6.3 Emergency Core Cooling Systems

6.3.1 Design Bases and Summary Description

Subsection 6.3.1 provides the design bases for the Emergency Core Cooling Systems (ECCS) and a summary description of these systems as an introduction to the more detailed design descriptions provided in Subsection 6.3.2 and the performance analysis provided in Subsection 6.3.3.

6.3.1.1 Design Bases

6.3.1.1.1 Performance and Functional Requirements

The ECCS is designed to provide protection against postulated loss-of-coolant accidents (LOCAs) caused by ruptures in primary system piping. The functional requirements (e.g., coolant delivery rates) specified in detail in Table 6.3-1 are such that the system performance under all LOCA conditions postulated in the design satisfies the requirements of 10CFR50, Paragraph 50.46 (Acceptance Criteria for Emergency Core Cooling System for Light-Water-Cooled Nuclear Power Reactors). These requirements are summarized in Subsection 6.3.3.2. In addition, the ECCS is designed to meet the following requirements:

(1) Protection is provided for any primary system line breakup, including the double-ended break of the largest line.

(2) Three high-pressure cooling systems are provided, each of which is capable of maintaining water level above the top of the core and preventing ADS actuation for breaks of lines less than 25A.

(3) No operator action is required until 30 min after an accident to allow for operator assessment and decision.

(4) The ECCS is designed to satisfy all criteria specified in Section 6.3 for any normal mode of reactor operation.

(5) A sufficient water source and the necessary piping, pumps and other hardware are provided so that the containment and reactor core can be flooded for possible core heat removal following a LOCA.

6.3.1.1.2 Reliability Requirements

The following reliability requirements apply:

(1) The ECCS must conform to all licensing requirements and good design practices of isolation, separation and common mode failure considerations.

(2) In order to meet the above requirements, the ECCS network has built-in redundancy so that adequate cooling can be provided, even in the event of specified failures. As a minimum, the following equipment make up the ECCS:

(a) Reactor Core Isolation Cooling (RCIC) Loop

(b) High Pressure Core Flooder (HPCF) Loops

(c) Low Pressure Flooder (LPFL) mode of Residual Heat Removal (RHR) Loops

(d) Automatic Depressurization System (ADS)

(3) The system shall be designed so that a single active or passive component failure, including power buses, electrical and mechanical parts, cabinets and wiring will not disable the ADS.

(4) In the event of a break in a pipe that is not part of the ECCS, no single active component failure in the ECCS shall prevent automatic initiation and successful operation of less than the following combinations of ECCS equipment:

(a) One HPCF + RCIC + two LPFL + all ADS valves

(b) Two HPCF + three LPFL + all ADS valves

(c) Two HPCF + RCIC + three LPFL + all ADS valves minus one

(5) In the event of a break in a pipe that is a part of ECCS, no single active component failure in the ECCS prevents automatic initiation and successful operation of less than the following combination of ECCS equipment as identified in Subsection 6.3.1.1.2 (4), minus the ECCS in which the break is assumed.

These are the minimum ECCS combinations which result after assuming any failure (from item (4) above) and assuming that the ECCS line break disables the affected system.

(6) Long-term (10 min after initiation signal) cooling requirements call for the removal of decay heat via the reactor service water system. In addition to the break which initiated the loss-of-coolant event, the system must be able to sustain one failure, either active or passive and still have at least two low pressure pumps with heat exchangers receiving 100% service water flow and one ECCS pump that provides flow to the vessel, which can be one of the low pressure pumps.

(7) Offsite power is the preferred source of power for the ECCS network, and every reasonable precaution must be made to assure its high availability. However, onsite emergency power shall be provided with sufficient diversity and capacity so that all the above requirements can be met even if offsite power is not available.

(8) The onsite diesel fuel reserve is in accordance with Regulatory Guide 1.137.

(9) The diesel-load configuration provides one diesel generator for each of the three ECCS divisions.

(10) Systems which interface with, but are not part of, the ECCS are designed and operated such that failure(s) in the interfacing systems shall not propagate to and/or affect the performance of the ECCS.

(11) Each system of the ECCS, including flow rate and sensing networks, is capable of being tested during plant operation, including logic required to automatically initiate component action.

(12) Provisions for testing the ECCS network components (electronic, mechanical, hydraulic and pneumatic, as applicable) are installed in such a manner that they are an integral part of the design.

6.3.1.1.3 ECCS Requirements for Protection from Physical Damage

The ECCS piping and components are protected against damage from:

(1) Movement

(2) Thermal stresses

(3) Effects of the LOCA

(4) Effects of the safe shutdown earthquake

The ECCS is protected against the effects of pipe whip, which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints, or energy-absorbing materials if required. One of these three methods will be applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level.

The ECCS piping and components located outside the primary containment are protected from internally and externally generated missiles by the reinforced concrete structure of the Reactor Building ECCS pump rooms. In addition, the watertight construction of the ECCS pump rooms, when required, protects against mass flooding of redundant ECCS pumps.

Mechanical separation outside the drywell is achieved as follows:

(1) The ECCS shall be separated into three functional groups:

 (a) RCIC + 1 RHR + ADS

 (b) 1 HPCF + 1 RHR + ADS

(c) 1 HPCF