line results in activity release and a resulting actuation of area air radiation monitors.

The loss of coolant reduces the pressurizer level and creates a demand for makeup to the reactor coolant system. Upon indication of a sample line break, the operator would take action to isolate the break.

The sample line includes a flow restrictor at the point of sample to limit the break flow to less than 130 gpm. The liquid sampling lines are 1/4 inch tubing which further restricts the break flow of a sampling line outside containment. Offsite doses are based on a conservative break flow of 130 gpm with isolation after 30 minutes.

IRIS will not present significant differences from the Westinghouse AP600/AP1000 designs, and therefore an assessment of this event is not provided at this moment.

2.6.3 Steam Generator Tube Rupture

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods within the allowance of the Technical Specifications. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system.

The assumption of a complete tube severance is conservative because the steam generator tube material (Alloy 690) is a corrosion-resistant and ductile material and IRIS steam generators tubes are mostly in compression, with primary water flowing outside them. The more probable mode of tube failure is tube collapse, or one or more smaller leaks of undetermined origin. Activity in the secondary side is subject to continual surveillance with radiation monitoring on each steam line, and an accumulation of such leaks, which exceeds the limits established in the Technical Specifications, is not permitted during operation.

The IRIS design provides a simple and effective set of automatic protective actions to mitigate the consequences of a SGTR. The automatic actions include reactor trip, isolation of the faulted steam generator and eventual actuation of the emergency heat removal system (EHRS). Since the steam generators are designed for full primary design pressure up to the isolation valves, the isolation of the faulted steam generator does not result in the over-pressurization of any equipment, and this isolation terminates the release of radioactivity.

Different scenarios can be assumed, depending on the size of the break. For smaller leaks, the loss of primary fluid mass will be within the capability of the makeup system, and release of radioactivity to the secondary side may remain within the technical specification limit and thus not require any action from the protection system. For any breaks where the secondary activity exceeds the Technical Specification limit, protection is provided by the high radiation monitors on the steam line. This safety grade function generates a reactor trip and isolates the steam generator pair associated with the faulted line. As a backup to the radiation monitors, the low pressurizer level trip signal may be reached if leakage from the reactor coolant system to the steam generators goes beyond the capability of the makeup system.

Following isolation of the faulted steam generator(s), the other steam generators will be available for decay heat removal, either using the startup feedwater and turbine bypass system, or, if either is not available, through the EHRS.

Once the faulted steam generator pair is isolated, the flow from the reactor coolant system will rapidly fill the isolated pair. Since the isolated volume of the steam system is small compared to the primary system, this will not lead to a loss of mass that would have a negative impact on the core cooling capability of the reactor coolant system. The steam system and EHRS connected to the faulted steam generators will thus provide a “secondary” pressure boundary that will prevent any release of radioactivity to the public. Since the isolated steam generator pair is not provided with safety valves, there is no potential for release of radioactivity to outside the containment unless other failures are postulated.

If start-up feed water and/or steam dump to the condenser are not available, the EHRS connected to the other intact steam generators will cool down the plant and depressurize the primary side, thus resulting in automatic cooldown and depressurization of the reactor coolant system. No actuation of the ADS to depressurize the plant is required.

No detailed analysis is provided because of the benign response of IRIS to a SGTR (i.e., no activity release, no SG overpressure, etc.). The inherent features of the design in limiting the consequences of this event and preventing any release of radioactivity to the public, provide adequate assurance at this stage of the design that the plant response to a SGTR will be significantly improved over current PWRs.

2.6.4 Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

A LOCA is the result of a pipe rupture of the reactor coolant system pressure boundary.

Conventionally, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered a Condition IV event (a limiting fault) because it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see subsection 15.0.1). A minor pipe break (small break), as considered in this subsection, is defined as a rupture of the reactor coolant pressure boundary (Section 5.2) with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event because it is an infrequent fault that may occur during the life of the plant.

In IRIS, there are no pipes with a cross-sectional area larger than about 0.09 ft^2 that are part of the reactor coolant system pressure boundary. Therefore, no large break scenario needs to be addressed, and large break LOCAs are eliminated by design in

IRIS. Consideration must still be provided for the small break LOCAs, for pipes with a total cross-sectional area of up to 0.09 ft^2.

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- Localized cladding oxidation shall not exceed 17 percent of the total cladding thickness before oxidation.
- The amount of hydrogen generated from fuel element cladding reacting chemically with water or steam shall not exceed 1 percent of the total amount if all metal cladding were to react.
- The core remains amenable to cooling for any calculated change in core geometry.
- The core temperature is maintained at a low value, and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are established to provide significant margin in emergency core cooling system performance following a LOCA. In IRIS, these acceptance criteria are conservatively satisfied since the core remains covered during the whole duration of the transient, and thus an effective cooling is provided to prevent excessive temperatures and oxidation rates.

The small break LOCA is an event for which IRIS presents significant differences from current PWRs. Note that the IRIS response for other events analyzed in this document are phenomenologically very similar to current PWRs, and therefore only minor modifications to the evaluation models currently used are necessary. However, the small break LOCA presents fundamental differences that require a novel approach. Therefore, the analyses presented in this document are not to be considered as the final safety assessment for IRIS, but are provided to allow a more complete understanding of the evolution of this transient for IRIS. These analyses should only be considered as a preliminary assessment of both the available margin and the conformance to the acceptance criteria.

2.6.4.1 Overall Approach to SBLOCA Analyses in IRIS

The approach used for SBLOCA analyses at this stage of the IRIS program is different from what has been used for the other events presented in this report. While for all the other transients and accidents the approach is that the IRIS response is essentially the same as current PWRs (e.g., AP1000) from a phenomenological point of view and therefore current Westinghouse evaluation models can be used with relatively minor modifications, the SBLOCA in IRIS presents sufficient difference to require an ad hoc approach.

The Evaluation Model Development and Assessment Procedure (EMDAP), outlined in DG-1120 (Reference 1), can be used to describe the basic rationale being used in the development and qualification of an evaluation model for IRIS SBLOCA analysis.

The principles of an EMDAP were developed and applied in a study on quantifying reactor safety margins (Reference 2). In that report the code scaling, applicability, and uncertainty (CSAU) evaluation methodology was applied to a large-break LOCA. While the goal of that effort was related to code uncertainty evaluations, the principles derived to achieve that goal involved the entire process of evaluation model development and assessment. This process was used as a basis for the definition of the EMDAP in DG-1120.

The basic principles of evaluation model development and assessment that constitute an EMDAP are not discussed here in detail, but some consideration that apply to this effort are provided:

1. **Determine requirements for the evaluation model**. This preliminary step is to define the requirements that will guide the development of the evaluation model. The fundamental outcome of this step is the identification of the relevant physical phenomena and their ranking in a PIRT (phenomena identification and ranking table). Quoting from DG-1120: "*the phenomena assessment process is central to ensuring that the evaluation model can analyze the particular event appropriately and that the validation process addresses key phenomena for that event.*"

2. **Develop an assessment base consistent with the determined requirements**. Once the key physical phenomena that the evaluation model must address are identified, the next step consists in determining an appropriate assessment data base. Quoting again from DG-1120: "*since an evaluation model can only approximate physical behavior for postulated events, it is important to validate the calculational devices, individually and collectively, using an appropriate assessment base. The data base may consist of already existing experiments or new experiments may be required for model assessment, depending on the results of the required determination.*"

3. **Develop the evaluation model**. Only at this point the evaluation model development should be initiated, so to provide adequate capability with regards to the requirements defined in the first step.

4. **Assess the adequacy of the evaluation model**. The effective adequacy of the evaluation model developed in step 3 should be assessed against the assessment data base defined in step 2.

5. **Follow an appropriate quality assurance protocol during the EMDAP**. No comment is provided here on this element.

6. **Provide comprehensive, accurate, up-to-date documentation**. No comment is provided here on this element.

During the initial IRIS assessment it was determined that an additional propaedeutic step was required for IRIS SBLOCA analysis. This is due to the novel design of the emergency core cooling system that leads to an accident evolution that is different from that of current pressurized water reactors. The PIRT development relies mostly on expert-based considerations, which in turn requires that the team of experts that develops the PIRT have a thorough and complete understanding of the expected system response during the considered events. While this can be easily anticipated for the development of evaluation models for current plant designs (in fact the EMDAP originated from the CSAU program for the development and assessment of best estimate LBLOCA), the situation is different for novel plant designs such as IRIS. Development of an appropriate PIRT for IRIS requires that enough information is provided to characterize the transient evolution, thus the analyses presented in this report have been developed.

The approach used has been to define a preliminary evaluation model on the basis of the expected system behavior. This has been done first qualitatively with an expert based assessment of the IRIS response to a SBLOCA and of the system evolution during the event (this qualitative assessment has been discussed in Section 1.2.1 of this document). Based on this assessment a preliminary approach has been developed for the quantitative analysis of this event. The main purpose of this model was not to provide a safety assessment for IRIS, nor was it developed considering it a complete evaluation model. Rather, the objective was to improve the understanding of the IRIS response to SBLOCA events, and to allow an optimization of the mitigation strategy.

Given the purpose of these analyses, the evaluation of a "nominal" scenario has been performed. This means that in this assessment no consideration of uncertainties has been provided, except in the treatment of some parameters for which the conservative direction was readily apparent: this provides a reasonable assessment of the system response and especially is suited to the purpose of developing the analyses to understand the system, rather than to evaluate it to show how the acceptance criteria are satisfied.

The results presented in this report will be submitted to the team of experts that will take part to the PIRT development for IRIS, eventually supported by additional sensitivity analyses as required. The next step will thus be the development of a PIRT for IRIS, which will be completed and submitted to the NRC. The PIRT will then be reviewed to define an appropriate assessment database. This will identify those areas where additional testing is required to complete the assessment base, and therefore the final outcome of this activity will be the definition of a test program for IRIS.

2.6.4.2 Overview of preliminary analysis approach

Based on a preliminary, qualitative assessment of the relevant phenomena during an IRIS SBLOCA scenario, an evaluation model for a preliminary SBLOCA assessment was developed by the University of Zagreb and used to perform SBLOCA analyses. This model has been presented in several open literature papers (Reference 3 and 4), and is only briefly discussed here. All IRIS SBLOCA analysis and studies have been performed at the University of Zagreb, with support from Westinghouse mostly in the area of results interpretation and modeling assumptions.

The IRIS novel LOCA safety approach poses some new issues for computational and analysis methods since the IRIS integral reactor coolant system and containment are strongly coupled, and the system response is based on this interaction. A preliminary assessment has led to the conclusion that in order to develop an appropriate evaluation model for the IRIS SBLOCA, the containment/vessel coupling had to be correctly captured.

For this reason a coupled RELAP/GOTHIC model was developed by the University of Zagreb. A simple direct coupling of RELAP5 and GOTHIC was used with connections at the points of hydraulic contact (the break, ADS, and gravity makeup flow paths). The connections are comprised of a time dependent volume component on the RELAP5 side and a flow boundary condition on GOTHIC side. The existing detailed RELAP5 model of the reactor coolant system and of the engineered safety features is used for these analyses, together with a simplified GOTHIC model of the containment.

The details of the coupled model are not discussed here, since the focus is not on the specific features of the model, but rather on the preliminary analyses of the system. However, to provide an adequate understanding of the accident sequence, some additional consideration on the reactor coolant system and containment models are in order. The reactor coolant system model with RELAP5 is provided in Figure 2.6-1, while the simplified containment model with GOTHIC is shown in Figure 2.6-2.

The IRIS integral reactor coolant system nodalization for RELAP5 can be divided in the following main regions (refer to Figure 2.6-1) and component volumes (CVs):

- Lower downcomer (CV101)
- Lower plenum (CV102, CV103, CV104, CV105, CV106),
- Core and bypass region (CV110, CV115),
- Riser (CV120, CV121, CV122, CV 123, CV124, CV141 to 148),
- Pressurizer (CV130),
- Upper downcomer and Reactor Coolant Pump Suction Plenum (CV125, CV150, CV151),
- Reactor coolant pumps (RCP) (CV191 to 198),
- Primary side of the Steam Generator (SG) modules (CV201 to 208, CV211 to 218, CV221 to 228),
- Inactive volume around the SG modules (CV240), inactive volume inside the SG module central support column (241-248).
- Steam Generator Shroud check-valves. (CV161 to 168)

Each of the eight RCP/SG modules is explicitly modeled: this detailed nodalization of the coupled RCP and SG modules was selected in order to better address potential asymmetrical effects in the coolant system. This also addresses the interaction of SG modules and EHRS loops (asymmetry due to different length of feed and steam lines) to preclude possible artificial recirculation in parallel loops introduced due to lumping of multiple SGs and associated numerical effects.

The secondary system and the balance of plant are only modeled in detail up to the main feed and steam isolation valves. A simplified turbine and feedwater system is also provided. Two SG modules are connected to each feed/steam line, so that the system features four steam and four feed lines.

The Secondary Side of the SG modules is modeled by CV251 to 258, CV261 to 268, CV271 to 278, CV281 to 288, and CV291 to298. For SG1 the helical coil bundle is represented by CV271, the lower (feedwater) header by CV 251 and 261, and the upper (steam) header by CV 281 and 291. SG2 to 8 are modeled in a similar pattern.

Finally, the Engineered Safeguards Features of the plant are also modeled. Among these, the ones included in the RELAP5 model are the emergency heat removal system (EHRS) and the emergency boration system (EBS). Also, the refueling water storage tank (RWST) is modeled as the ultimate heat sink for the EHRS heat exchangers. These systems are sufficient for the analysis of all IRIS NON-LOCA transients and accidents. Since the remaining IRIS safety features (automatic depressurization system, ADS; pressure suppression system; long term core makeup system) establish an interaction between the integral reactor coolant system and the containment, their models are only included in the coupled RELAP/GOTHIC code nodalization. These systems are represented with simplified models in this nodalization.

The simplified GOTHIC containment model (Figure 2.6-2) shows six points where the reactor coolant system and containment are connected. Node 1F represents the break, and can be moved to different positions in the reactor coolant system. Connection 6F represents the ADS sparger in the suppression pool. Connections 2F and 3F represent the lines connecting the suppression pool water volumes to the direct vessel injection line. These lines provide water makeup from the suppression pool to the vessel once the containment and vessel pressure are equalized. Connections 4F and 5F represent the lines connecting the reactor cavity, which acts as a containment sump, to the reactor coolant system. These lines provide a source of makeup water from the containment to the reactor coolant system should the coolant level in the reactor coolant system drop below the water level in the reactor vessel cavity with the containment and vessel pressures equalized.

The containment model assumes perfect mixing, and includes preliminary estimates of the structural materials present in the containment. Also, the suppression pool water volume and the non-condensable gas storage volume located at a lower elevation are included in the model.

2.6.4.3 Preliminary SBLOCA Assessment for IRIS

Two limiting SBLOCA events have been preliminarily identified for IRIS. One is the complete rupture of a chemical volume and control system (CVCS) pipe, connected to the upper annular pump suction plenum of the IRIS reactor vessel, and which has been designed as a 4", schedule 160 pipe. To conservatively account for possible design changes, a pipe with a 4" internal diameter, connected to the lowest elevation in the pump suction plenum has been considered. The second limiting break is the double ended rupture of the direct vessel injection (DVI) line located in the lower annular region surrounding the steam generators. This line is connected to the reactor coolant system about 2 meters above the top of the upper core plate. The DVI is a 2", schedule 160 pipe. To conservatively account for possible design changes, a pipe with a 2" internal diameter, connected to the reactor coolant system about 1 m above the core has been considered in the analysis. This break is located relatively near to the top of the core, and thus will lead to a larger loss of coolant inventory from the reactor coolant system.

Preliminary results for both events are discussed in the following sections.

2.6.4.3.1 4" Break in the Chemical Volume and Control System

A sequence of events for the 4" break in the CVCS is provided in Table 2.6-1 and results are provided in Figures 2.6-3 to 2.6-15. Note that, given the objective of these analyses, some additional sensitivity cases beyond the base, reference case are performed to support the investigation.

A SBLOCA for IRIS can be divided in three distinct phases. First, the blow-down phase is defined as the period during which the reactor coolant system pressure is reduced and the containment pressure increases until the reactor coolant system and containment pressures equalize. This phase, due to the relatively small dimensions of the break and of the ADS lines, can be long for IRIS, in the order of 1600 seconds for the considered case. 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 considered concluded when the pressures in the vessel and in the containment are in equilibrium.

The blow down phase is followed by a depressurization phase. During this phase the containment and reactor vessel 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 and that are discussed later in detail (flooding of the cavity from the suppression pool, makeup of coolant from the suppression pool to the vessel). This phase has been preliminarily and tentatively defined as lasting until the containment pressure is below 2-3 barg (29 to 43.5 psig).

Finally, a long term cooling phase is established. This phase covers the long term cooling of the coupled containment and vessel 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. This phase has not been analyzed in this study. The lack of very long term results is not considered a significant limit to the current analyses and should not negatively impact the development of a PIRT.

Thus, only the first phase (blow-down) and part of the second phase (depressurization) are analyzed here.

Following the initiating event (a complete break of a CVCS piping connection), the LM (LOCA Mitigation) signal is rapidly actuated on a coincident low pressurizer pressure and high containment pressure. On a LM signal the following actions are initiated:

1. Containment penetrations are isolated,
2. 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,

3. The ADS is actuated and EBT discharge isolation valves are actuated.

During this blow-down phase, the reactor coolant system pressure (Figure 2.6-3) is rapidly reduced due to the loss of mass at the break and at the ADS, and due to heat removal by the EHRS. The rate at which pressure decreases in the reactor coolant system is initially rapid, slows when the coolant in the reactor coolant system reaches saturation conditions, and then increases again when the mixture level drops below the break position. Once the pressure in the reactor coolant system and in the containment (end of the blow-down phase) are equalized, the reactor coolant system 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. The total break flow rate is provided in Figure 2.6-4, which shows that the break flow rate during the blow-down phase is consistent with the reactor coolant system and containment pressure profile. The default RELAP critical flow model (Henry-Fauske) with a discharge coefficient of 1.0 is used to calculate critical flow in this analysis.

It should be noted that just following pressure equalization between the reactor vessel and containment vessel, the break flow rate becomes negative, meaning that the flow is from the containment to the vessel. This is due to the unique IRIS approach of providing heat removal inside the reactor vessel and thus reactor coolant system depressurization by the EHRS. Figure 2.6-5 shows the core thermal power (decay heat is calculated with the ANS-79 standard + 2 sigma) and the heat removal via the steam generators. Since more heat is always removed by the EHRS via the steam generators than is produced in the core, the EHRS depressurizes the reactor coolant system. Also, once the reactor coolant system and containment reach an equilibrium condition, the EHRS will continue to remove heat and depressurize the reactor coolant system, and thus the net flow is from the containment through the break and ADS to the reactor coolant system.

The containment pressure is shown in Figure 2.6-3. The containment pressure increases during the first phase of the transient, until the containment and vessel pressure equalize thus terminating the break flow from the vessel and marking the end of the blow-down phase. The containment pressure increase during the blowdown phase is limited by the condensation of the steam released from the vessel to the containment in the suppression pool water. Also, the steam released from the reactor coolant system through the ADS sparger is directly condensed in the suppression pool (see additional discussion in Section 2.6.4.3.2).

Once the reactor coolant system and containment pressures are equalized (end of the blowdown phase), the containment is coupled to the reactor coolant system and the containment pressure follows the same pressure transient as the reactor coolant system pressure decreases. The suppression pool atmosphere is not however coupled to the rest of the containment and to the reactor coolant system. This is due to the fact that the suppression pool design in IRIS that does not feature any vacuum-breaker as in conventional BWRs. As the containment pressure decreases and the suppression pool pressure remains essentially constant, the water level rises in the pool inlet piping connections from the containment due to this differential pressure, until the suppression pool water is finally discharged to the containment through the vents. With the vents working essentially in “reverse mode”, water from the suppression pool will flood the reactor cavity. This is a relatively important phenomenon in the depressurization phase and sensitivity analyses have shown that assumptions relative to this flow can have an effect on the containment pressure transient in this phase (the containment pressure is reduced if the flow from the suppression pool vent piping is assumed to be in the form of water droplets, while the minimum effect on containment pressure was verified to occur if a simple relocation of the water to the cavity without any heat transfer with the containment atmosphere is assumed). The flow of water from the suppression pool will in fact change the depressurization rate of the containment as shown by a change in slope of the depressurization rate shortly after the beginning of the depressurization phase.

Figure 2.6-6 shows the liquid levels in various volumes of the containment and can be used to better understand the time scale of this phenomenon. During the initial part of the transient the water level in the suppression pool inlet piping from the containment atmosphere is rapidly reduced as the containment pressurizes, and the vents begin discharging steam and non-condensable gas to the suppression pool. During the blow-down phase the level in the vent piping remains at the bottom of the vent, as the steam/gas mixture continues to be pushed from the containment into the suppression pool. At the end of the blowdown phase, when the containment starts to depressurize, the opposite effect occurs, as the vent piping slowly fills up with water until the water fills the vent piping and water is discharged to the containment. This discharge lasts until the water level in the suppression pool is sufficiently reduced to uncover the vent spargers. Once the spargers are uncovered, the suppression pool pressure equalizes with the containment pressure. The suppression pool level slowly continues to decrease as some flow is provided to the reactor coolant system through the makeup lines from the bottom of the suppression pool water volume.

Figures 2.6-6 also shows that the reactor cavity is partially flooded during the early part of the blowdown phase by the liquid discharged at the break. This initial flooding stops once the break uncovers and flow through the break becomes only steam. The cavity flooding is then completed in the depressurization phase, when the water from the suppression pool completely floods the cavity.

As discussed above, the suppression pool water level is located at a higher elevation than the reactor 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 reactor coolant system. Following the actuation of the valves on the makeup line that connects the suppression pool to the direct vessel injection line, water from the suppression pool will be driven by this gravitational head and pressure differential to the reactor coolant system. As shown in Figure 2.6-7, the liquid mass in the reactor coolant system stabilizes at the end of the blow-down phase and then increases during the depressurization phase. This is due to the injection of coolant from the suppression pool.

Figures 2.6-8 through 2.6-10 provide an indication of the water mixture level in the reactor vessel. Figure 2.6-8 shows the void fraction at the core exit. Figure 2.6-9 shows the liquid fraction in the upper riser region (Region 123 from Figure 2.6-1). The figures show that a large liquid fraction is always present in the core to remove decay heat. The mixture level can be estimated at the position of the sharp change in void fraction in the riser nodal volumes. From Figure 2.6-8 it can be inferred that the mixture level never drops below control volume 123-06 (the bottom 8 cells of component volume 123 represent the upper riser in the IRIS nodalization shown in Figure 2.6-1), which is well above the top of active fuel region. Figure 2.6-10 shows the collapsed liquid level in the reactor vessel, which is maintained above the top of the active core for the whole duration of the transient.

Some additional analyses were also performed to provide additional insights on the IRIS SBLOCA transient evolution, and to provide a basis to optimize the LOCA mitigation strategy. These analyses included evaluations to understand the oscillations in core exit void fraction (Figure 2.6-8), and an assessment of the importance of the heat capacity of the containment structures.

First, the oscillation in core exit void fraction and other local parameters are discussed. To understand this behavior, some additional information on the IRIS design is required.

During the initial part of the transient, the mixture level in the reactor coolant system rapidly decreases, until the pump suction in the upper part of the vessel is uncovered. At this point, single phase natural circulation would be impeded and the only circulation would be due to the steam produced in the core being condensed on the cooled steam generator tubes. To maintain single-phase circulation in the reactor vessel, the IRIS steam generators are provided with a set of specially designed check-valves mounted into each steam generator shroud (see the IRIS Plant Description Document, Reference 5). These check valves are designed to open when the RCPs are not operating; i.e., when the pressure of the primary fluid in the steam generator tube bundles is not higher than the pressure in the surrounding RV annular space. These valves open and maintain a single phase natural circulation path through the core, to the riser, into the annular space between the steam generators, into the steam generator tube bundle, and back to the downcomer, even when the pump suction is uncovered. The main purpose of these valves is to provide a natural circulation path when the reactor coolant shrinks in response to cooldown transients and during cooldown of the reactor to the safe shutdown conditions. By maintaining a path for single phase natural circulation, boiling in the core is prevented, since heat can be removed through the single phase natural circulation loop. In the reference IRIS design these check-valves are located at an elevation of about 2/3 up the overall steam generator height.

During a LOCA, single phase natural circulation is maintained until the mixture level drops below the steam generator shroud check-valves. If/when the level drops below these valves, the single-phase natural circulation flow path is interrupted. The oscillations shown in Figures 2.6-5 through 2.6-9 are due to this termination of single-phase natural circulation, and are in fact coincident with the time at which the water level drops below the check valve location. The oscillations disappear whenever the reactor vessel liquid mixture increases above the steam generator shroud check-valves. To confirm this interpretation, an additional case was analyzed where the steam generator check-valves and riser flow holes were moved to a lower elevation. Figure 2.6-11 shows the void fraction at the core exit for both low and high check-valve position (indicated as high and low RSC, riser-steam generator connections). With the lower elevation check-valve position, the oscillatory behavior disappears as expected. Figure 2.6-12 compares the liquid mass and the collapsed liquid level, showing that these numerical oscillations do not have a significant impact on the evolution of the main transient parameters. Figure 2.6-13 compares the pressure transient that occurs with the two different check-valves positions.

Note that although the impact on the reactor vessel water inventory is limited, maintaining single phase circulation tends to improve the transient response, by delivering subcooled water to the core and limiting boiling in the core, especially in the depressurization part of the transient. For this reason, a design change is currently being considered that would lower the position of the steam generator check-valves so as to maintain single-phase natural circulation for as long as possible following all design basis events.

Another analysis was performed to evaluate the impact of the containment heat structures on the transient response. The simplified model used to provide the previous results did not assume heat transfer from the containment atmosphere to the containment structures and to the environment. The effect of considering the thermal inertia in the containment structures will be to reduce the initial pressurization in the containment and thus lead to a lower containment peak pressure. On the other hand it is expected that in IRIS, the reactor coolant system inventory will be affected negatively by a lower containment pressure, since this will tend to increase the duration of the blow-down phase. It is therefore expected that the IRIS evaluation model for SBLOCA will have to include two different sets of analyses, one with assumptions that maximize the pressurization of the containment (a containment design pressurization case), and one with assumptions that minimize the containment pressure (a minimum reactor coolant system inventory case). Figure 2.6-14 shows the containment pressure and reactor coolant system pressure for the two cases, with and without heat structures in the containment. As expected, the peak pressure is lower for the case that assumes heat structures, although only a small increase in the duration of the blow-down phase is evident. As a consequence, Figure 2.6-15 show that only a minor difference exists in the collapsed liquid level for the two assumptions. This would seem to suggest that modeling of the containment heat structures does not significantly impact the overall evolution of the transient. Some additional observations and conclusions will be provided in Section 2.6.4.3.2.

This preliminary assessment should provide a sufficient overview of the IRIS SBLOCA sequence for a break at high elevation on the reactor vessel and will be used to provide initial information to the PIRT development team.

2.6.4.3.2 2" Double Ended Break in the DVI Line

The second limiting break considered in this analysis is the guillotine rupture of the direct vessel injection (DVI) line. Although this is a smaller line than the 4" CVCS line analyzed in section 2.6.4.3.2, it requires a separate analysis due to its position nearer to the top of the active core. The DVI line is connected to the reactor coolant system in the annular region near the bottom of the steam generators. In the current design a 2", schedule 160 pipe is used for the DVI. To cover eventual design changes, a pipe with an internal diameter of 2" is considered here.

Only cases for a double ended rupture have been presented in this section. An additional set of analyses for a split break was also performed to verify that the double ended break was more limiting from the point of view of liquid level and containment pressurization.

A sequence of events for the 2" break in the CVCS is provided in Table 2.6-1 and results are provided in Figure 2.6-16 to 2.6-31. The transient evolution is similar to the one discussed for the 4" break, but some important differences exist and are discussed here in some detail. In particular, evaluations of the results obtained from these analyses have suggested some minor design modifications.

Only results for a double ended rupture of the DVI line are presented here. A split break of the DVI line was also analyzed. The double ended break was verified to be conservative from the point of view of liquid level and containment pressurization, and therefore only these results are presented here. A more complete spectrum of break sizes and location will be explored in the development of IRIS evaluation models.

The base case analysis has assumptions similar to the base case for the 4" break in the CVCS line discussed in the previous section. As it will be discussed, critical assumptions in this analysis include the layout of the ADS system and neglecting the heat structures in the containment in the reference case.

Due to the smaller size of the break and the fact that the break discharges a low quality mixture for the whole duration of the blow-down phase after saturation conditions are reached in the reactor coolant system (i.e. the break level is always below the mixture level in the reactor coolant system), the duration of this phase is significantly longer than in the case of the 4" CVCS break. Also, the containment pressurization rate is milder and the peak pressure is lower (Figure 2.6-16) than in the case of the 4" break. The reason for the reduced containment impact is that the energy release rate at the break is reduced which leads to a slower relocation of the non-condensable gas to the suppression pool, and the steam discharged by the ADS to the suppression pool is suppressed more efficiently. Due to the slower depressurization rate in the reactor coolant system and lower pressurization of the containment, the duration of the blowdown phase is increased from 1600 seconds (following the 4" CVCS break) to almost 3000 seconds. During this transient the reactor coolant system and the containment coupling is delayed, since the break is always below the reactor vessel mixture level and since the cavity floods to cover the break on the containment side. Thus there is no connection between the containment atmosphere and the vessel atmosphere. As the reactor vessel pressure decreases and containment vessel pressures increases, they equalize, and then the reactor coolant system pressure decreases due to condensation of steam inside the reactor vessel on the steam generator tube surface cooled by the EHRS.

As the reactor coolant system pressure is decreased, the higher pressure in the suppression pool will force water to the reactor coolant system through the makeup lines from the pool to the reactor vessel. Eventually water also flows from the suppression pool to the reactor vessel through the ADS line. The suppression pool water level slowly decreases as fluid mass is provided to the vessel. As the suppression pool atmosphere is depressurized, the containment will be depressurized as steam flows into and is condensed in the suppression pool. This situation will be maintained until the liquid level in the suppression pool decreases, due to mass addition to the reactor coolant system, sufficiently to uncover the vent spargers, at which time the atmosphere of the containment, suppression pool and reactor vessel are in communication. The location of the ADS sparger in the suppression pool has a significant impact on this physical sequence and alternative ADS piping layouts that will allow a faster communication between the reactor vessel, containment vessel and suppression pool are discussed later.

This interpretation of the events following the 2” DVI line break is confirmed by Figure 2.6-19, which shows the liquid level in relevant volumes in the containment. Note that in this case the cavity is flooded by water from the reactor coolant system, rather than with water from the suppression pool. The reactor cavity is designed in such a way that all low level penetrations (i.e. the DVI lines) and associated valves are located in rooms connected to the cavity, so that a break in these lines results in an automatic flooding of the cavity.

The break flow rate is shown in Figure 2.6-17. Note that beside the overall break flow rate, the contribution from each side of the double ended break is also shown (Break1 in the Figure refers to the vessel-side, while Break2 refers to the EBS side).

Parameters that provide an evaluation of the mixture level in the reactor coolant system and of the amount of fluid available for core cooling are provided in Figures 2.6-20 to 2.6-23. From Figure 2.6-22 it can be inferred that the mixture level never drops below control volume 120-12 (the top cells of component volume 120 represent the top region of the lower riser in the IRIS nodalization shown in Figure 2.6-1), which is several meters above the top of active fuel region. As shown by the collapsed reactor vessel liquid level in Figure 2.6-23, a large coolant inventory is available throughout the transient to assure that the core will remain covered and be effectively cooled.

Some additional sensitivity analyses have been performed for this 2” DVI break case. As discussed in the previous section, the oscillations in the local void fraction at the top of the core during a portion of the transient are due to the location of the steam generator check-valves. Figures 2.6-24 and 2.6-26 compare the transient evolution for different positions of the check-valves. The same conclusions discussed in the previous section remain valid.

A more critical item is the modeling of heat structures in the containment. As discussed in the previous section, the simplified base case model used for the previous 4” CVCS break results does not assume heat transfer from the containment atmosphere to the containment structures. The effect of considering the thermal inertia in the containment structures was to reduce the containment pressurization rate and containment peak pressure. On the other hand, the IRIS reactor coolant system liquid inventory was negatively affected by the lower containment pressure, since this increased the duration of the blow-down phase. This effect was shown to be relatively minor for the high break, but its impact is significantly larger in this analysis.

Figure 2.6-27 shows the reactor coolant system and containment pressures, without thermal structures, and with the low steam generator check valves (as presented above and shown in Figures 2.6-24 to 2.6-26). The case with thermal structures leads to reduced containment pressurization and to a longer duration of the blow down phase. The effect is significantly larger than for the 4” break analyzed in the previous section, and results in a reduction in the total coolant inventory in the reactor coolant system and in the collapsed liquid level, as shown in Figure 2.6-28. However, Figure 2.6-29 shows that the core void fraction remains below 70%, and that the mixture level is still in the lower riser region, significantly higher than the top of the core. The high water level in the cavity is shown in Figure 2.6-30. The high mixture level in the reactor vessel and the water makeup from the suppression pool provides adequate assurance that acceptable performance (i.e. the core remaining covered) is achieved. However, it is clear that the effect on reactor vessel water level is large and results in a significant penalty on critical parameters including the reactor vessel liquid level. Also, this effect will continue to be important for breaks smaller than a full double ended rupture of the DVI line.

To explain this behavior, the physical evolution of the 2" DVI break case is considered in more detail. Because the break is always below the mixture level in the reactor coolant system, a low quality two phase mixture is released at the break. Given the low energy released through the break (both due to the small break size and to the fact that liquid is discharged) the containment pressurization is much slower than for the high break case, and the effect of the thermal inertia of the containment vessel structures is more important. Also since the ADS is connected directly to the suppression pool, steam released from the reactor coolant system through the ADS is effectively condensed in the pool. This further acts to minimize containment pressurization prolonging break flow into the containment.

The fact that the ADS discharge flow is sparged into the suppression pool water, has a large impact on delaying the pressure equalization between the reactor vessel and the containment vessel. While the current ADS discharge location provides an adequate response and sufficient confidence that the acceptance criteria (core remains covered and containment pressure remains below the design value) are satisfied, the design can still be optimized to improve its performance. In particular, the delay in coupling the reactor vessel and containment that results from the current ADS design is not desirable.

This uncoupled behavior is confirmed by an additional analysis that was performed assuming that the ADS discharges directly to the containment atmosphere. As shown in Figure 2.6-31, with the ADS directly discharging to the containment atmosphere, the pressure in the containment starts increasing when the ADS line is open, leading to a transient response similar to the one for the 4" break analyzed in the previous section. Note that for the 4" break analyzed in Section 2.6.4.3.1, the direct ADS connection to containment does not significantly impact the peak containment pressure, as shown in Figure 2.6-32.

The ADS design is currently being reviewed, specifically, a dedicated quench tank for the ADS is being considered as an alternative option. This tank would be directly connected to the containment atmosphere and with a limited water inventory so that it would lead to a release of steam (and consequent pressurization of the containment) once the quenching liquid is saturated. Note that a direct connection of the ADS to the containment atmosphere is not desirable, since a delay in affecting the containment when/if the ADS is opened should be maintained to minimize the potential for plant impact due to a spurious operation or the use of the ADS in some specific conditions. The final design will be defined before completion of the PIRT.

2.6.5 References

1. "Transient and Accident Analysis Methods", Draft Regulatory Guide (DG) 1120, USNRC, June 21, 2002 (Available in PERR by Accession Number ML003770849).

2. B. Boyack et al., “Quantifying Reactor Safety Margins, Application of Code, Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break, Loss-of- Coolant Accident,” NUREG/CR-5249, USNRC, December 1989.

3. D. Grgic, T. Bajs, L. Oriani, L.E. Conway, “*Development of RELAP5 Nodalization for IRIS Non-LOCA Transient Analyses,*” American Nuclear Society Topical Meeting in Mathematics & Computations (M&C), April 6-10, 2003, Gatlinburg, TN, USA.

4. Oriani, L., L. E. Conway, D. Grgic, T. Bajs, M. E. Ricotti, A. Barroso, “*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.

5. "IRIS Plant Description Document," WCAP-16062-P and NP, March 21, 2003.

Table 2.6-1
TIME SEQUENCE OF EVENTS FOR IRIS SBLOCA

(Sheet 1 of 2)

Accident	Event	Time (seconds)
SBLOCA – 4" Break in CVCS Line Section 2.6.4.3.1	Break occurs	0.0
	Low Pressurizer Level Trip setpoint reached	10.6
	Rods begin to fall into core	12.6
	LM (LOCA Mitigation) signal setpoint reached.	19.7
	EHRS actuation signal generated, ADS valves begin to open	21.7
	Reactor coolant pumps begin to coastdown	25.6
	EBT valve completely open	29.5
	EHRS valves completely open (all four trains)	33.7
	ADS valves completely open	51.7
	Containment and Pressure Vessel pressure equalizes (end of blowdown phase)	~1600
	Injection from the suppression pool to the reactor coolant system begins	~1600
	Suppression Pool vents fill and containment flooding begins	~3000

Table 2.6-1
TIME SEQUENCE OF EVENTS FOR IRIS SBLOCA

(Sheet 2 of 2)

Accident	Event	Time (seconds)
SBLOCA – 2" DE Break in DVI Line Section 2.6.4.3.2	Break occurs	0.0
	Low Pressurizer Level Trip setpoint reached	36.1
	Rods begin to fall into core	38.1
	LM (LOCA Mitigation) signal setpoint reached	50.2
	Reactor coolant pumps begin to coastdown	51.1
	EHRS actuation signal generated, ADS valves begin to open	52.2
	EBT valve completely open	63.2
	EHRS valves completely open (all four trains)	64.2
	ADS valves completely open	82.2
	Injection from the suppression pool to the reactor coolant system begins	~2600
	Containment and Pressure Vessel pressure equalizes (end of blowdown phase)	~2800
	Suppression Pool vents fill and containment flooding begins	N.A.

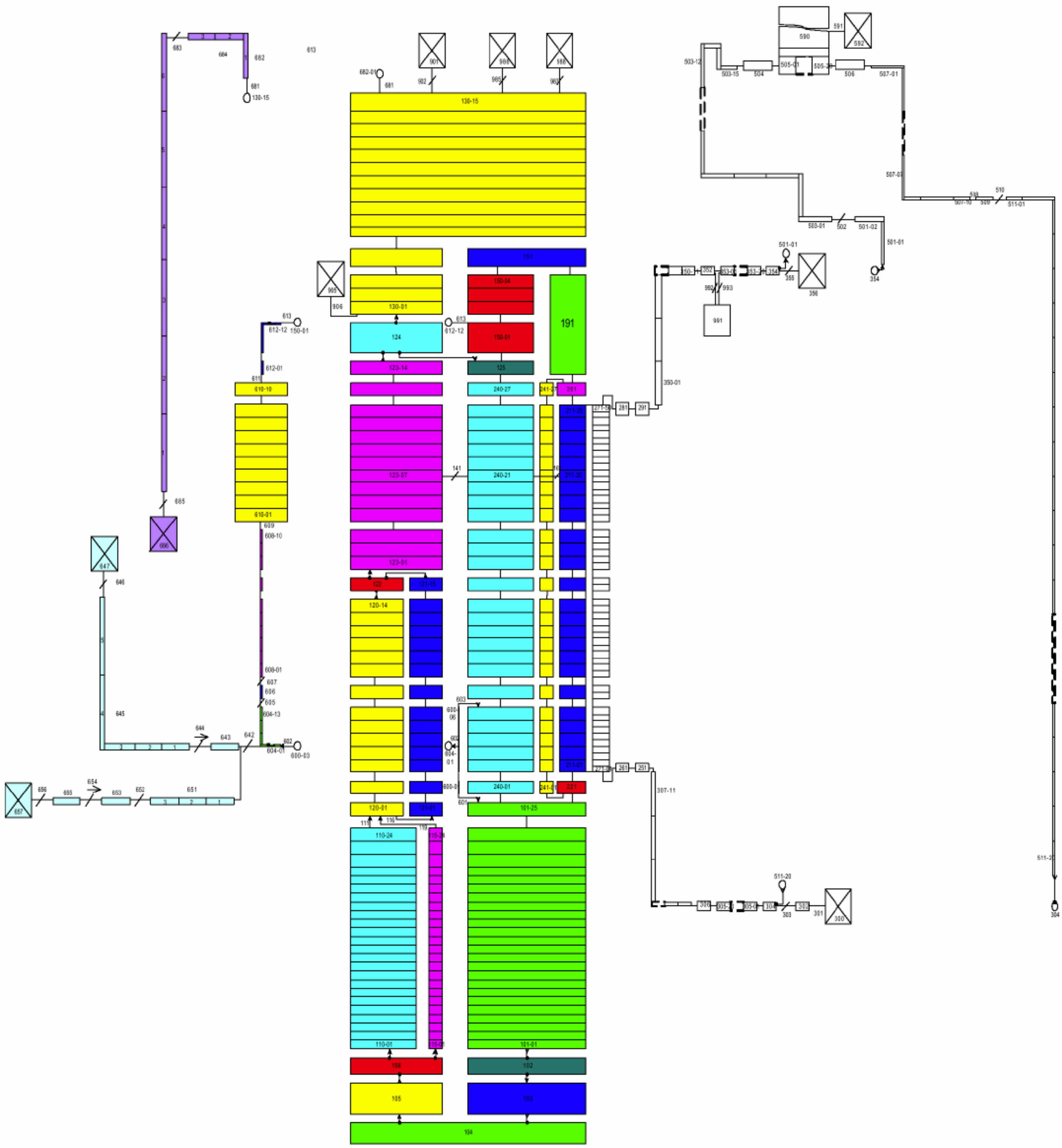


Figure 2.6-1
Reactor Coolant System Model for RELAP5 Used in IRIS Safety Analyses

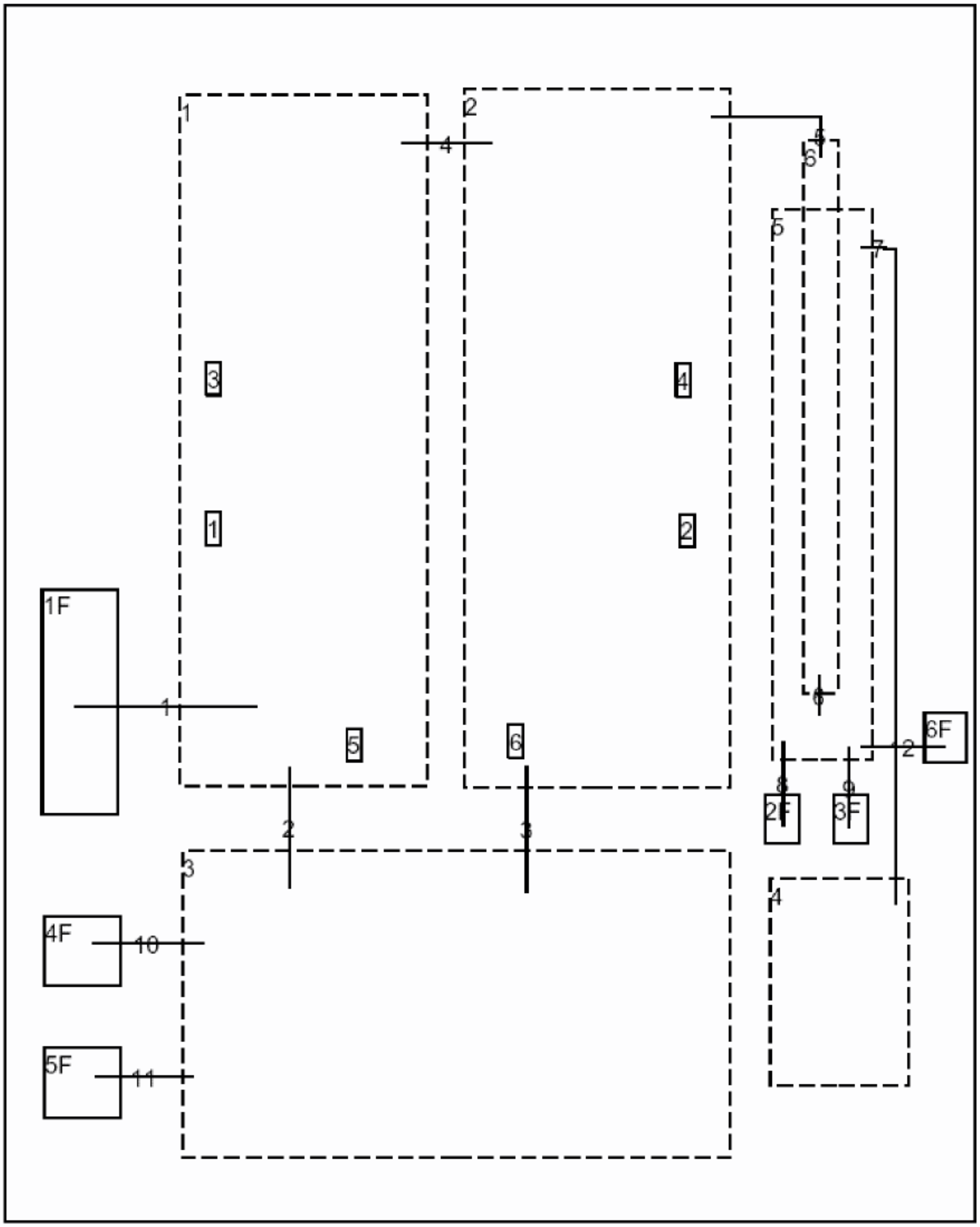
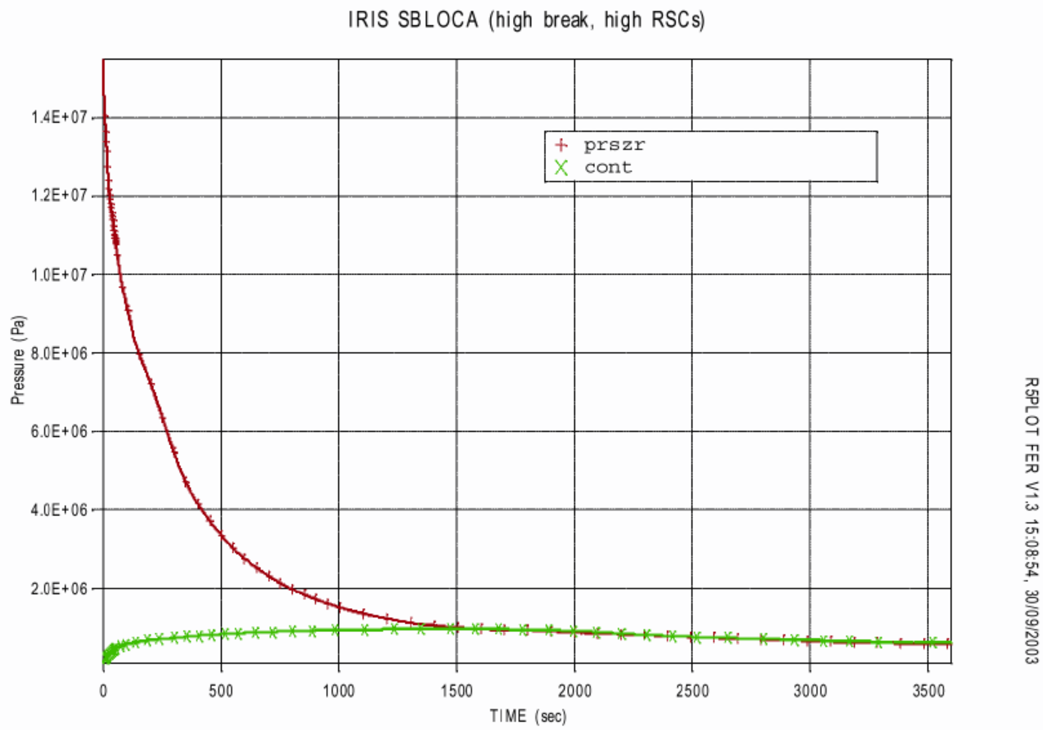


Figure 2.6-2
Simplified Containment Model for GOTHIC 3.4 Used in IRIS Safety Analyses



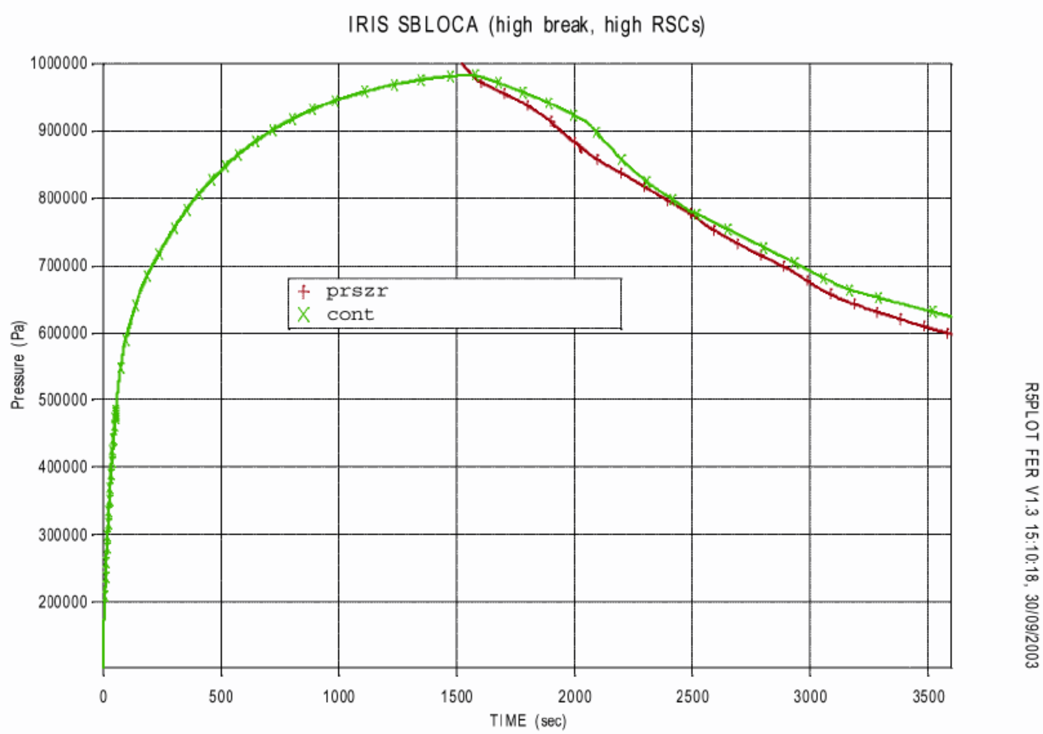
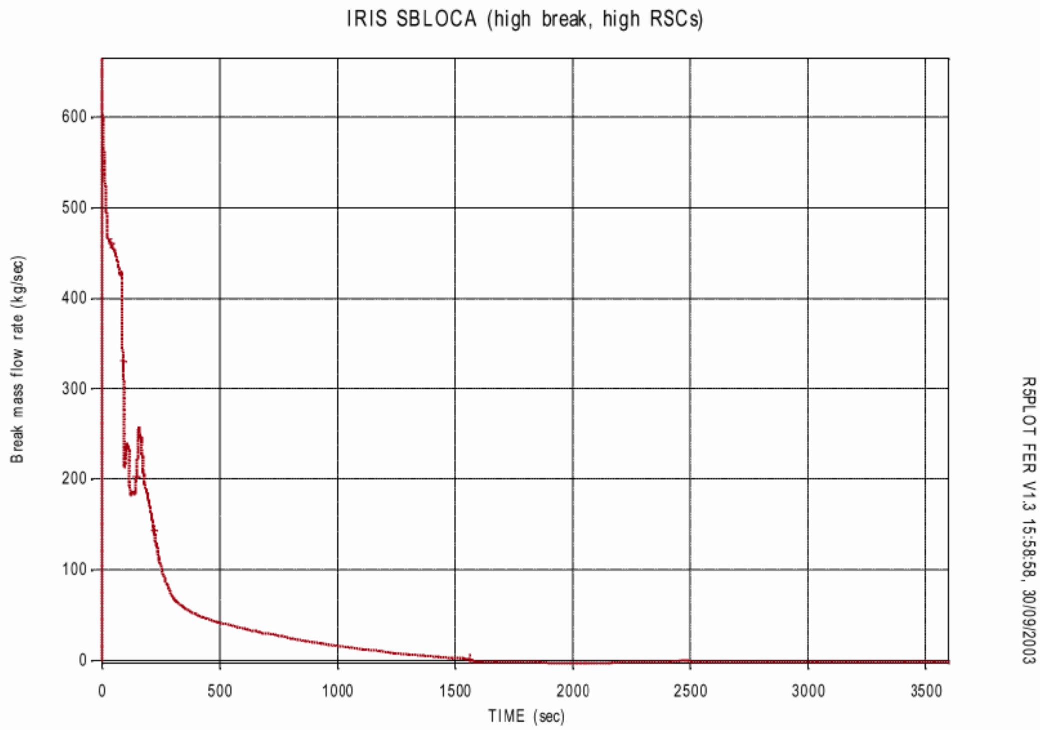


Figure 2.6-3
4" CVCS Line Break - Containment and Pressurizer Pressure Transient (with detail)



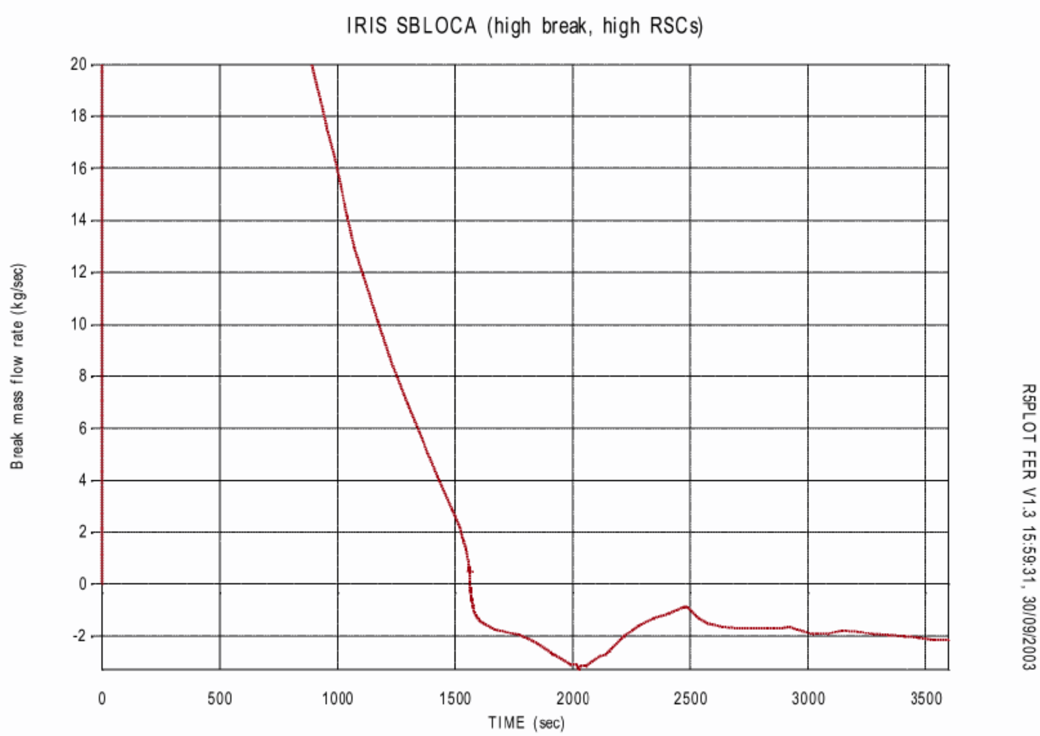
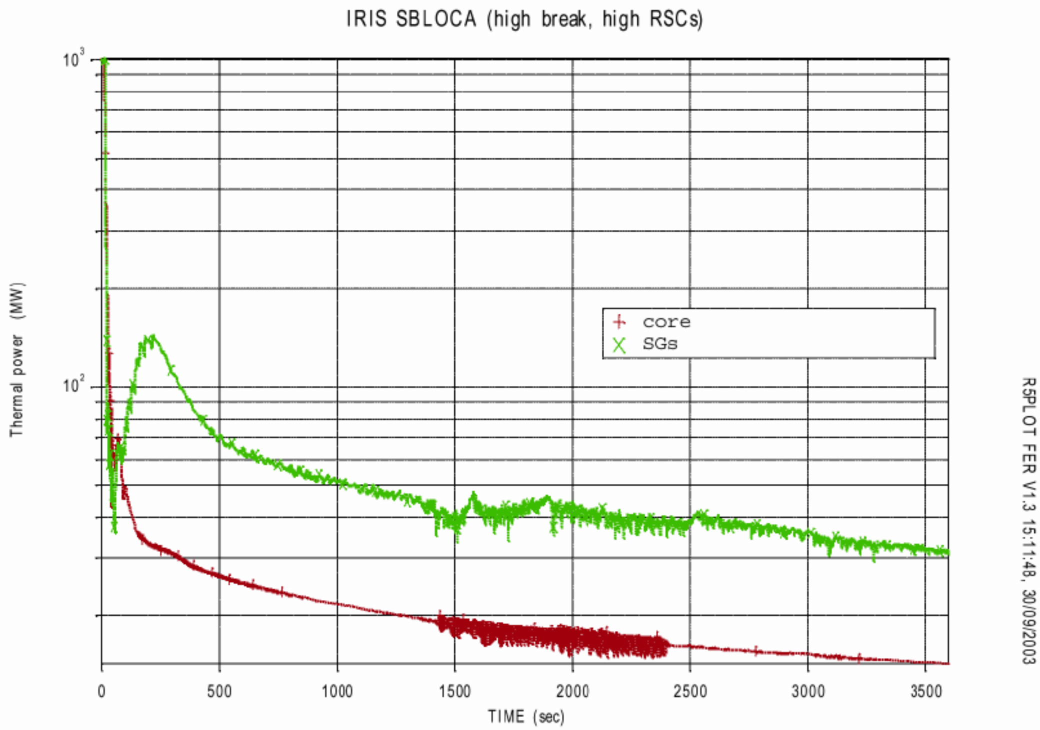


Figure 2.6-4
4" CVCS Line Break – Break Flow Rate (with detail)



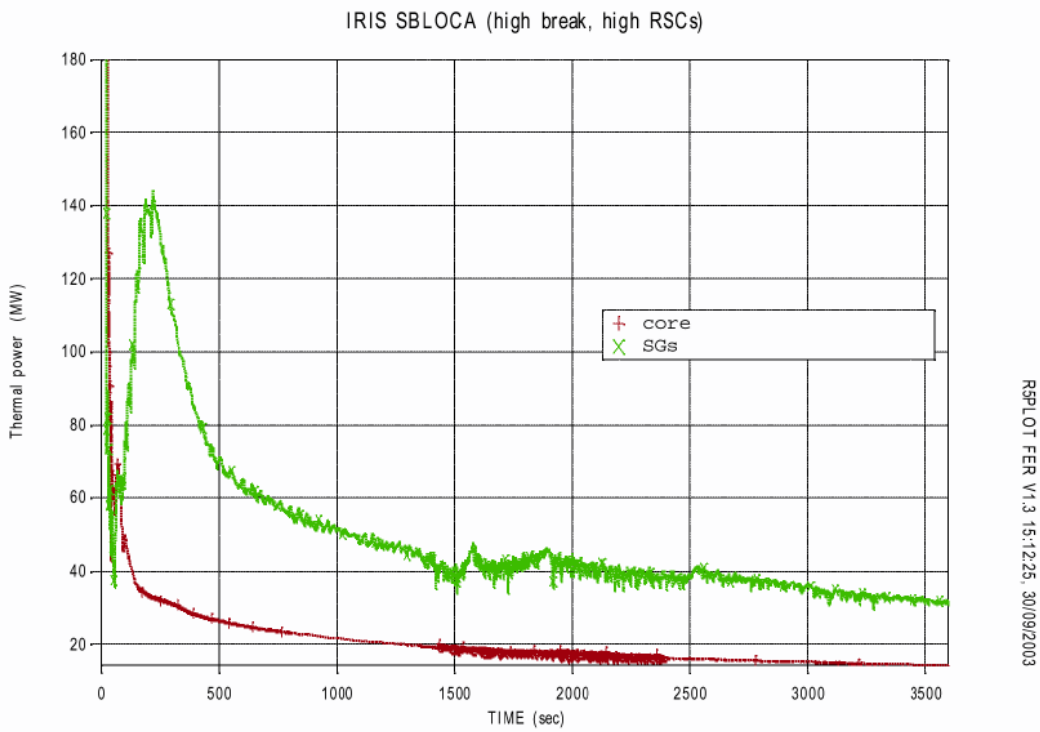


Figure 2.6-5
4" CVCS Line Break – Overall Core and Steam Generator Heat Removal Transient (with detail)

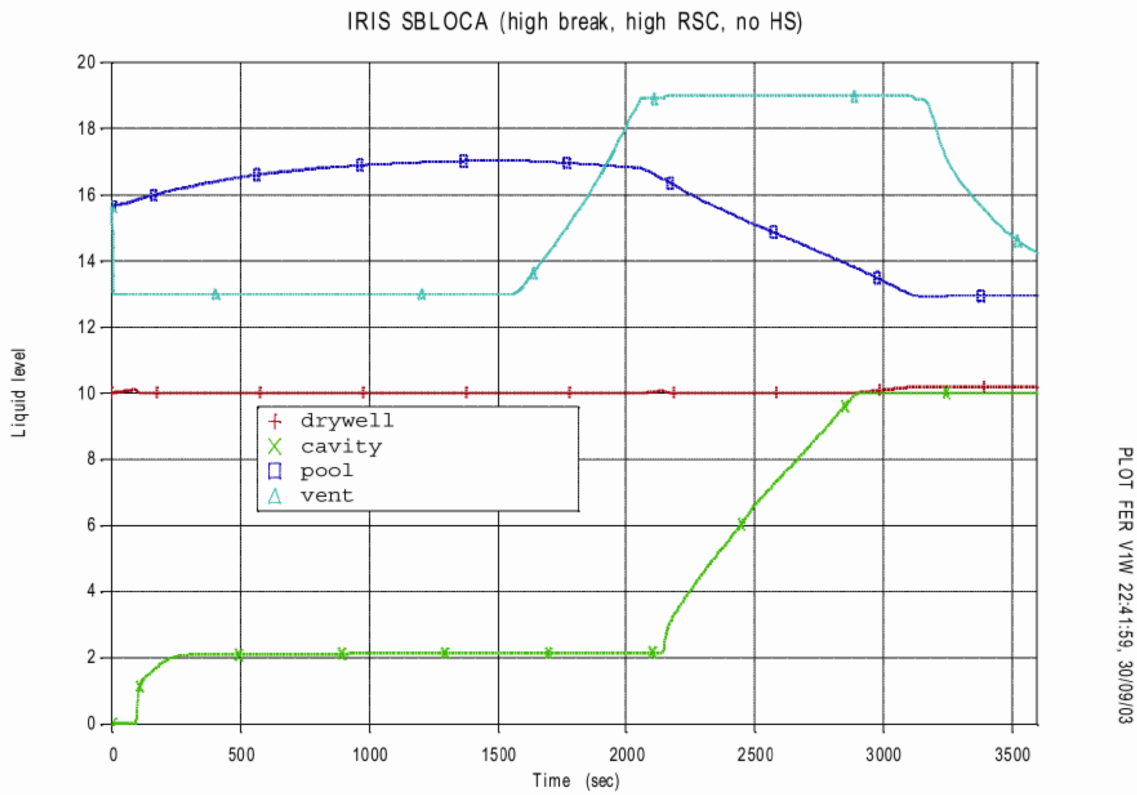
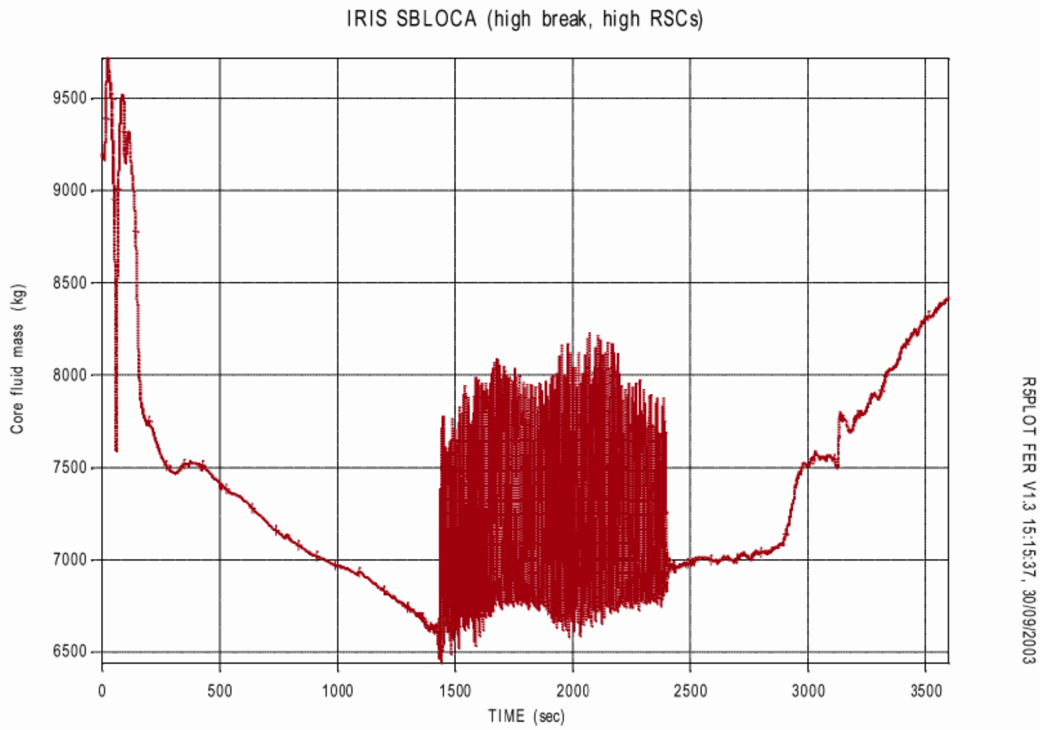


Figure 2.6-6
4" CVCS Line Break – Containment Liquid Mass Transient



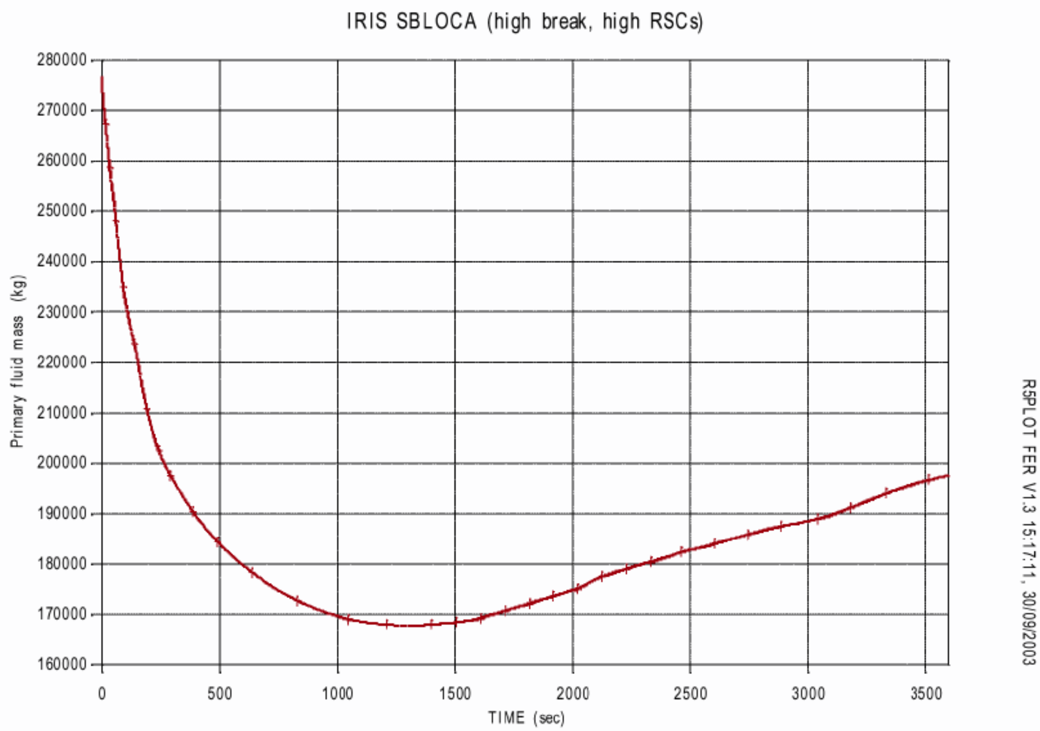


Figure 2.6-7
4" CVCS Line Break –Reactor Coolant System and Core Liquid Mass Transient

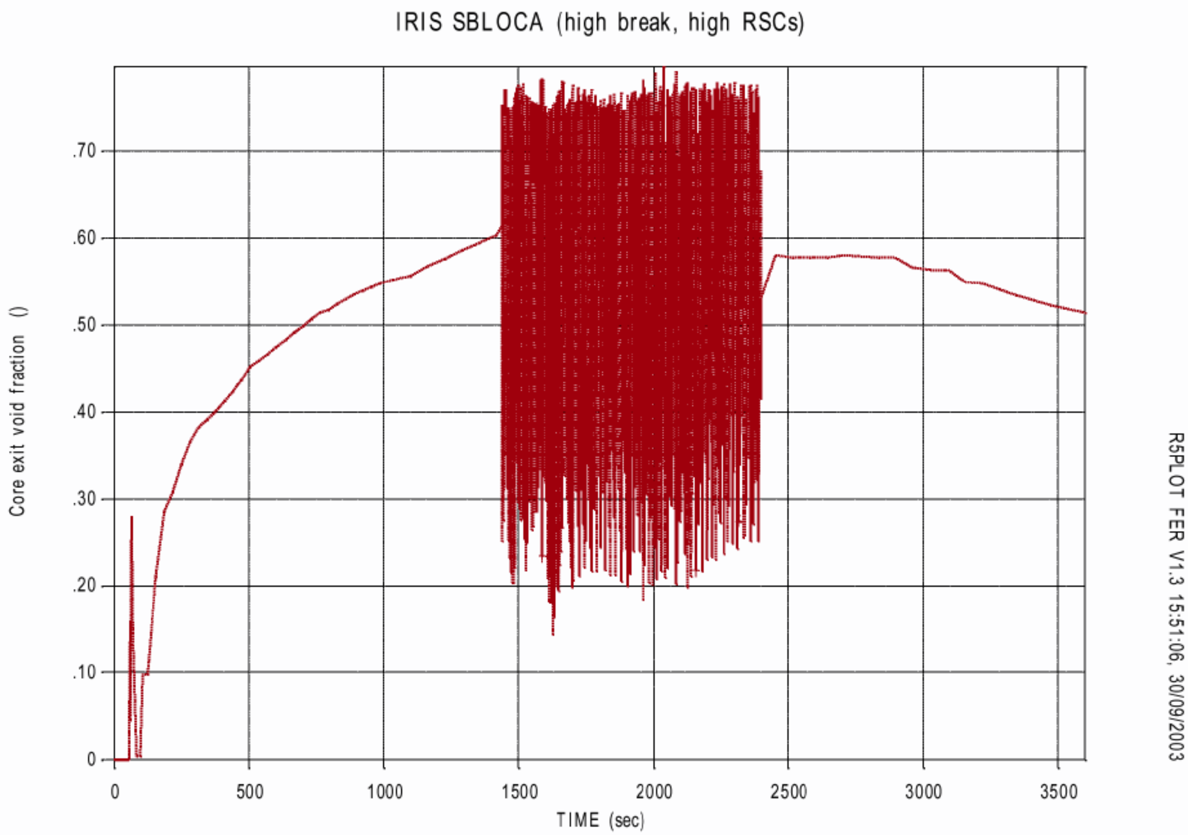


Figure 2.6-8
4" CVCS Line Break – Core Exit Void Fraction Transient

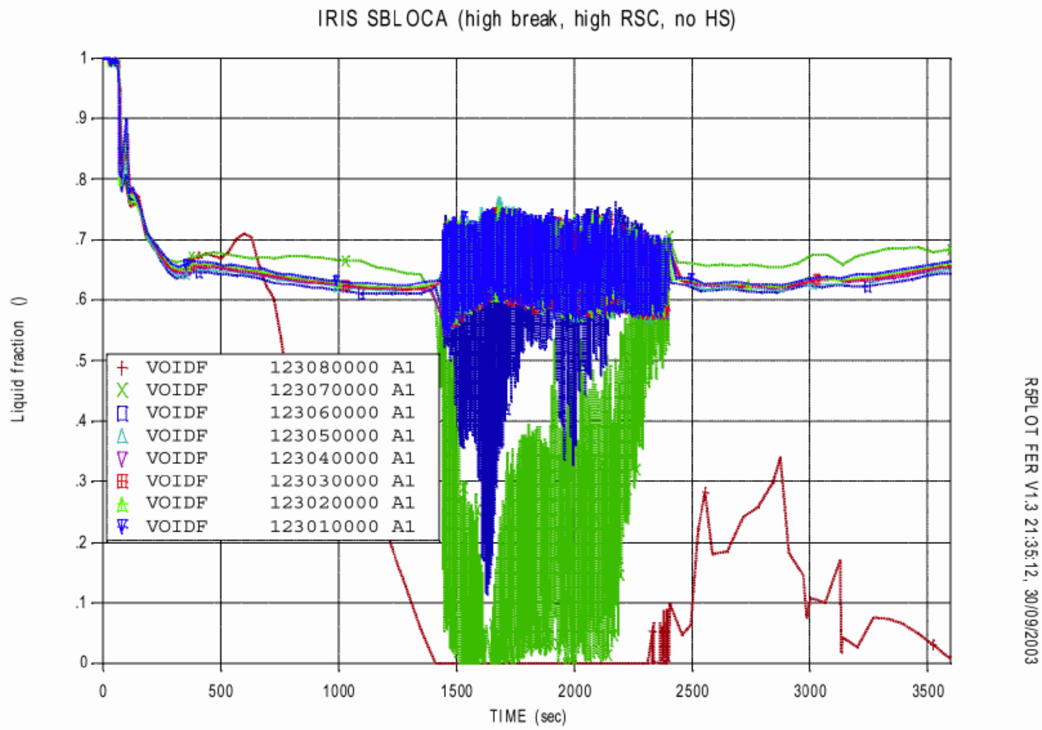


Figure 2.6-9
4" CVCS Line Break – Liquid Fraction in the Riser Transient

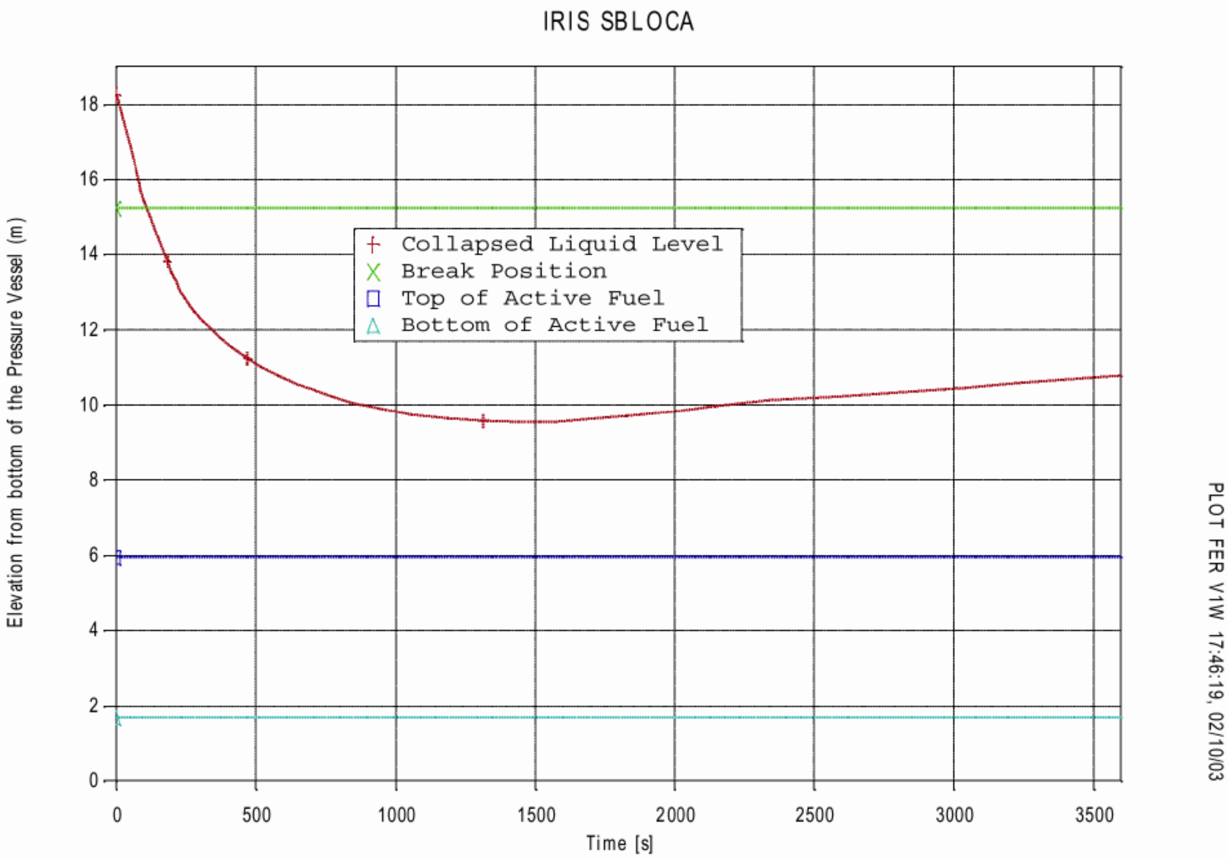


Figure 2.6-10
4" CVCS Line Break – Collapsed Liquid Level Transient

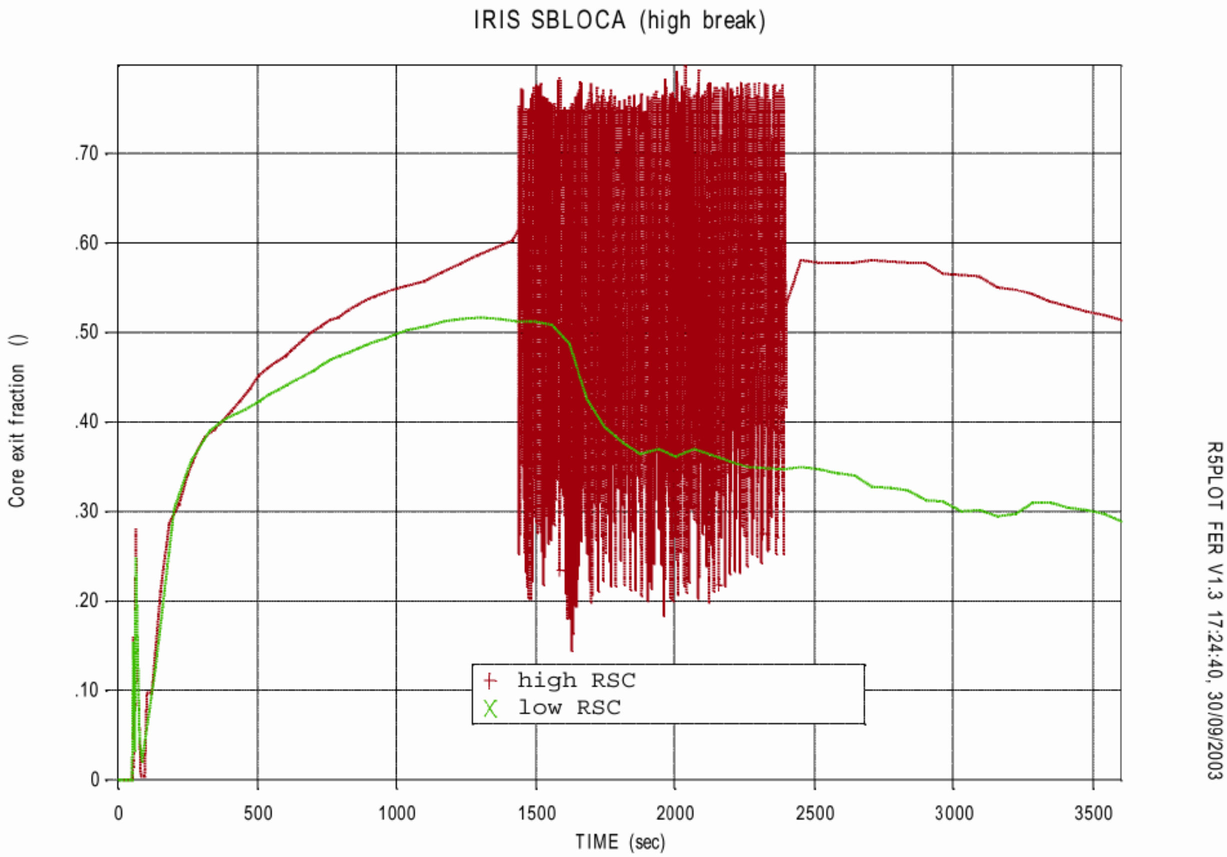
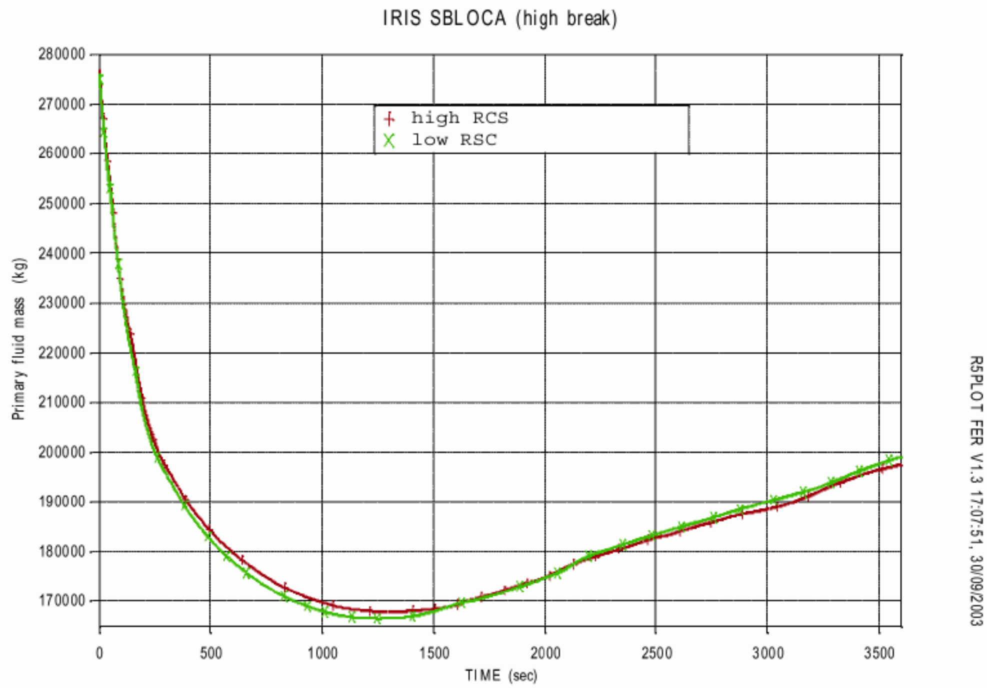


Figure 2.6-11
4" CVCS Line Break – Core Exit Void Fraction Transient
(Different position of the SG check-valves)



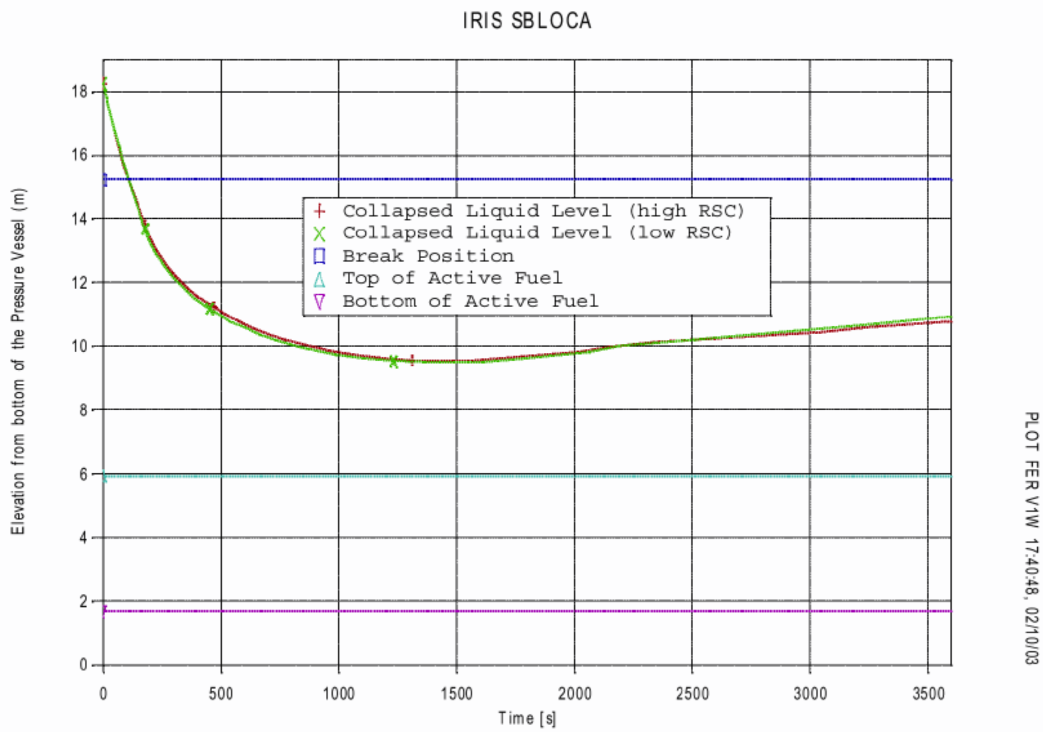
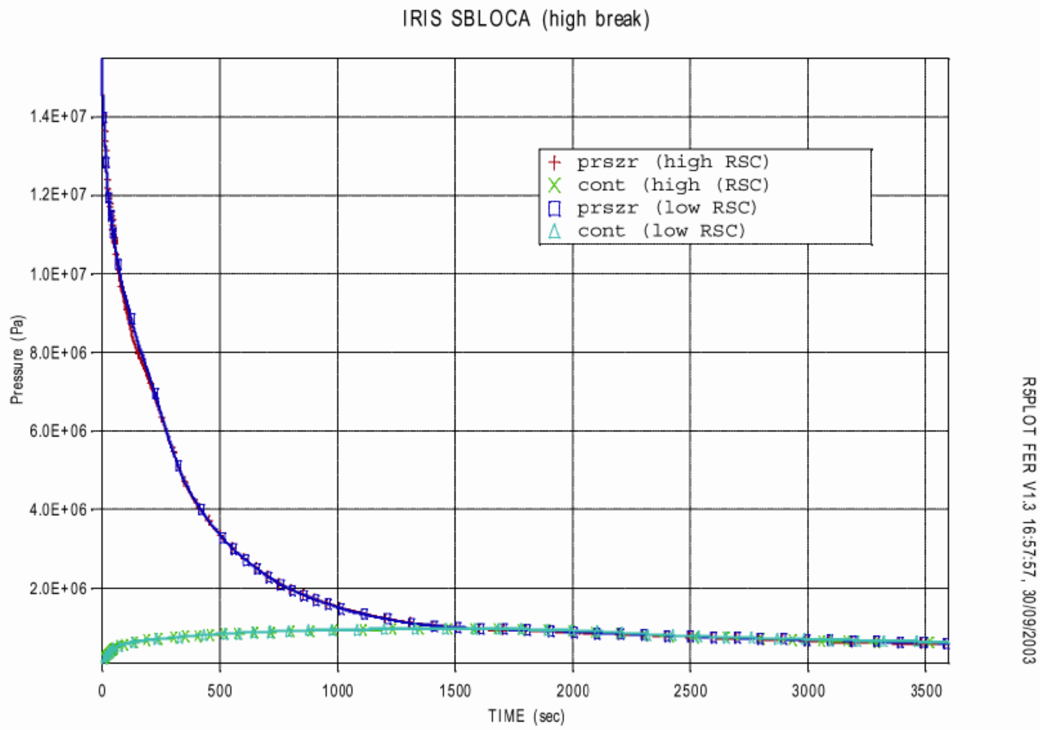


Figure 2.6-12
4" CVCS Line Break – Reactor Coolant System Fluid Mass and Collapsed Liquid Level Transient (Different position of the SG check-valves)



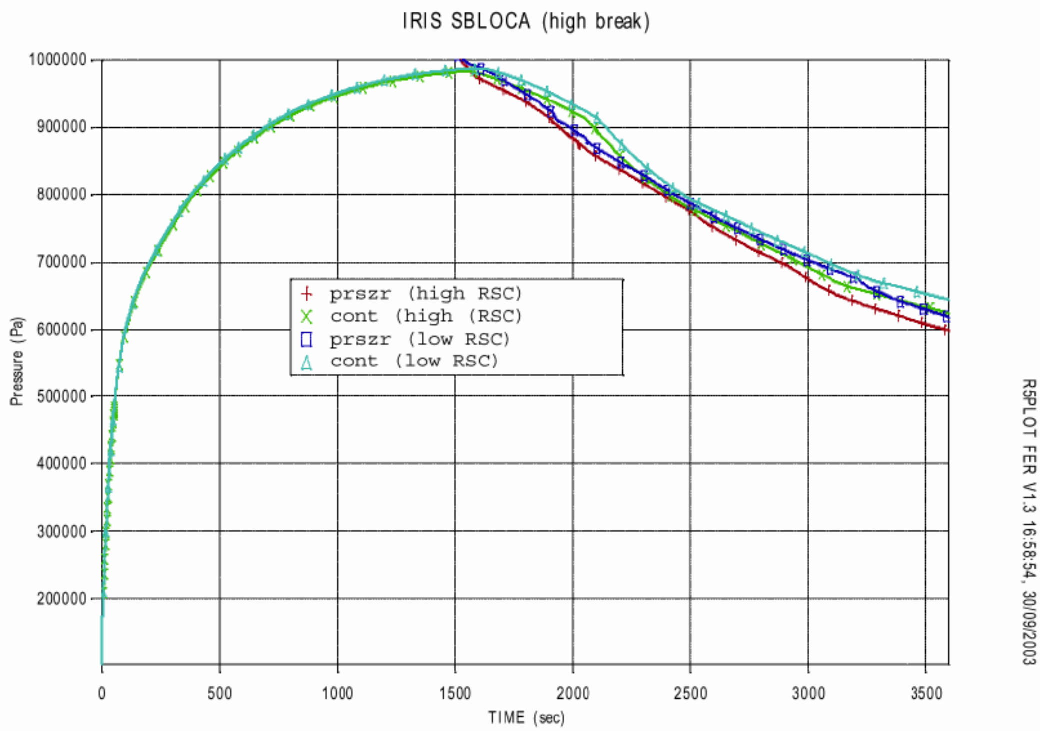
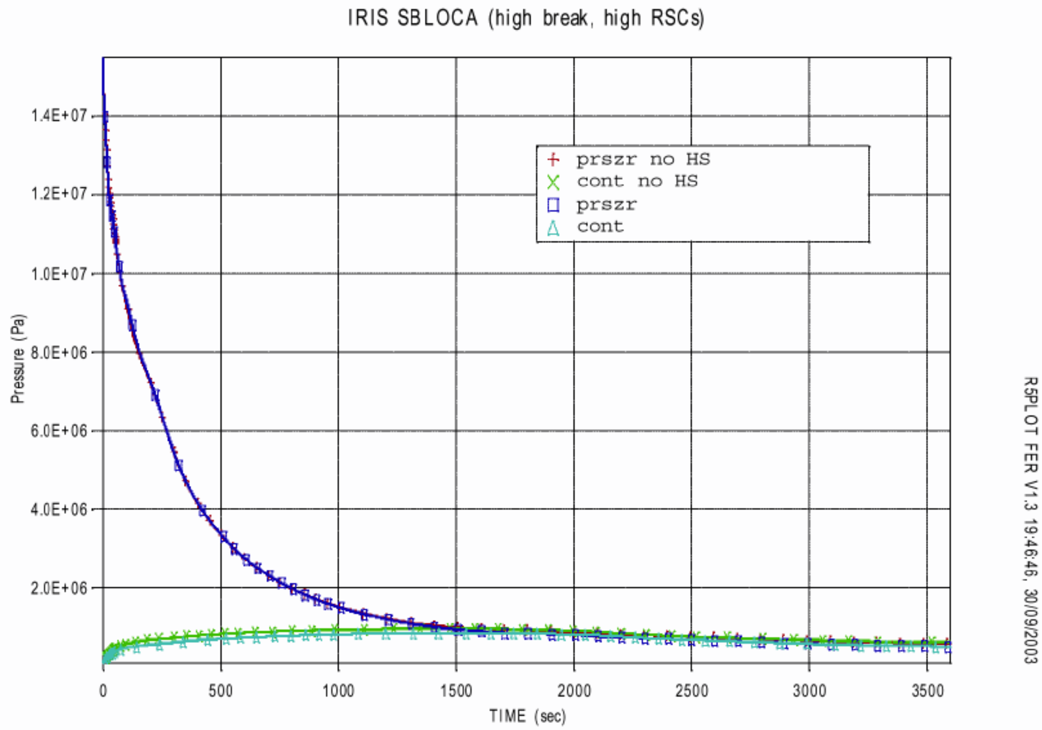


Figure 2.6-13
4" CVCS Line Break – Containment and Pressurizer Pressure Transient (with detail)
(Different position of the SG check-valves)



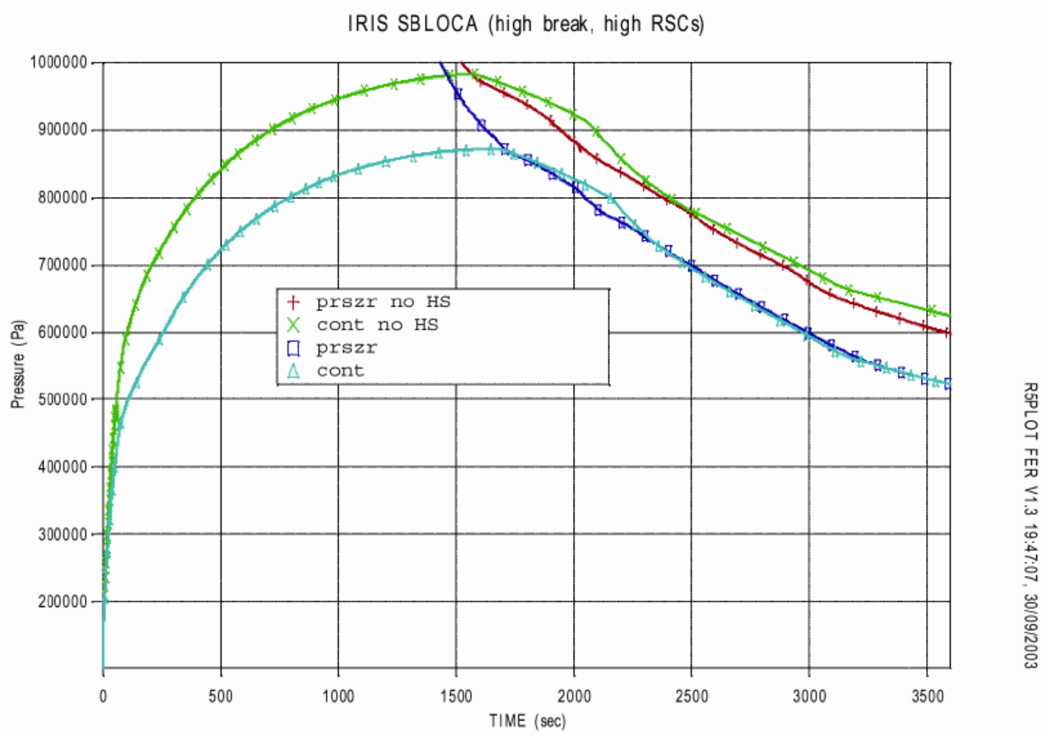


Figure 2.6-14
4" CVCS Line Break – Containment and Pressurizer Pressure Transient (with detail)
(Thermal Structures in the Containment)

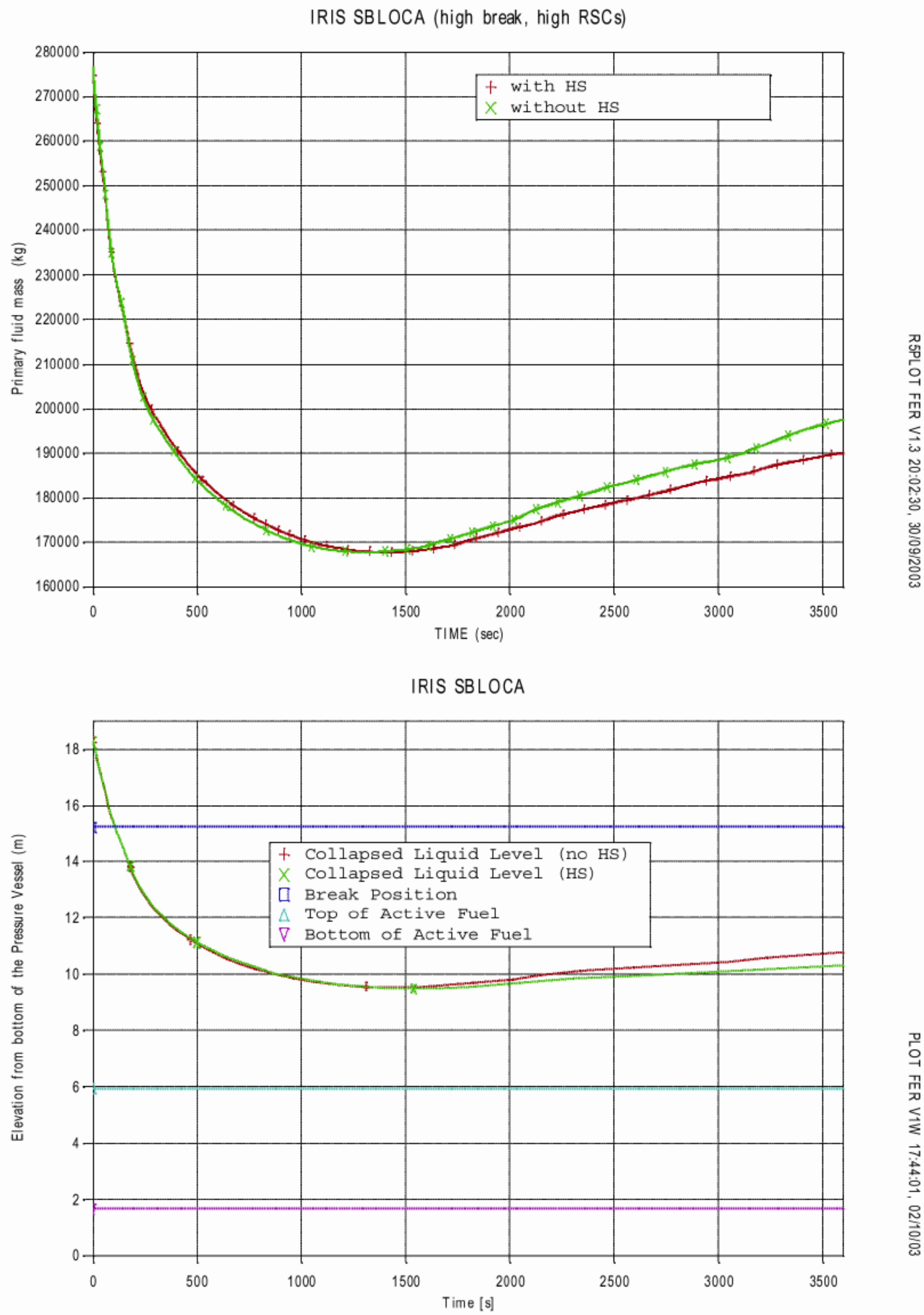
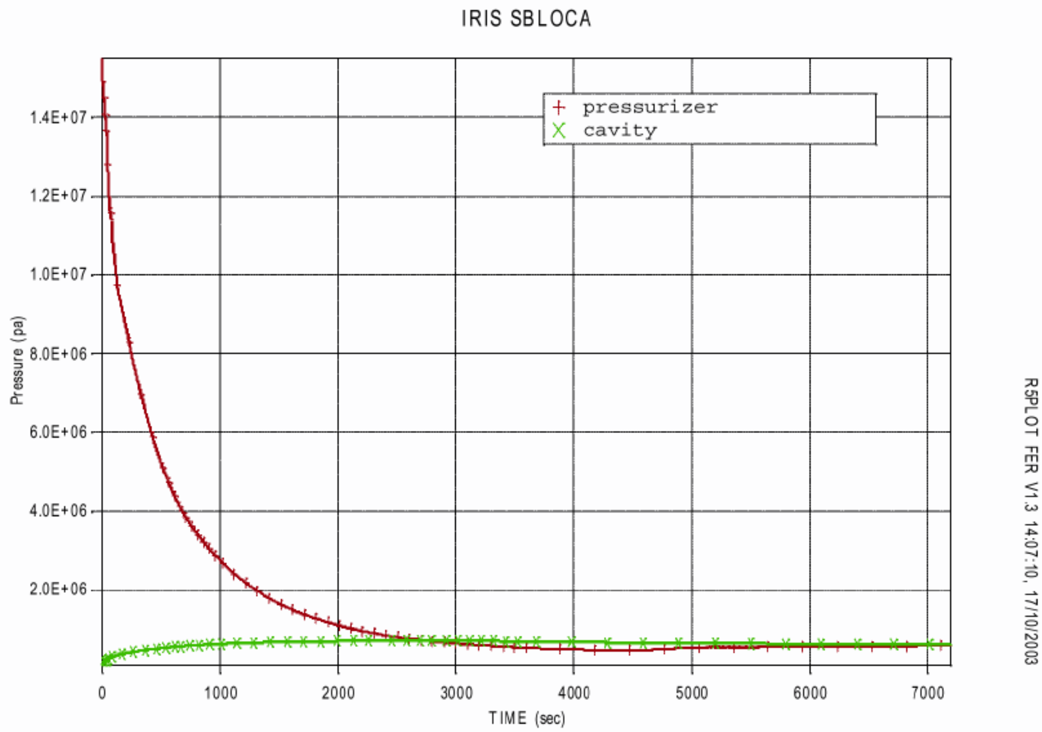


Figure 2.6-15
4" CVCS Line Break – Reactor Coolant System Fluid Mass and Collapsed Liquid Level Transient (Thermal Structures in the Containment)



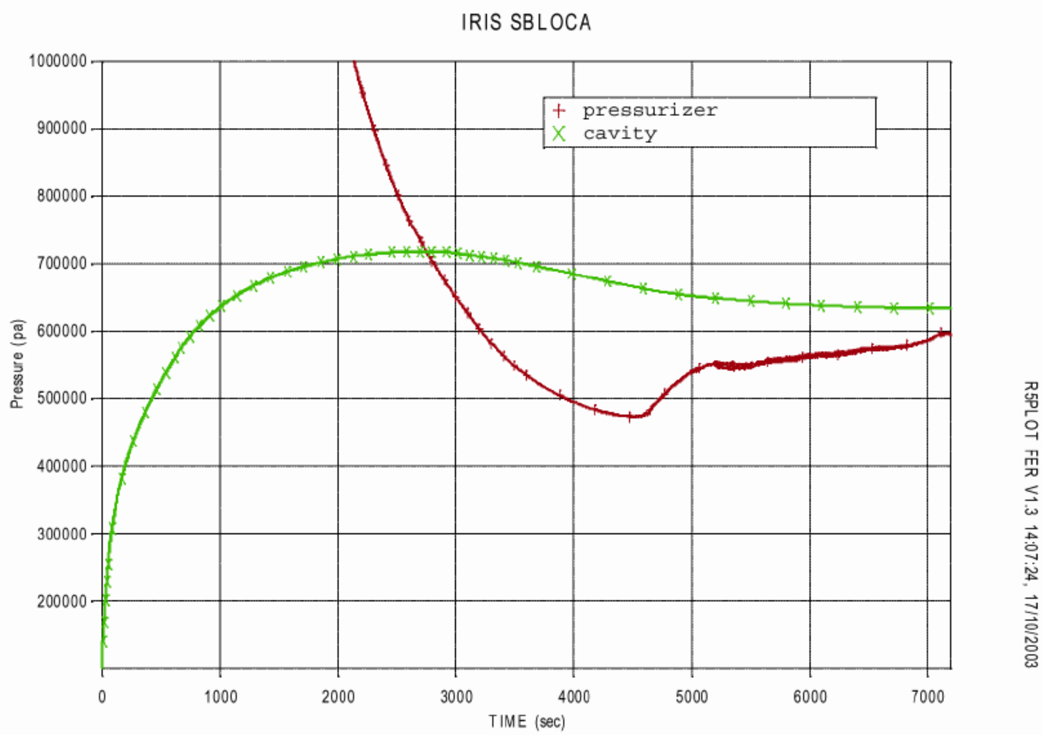


Figure 2.6-16
2" DEDVI – Containment and Pressurizer Pressure Transient (with detail)

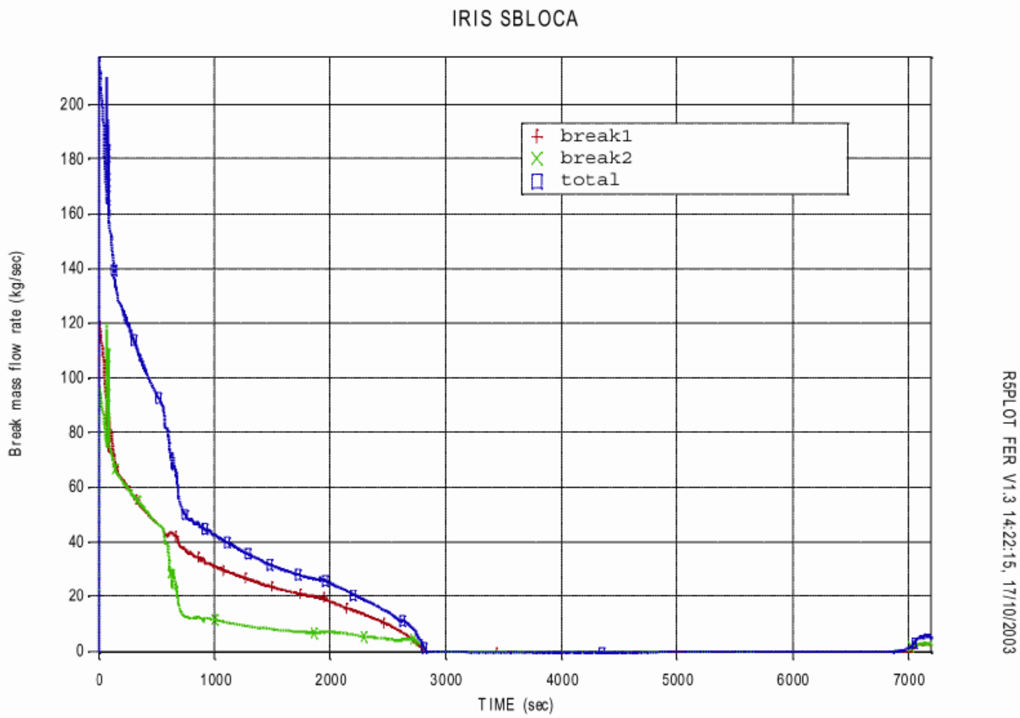
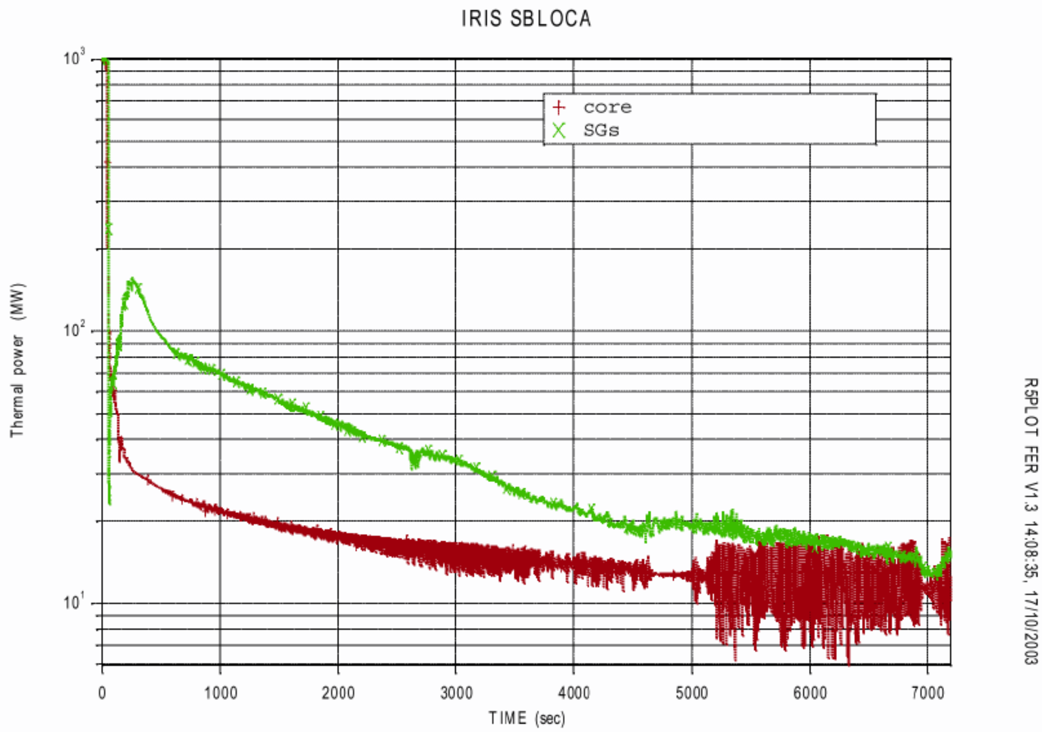


Figure 2.6-17
2" DEDVI – Break Flow Rate (with detail)



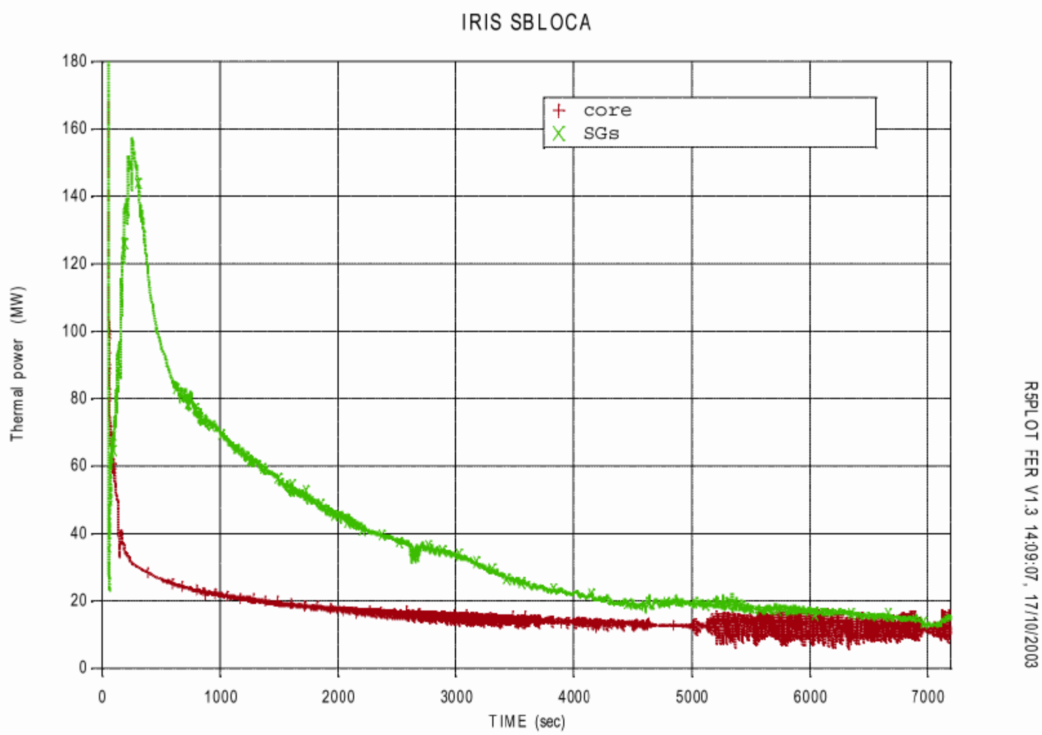


Figure 2.6-18
2" DEDVI – Overall Core and Steam Generator Heat Removal Transient (with detail)

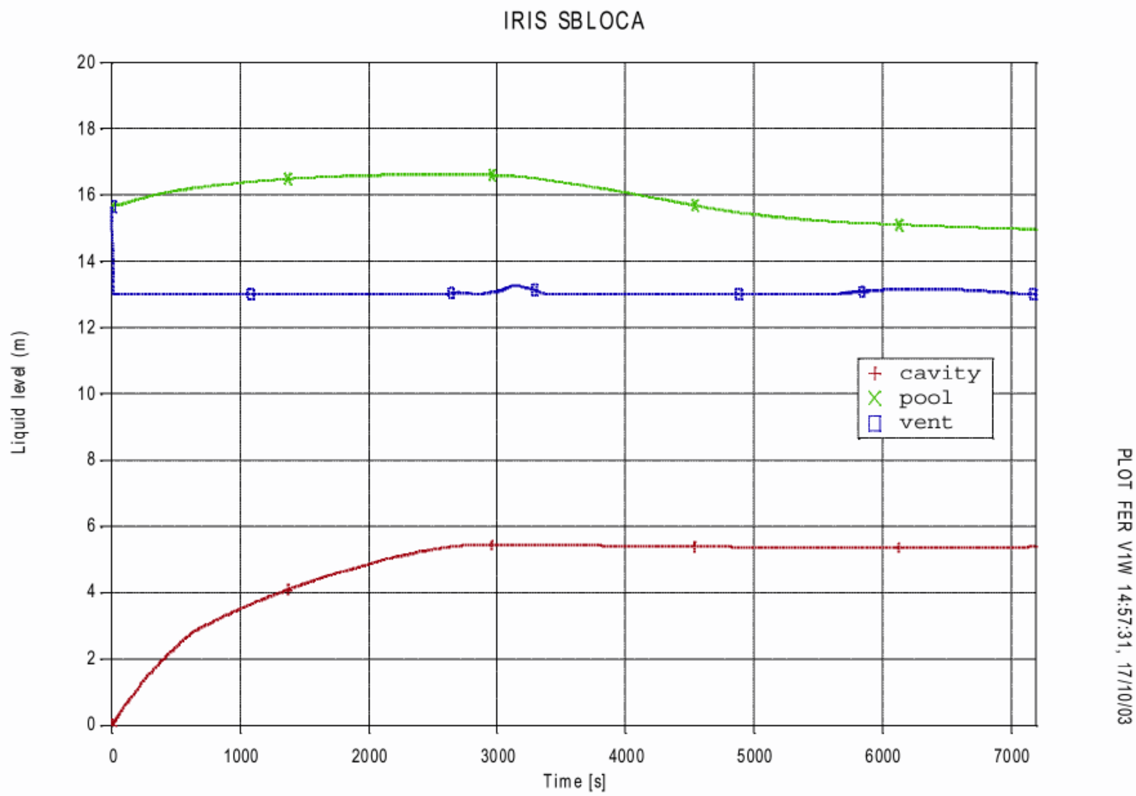
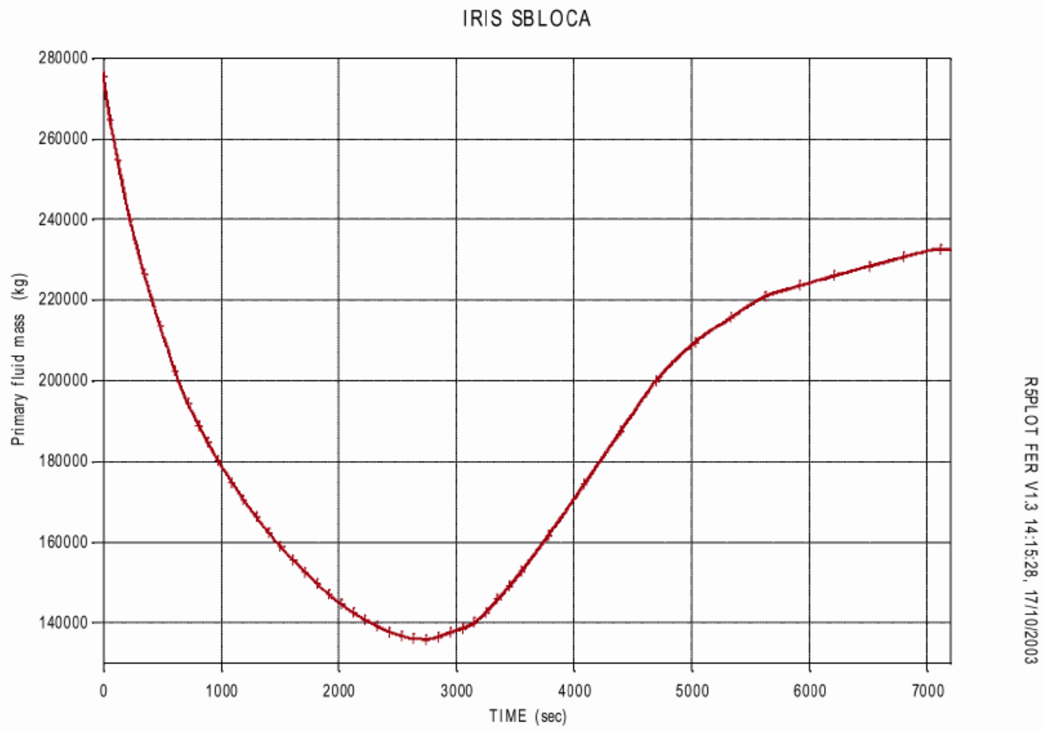


Figure 2.6-19
2" DEDVI – Containment Liquid Mass Transient



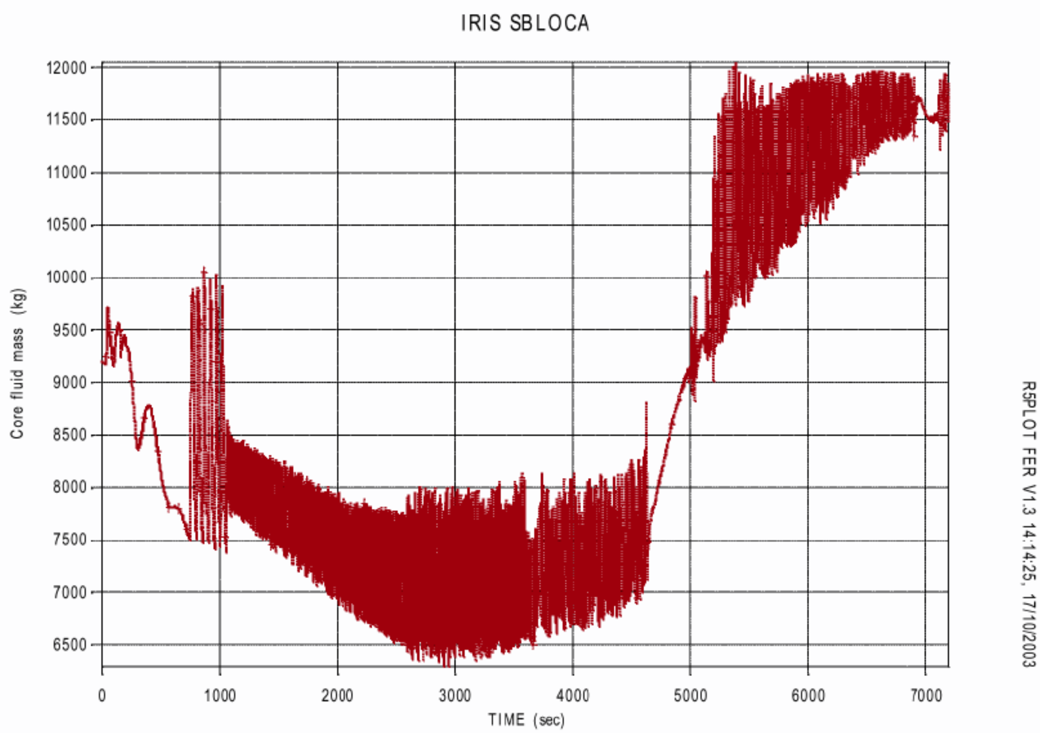


Figure 2.6-20
2" DEDVI – Reactor Coolant System and Core Liquid Mass Transient

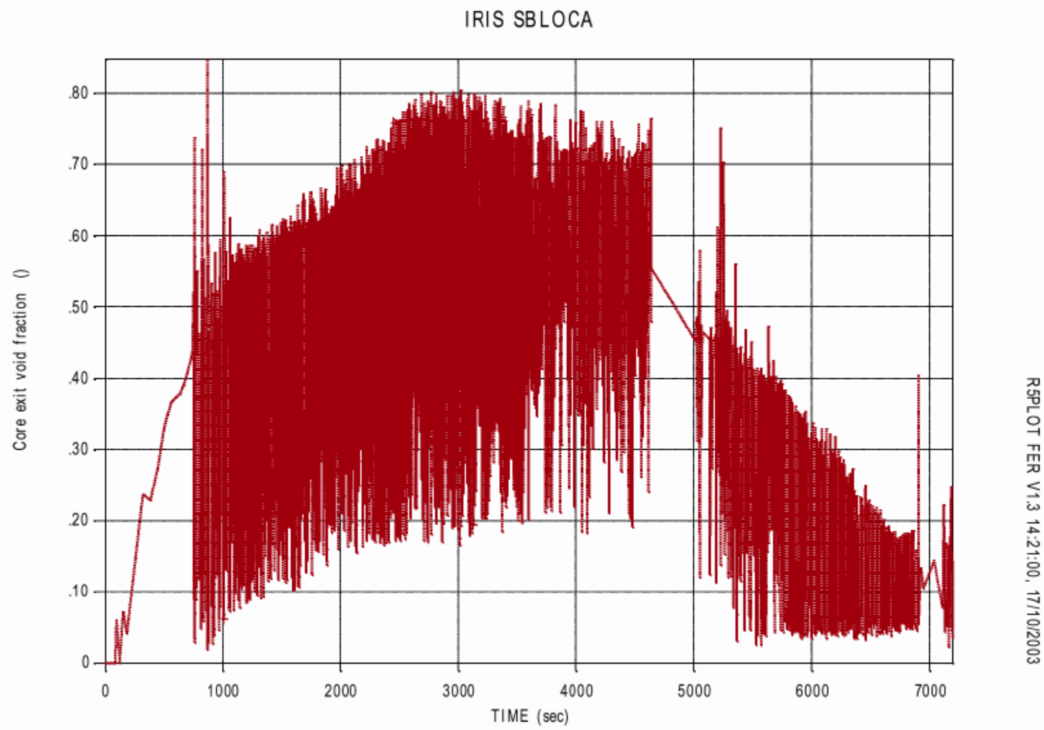


Figure 2.6-21
2" DEDVI – Core Exit Void Fraction Transient

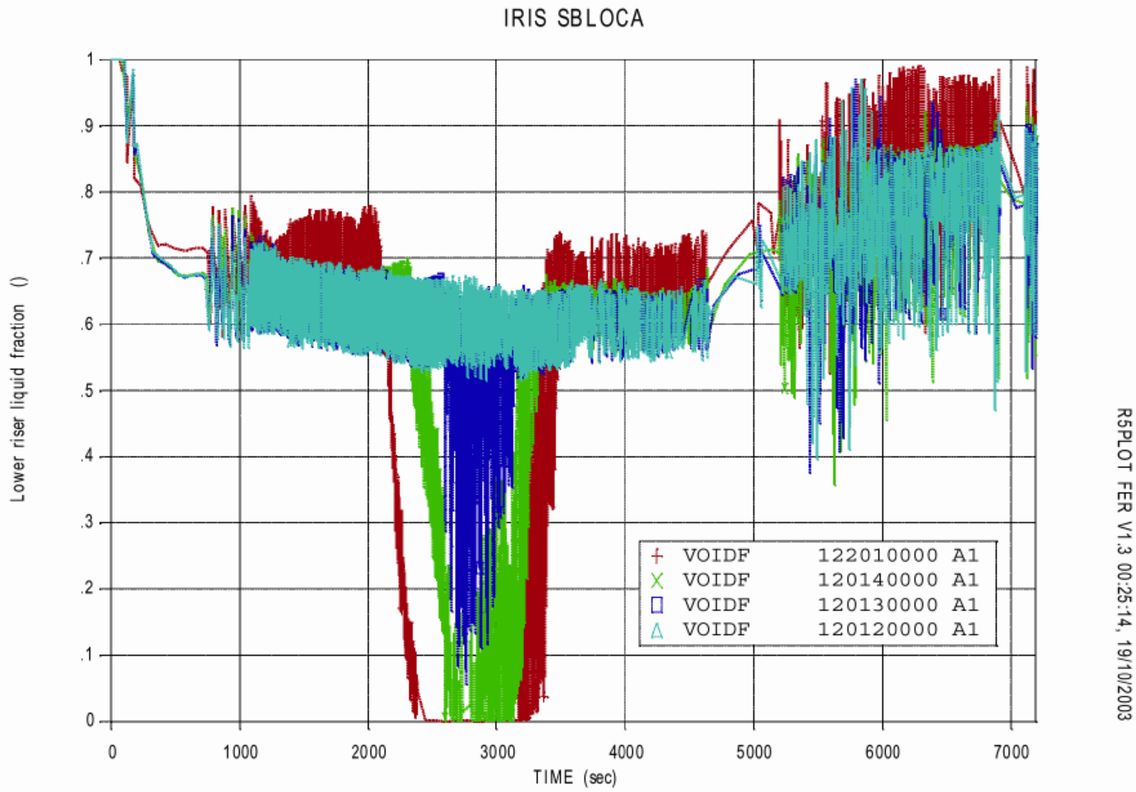


Figure 2.6-22
2" DEDVI – Liquid Fraction in the Riser Transient

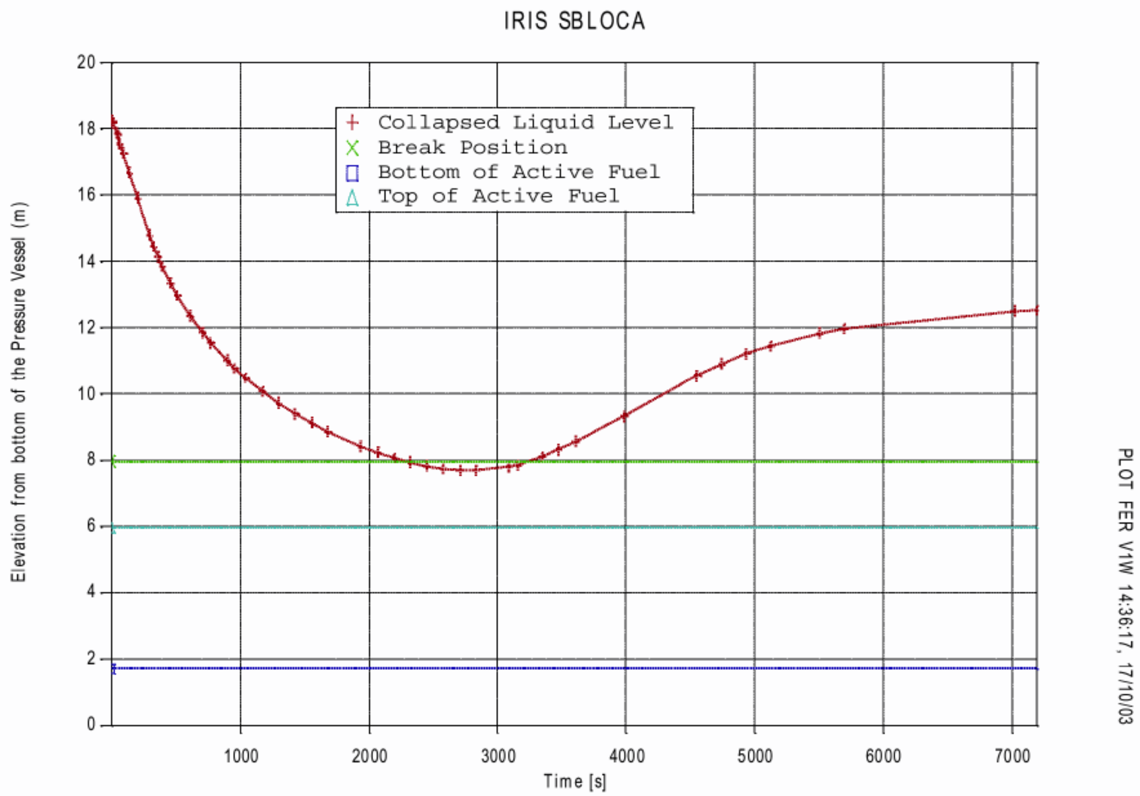


Figure 2.6-23
2" DEDVI – Collapsed Liquid Level Transient

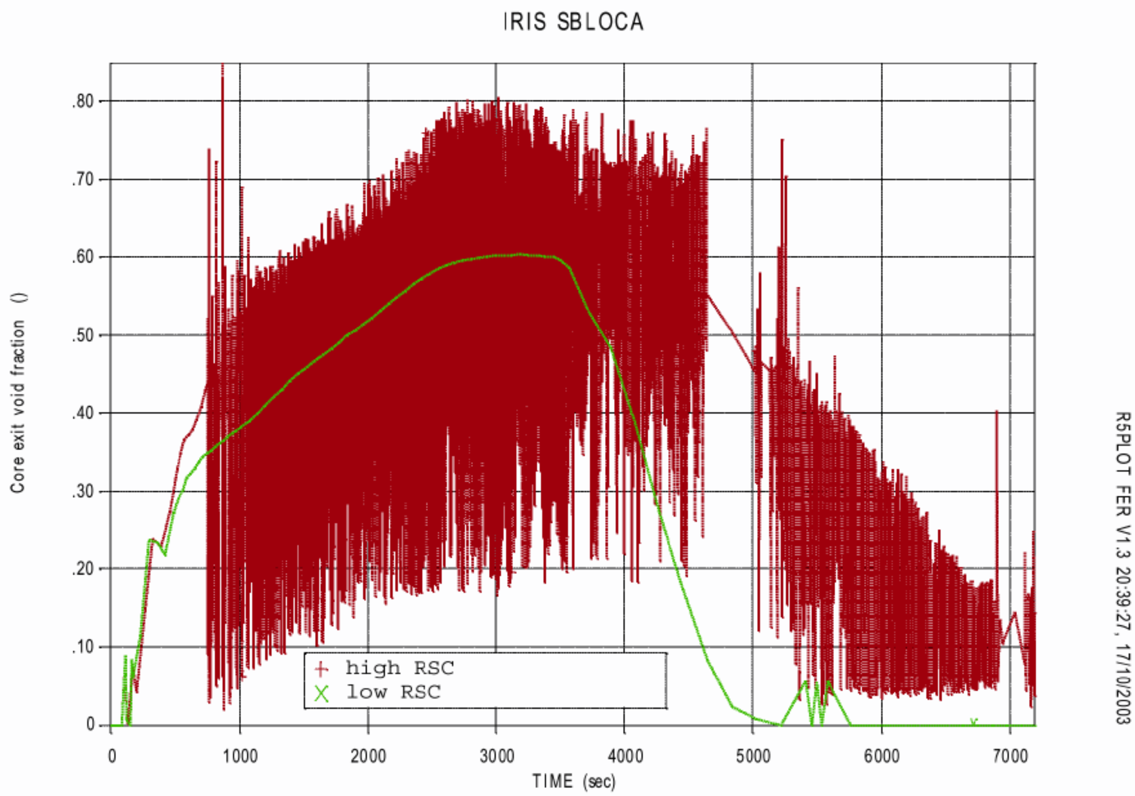
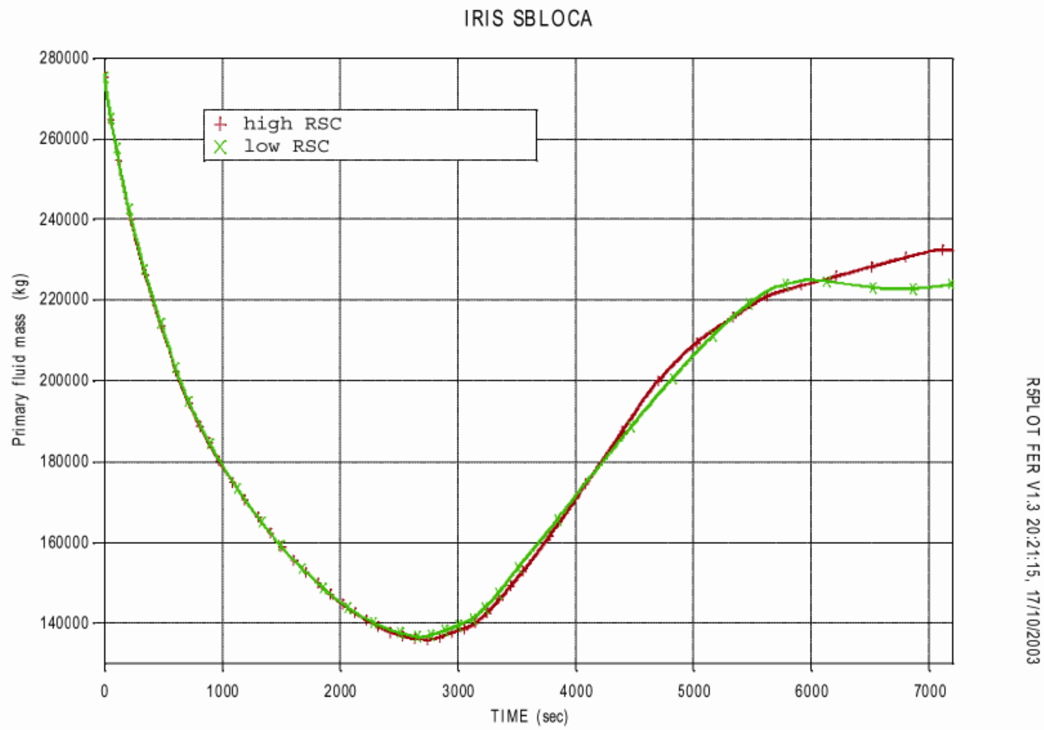


Figure 2.6-24
2" DEDVI – Core Exit Void Fraction Transient
(Different position of the SG check-valves)



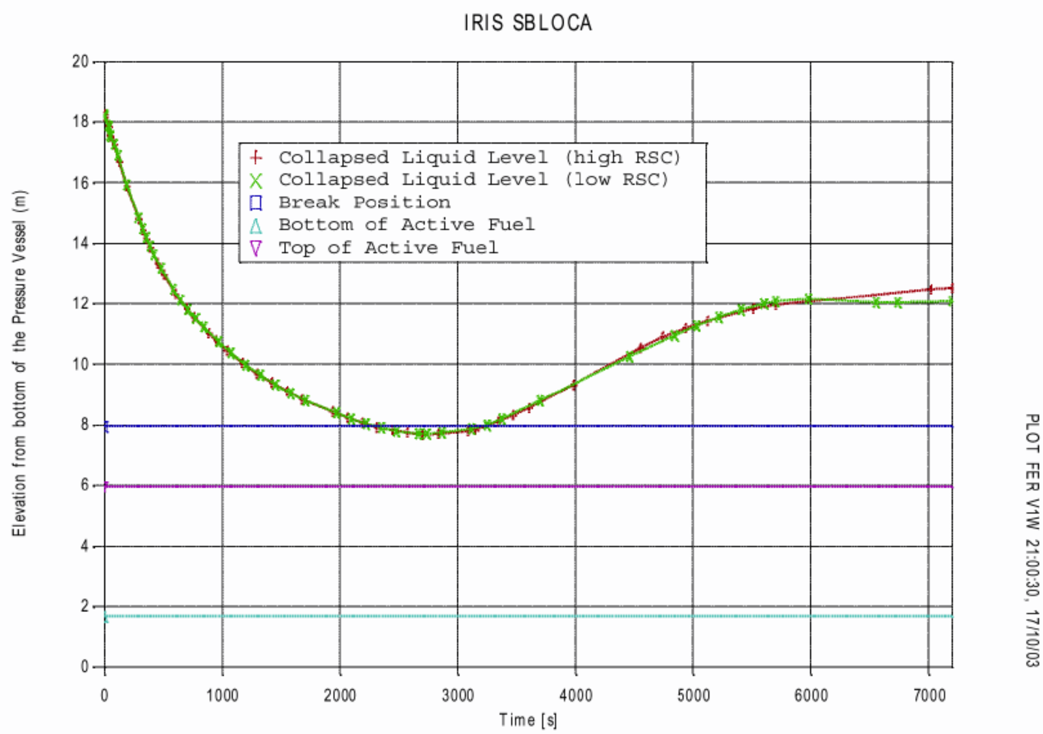
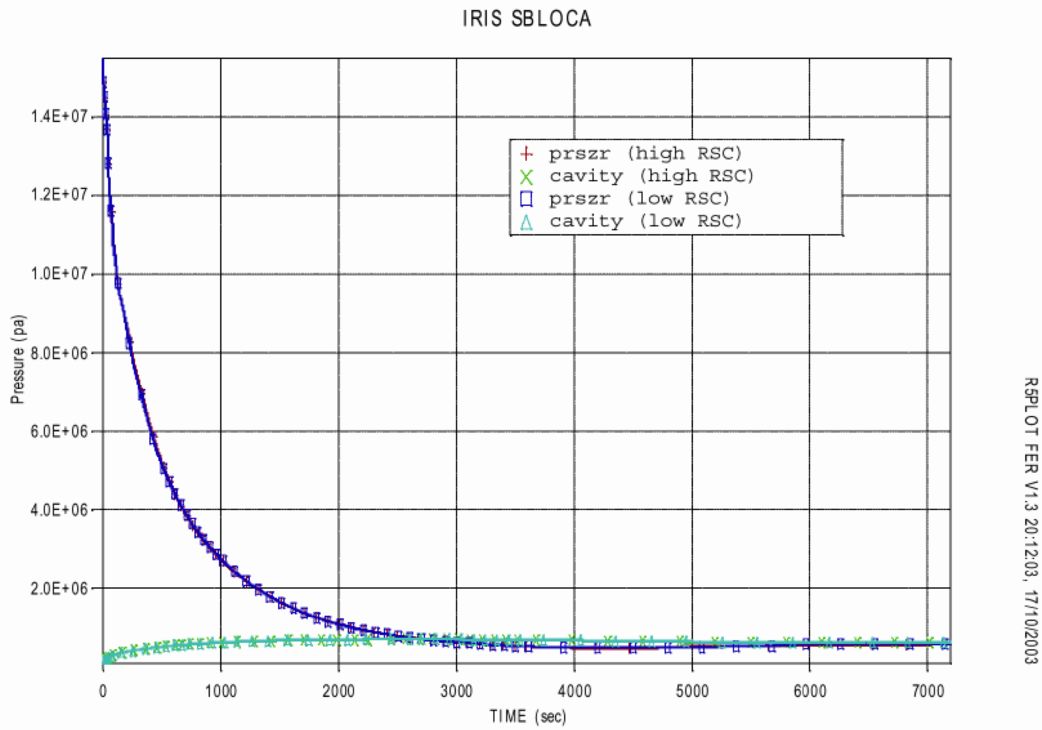


Figure 2.6-25
2" DEDVI – Reactor Coolant System Fluid Mass and Collapsed Liquid Level Transient (Different position of the SG check-valves)



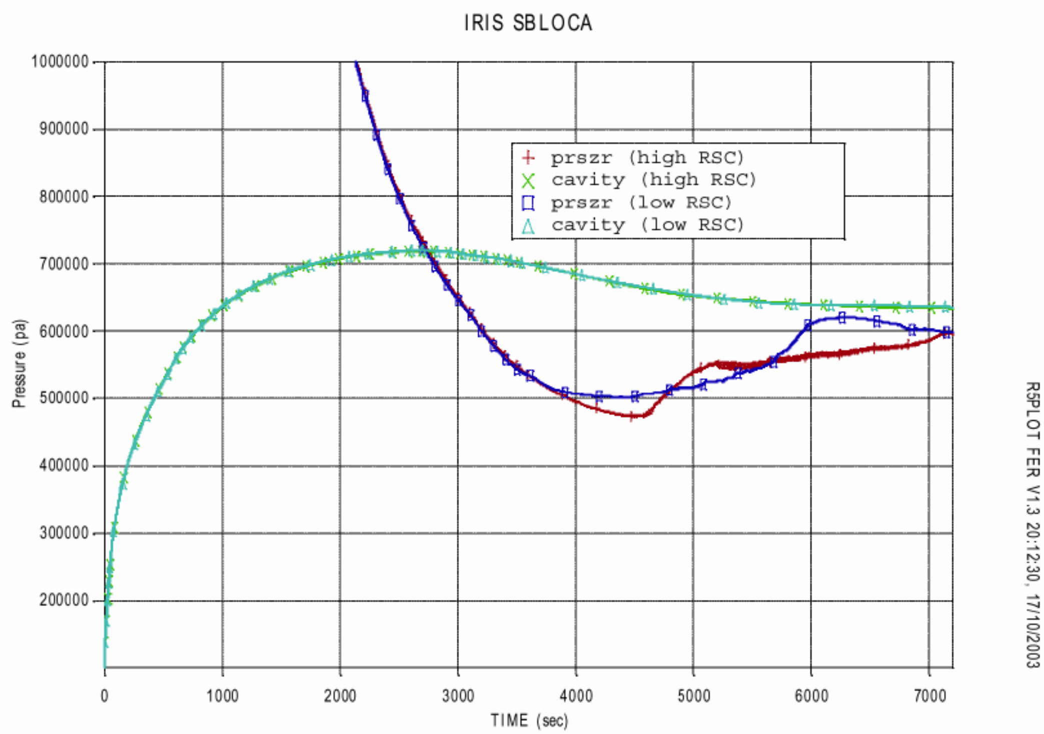
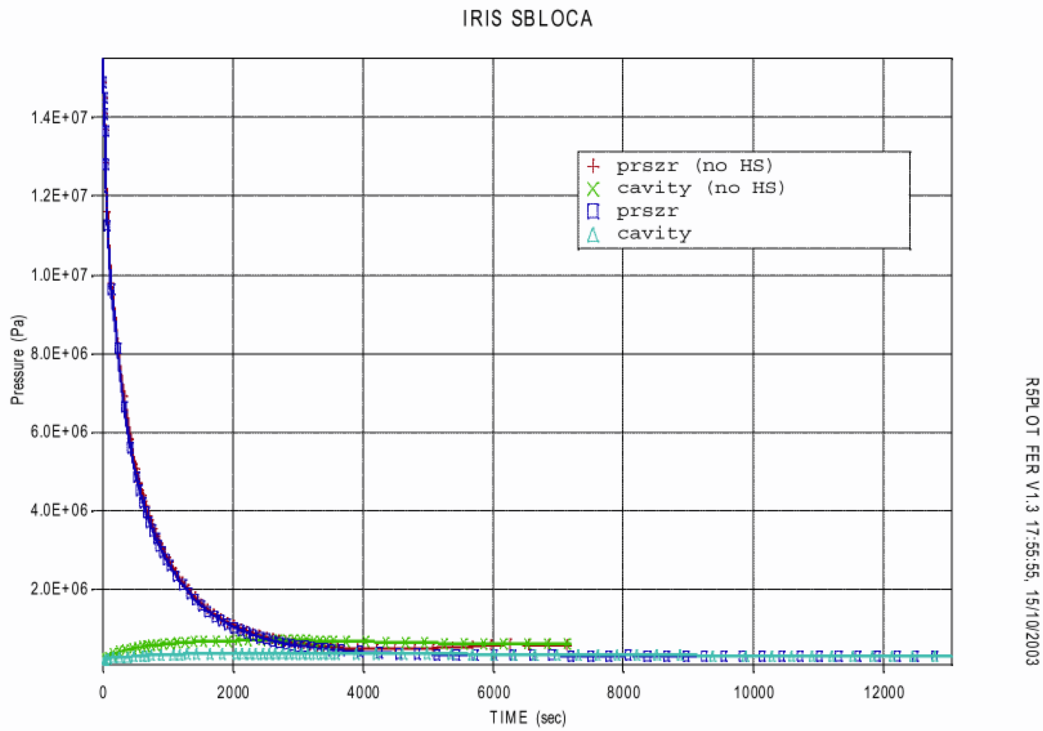


Figure 2.6-26
2" DEDVI – Containment and Pressurizer Pressure Transient (with detail)
(Different position of the SG check-valves)



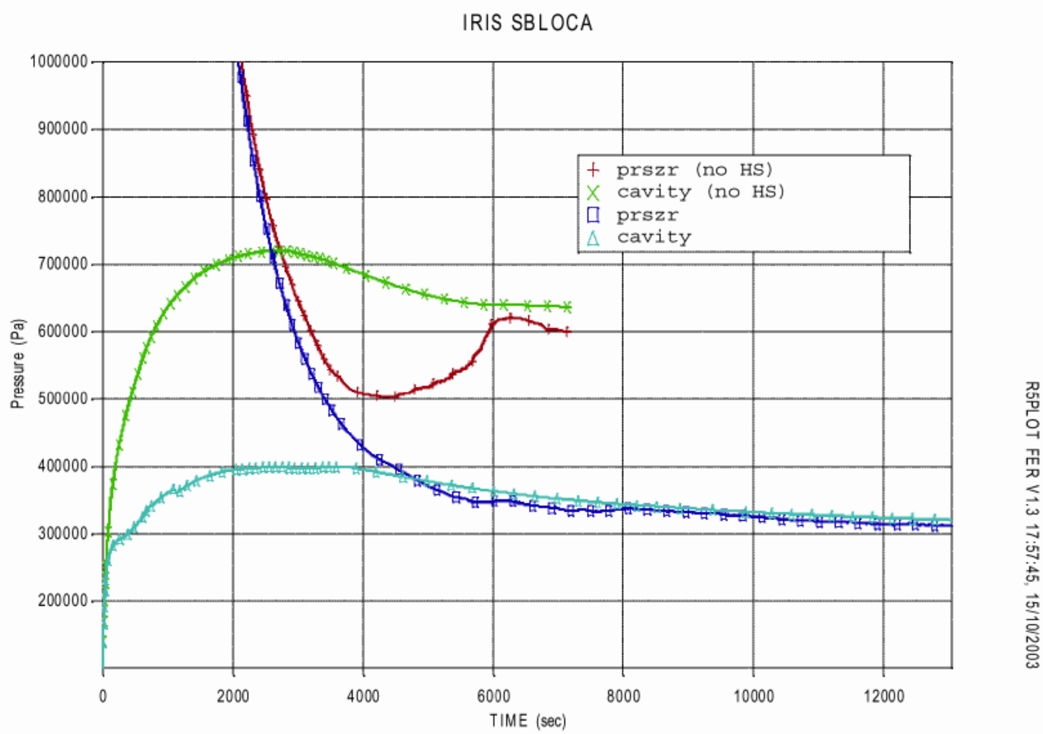
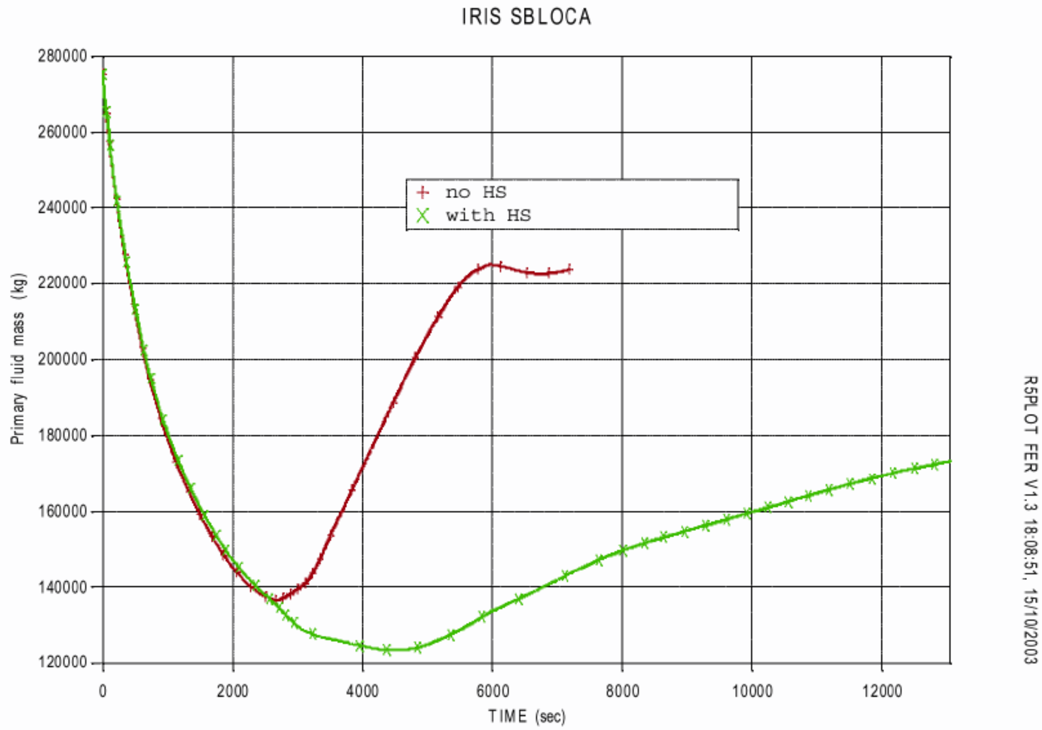


Figure 2.6-27
2" DEDVI – Containment and Pressurizer Pressure Transient (with detail)
(Thermal Structures in the Containment)



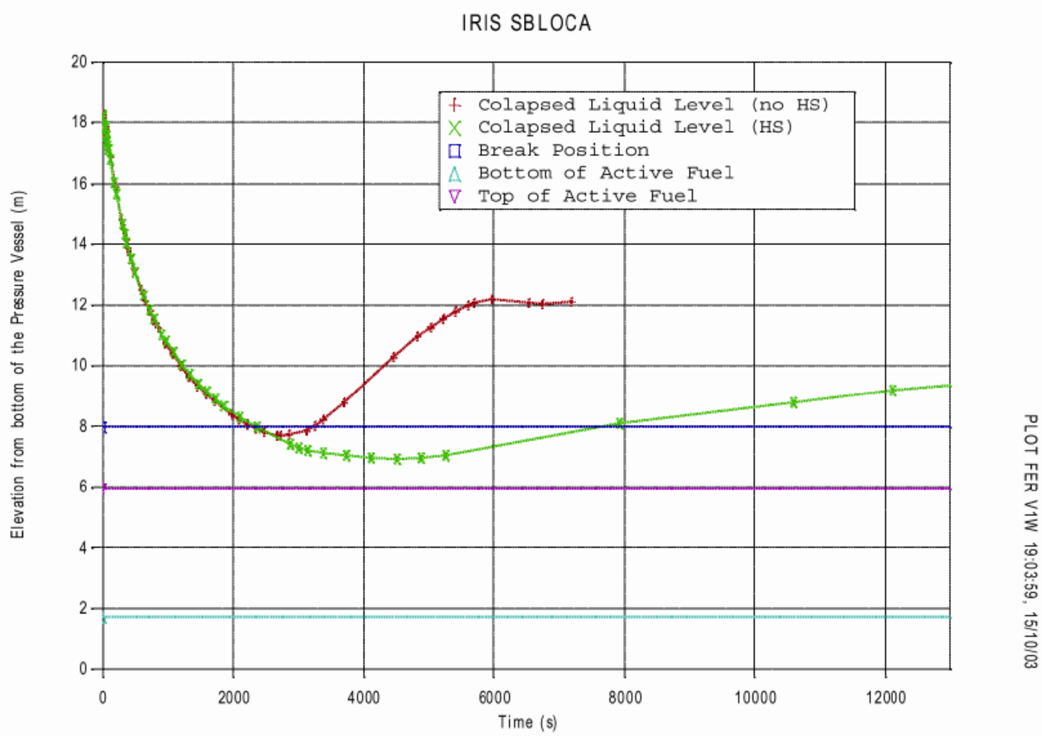


Figure 2.6-28
2" DEDVI – Reactor Coolant System Fluid Mass and Collapsed Liquid Level Transient (Thermal Structures in the Containment)

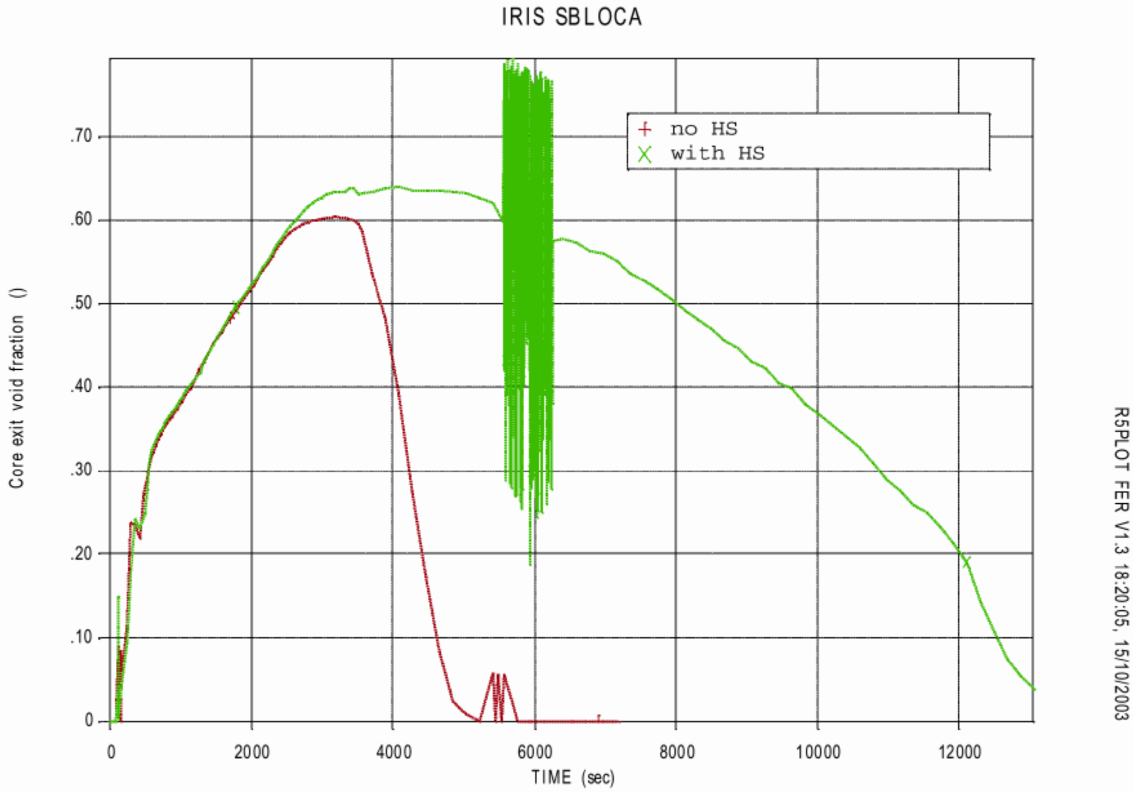


Figure 2.6-29
2" DEDVI – Core Exit Void Fraction Transient
(Thermal Structures in the Containment)

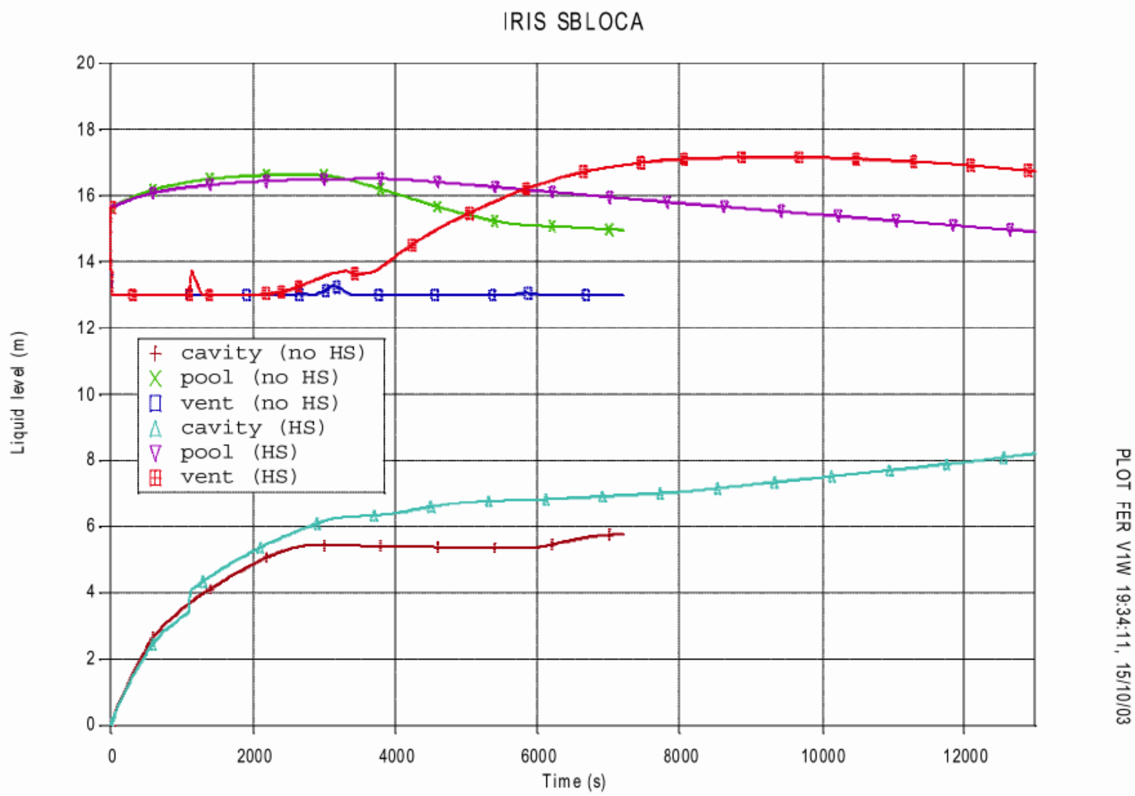


Figure 2.6-30
2" DEDVI – Containment Liquid Mass Transient
(Thermal Structures in the Containment)

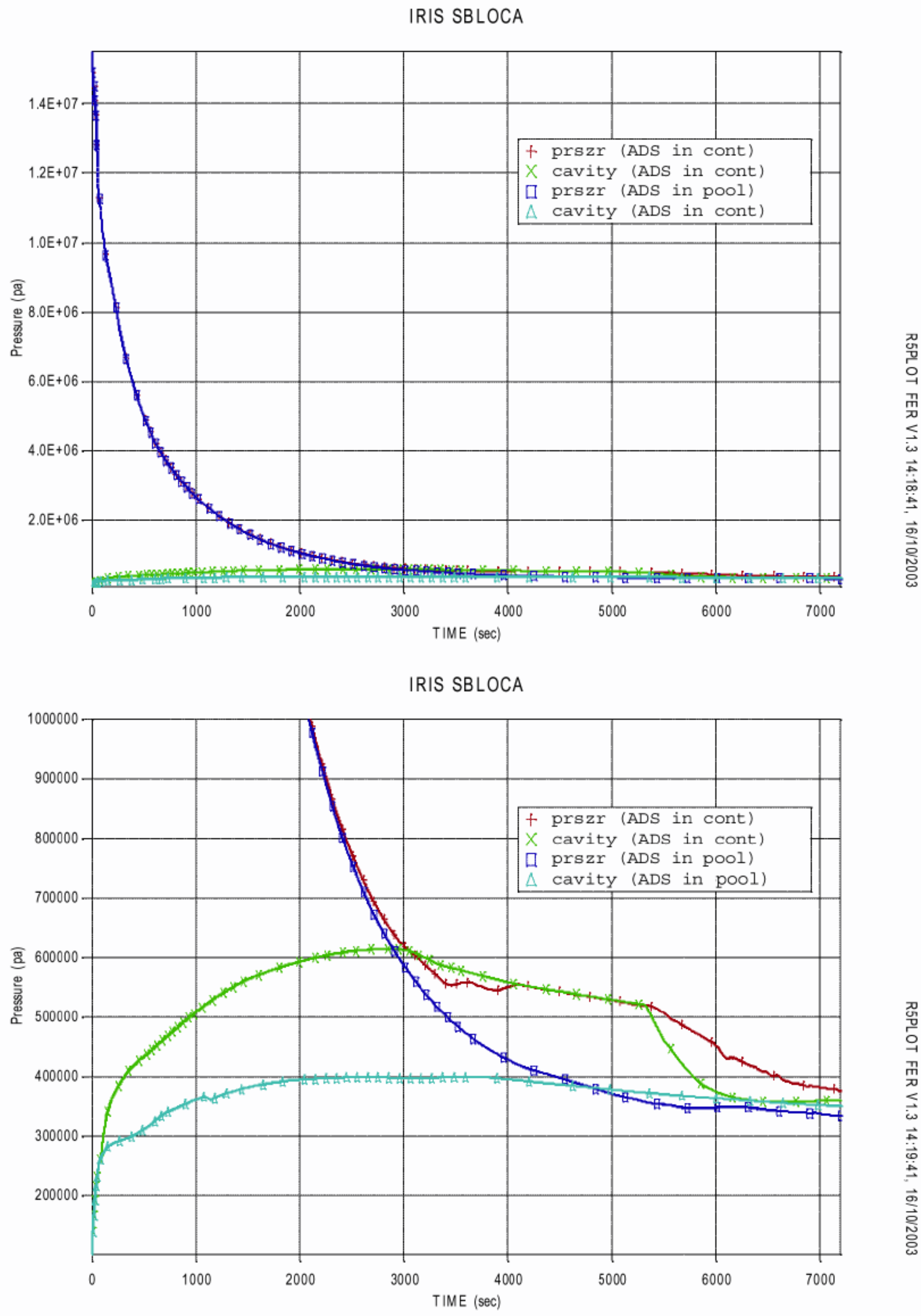
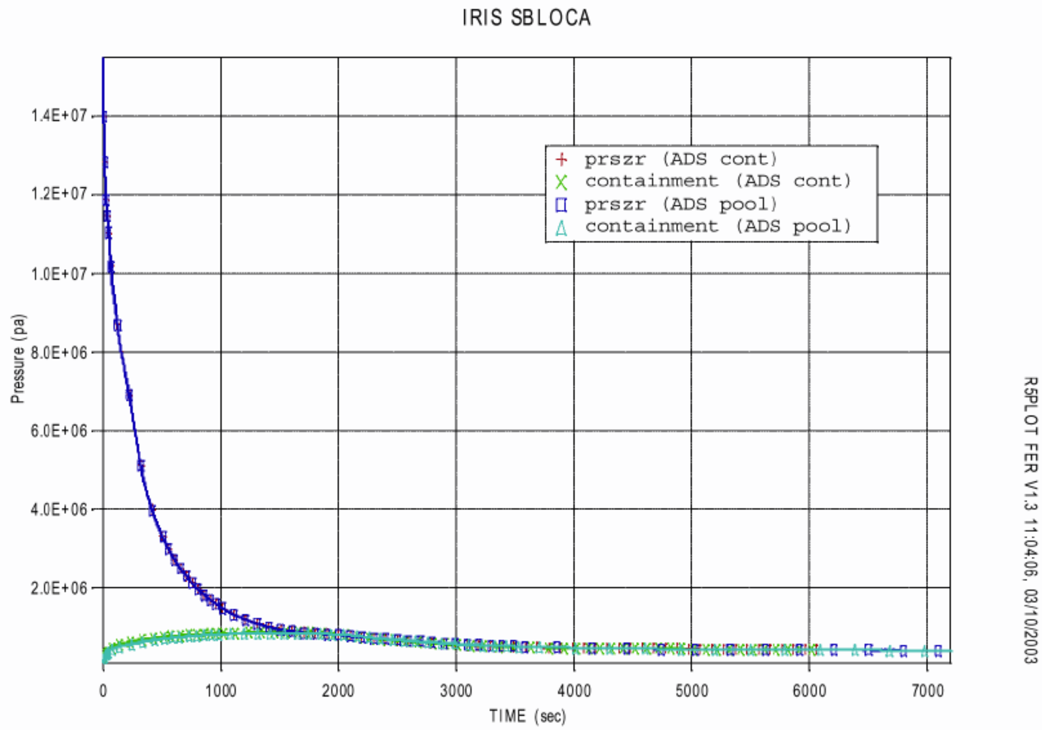


Figure 2.6-31
2" DEDVI – Containment and Pressurizer Pressure Transient (with detail)
(Thermal Structures in the Containment, Different ADS Layout)



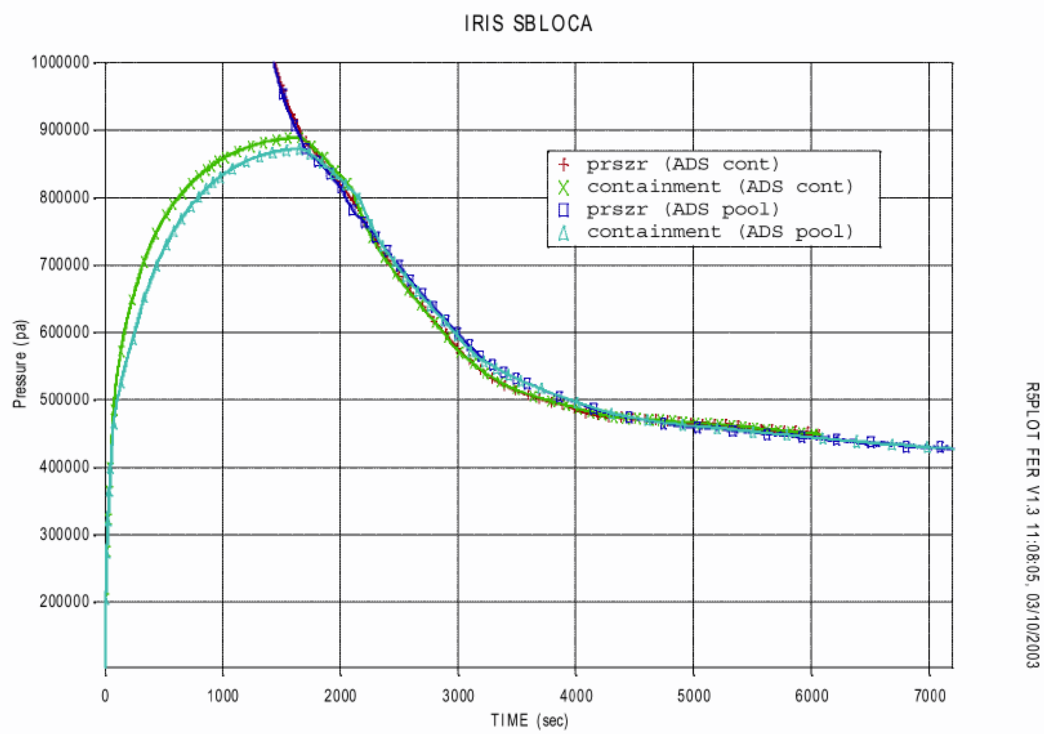


Figure 2.6-32
4" CVCS Line Break – Containment and Pressurizer Pressure Transient (with detail)
(Thermal Structures in the Containment, Different ADS Layout)

2.7 Radioactive Release from a Subsystem or Components

This group of events includes the following:

- Radioactive gas waste system leak or failure
- Radioactive liquid waste system leak or failure
- Postulated radioactive release due to liquid tank failures
- Fuel handling accident
- Spent fuel cask drop accidents

These events will not present in IRIS any significant difference from other Westinghouse PWRs. No additional considerations are provided at this time.

2.8 Anticipate Transients Without SCRAM (ATWS)

2.8.1 General Background

An anticipated transient without scram (ATWS) is an anticipated operational occurrence during which an automatic reactor scram is required but fails to occur due to a common mode fault in the reactor protection system or other reason. Under certain circumstances, failure to execute a required scram during an anticipated operational occurrence could transform a relatively minor transient into a more severe accident. As for other Westinghouse plants, ATWS events are not considered to be in the design basis for IRIS.

2.8.2 Anticipate Transients Without Scram in IRIS

For Westinghouse plants the ATWS rule (10 CFR 50.62) requires the installation of ATWS mitigation systems actuation circuitry (AMSAC), which consists of circuitry, separate from the reactor protection system, to trip the turbine and initiate decay heat removal. Note that IRIS conforms to this as well as to the other requirements of the ATWS rule.

The basis for the ATWS rule requirements, as outlined in SECY-83-293 (Reference 1), is to reduce the risk of core damage caused by ATWS to less than 10^{-5} per reactor year.

IRIS includes a diverse actuation system, which provides the AMSAC protection features required for Westinghouse plants by 10CFR 50.62, plus a diverse reactor scram. The AMSAC protection has been applied to account for the specific features of the IRIS design to ensure that the objectives of the ATWS rule are met.

2.8.3 Conclusions

The IRIS is equipped with a diverse actuation system, which provides the function of AMSAC. The ATWS core damage frequency for the IRIS will be demonstrated to be well below the SECY-83-293 goal of 10^{-5} per reactor year. The IRIS design will therefore meet the objectives of the ATWS rule and its ATWS core damage frequency safety goal basis.

2.8.4 References

[1] Dircks, W. J., “Amendments to 10CFR50 Related to Anticipate Transients Without Scram (ATWS) Events,” SECY-83-293, U.S. NRC, July 19, 1983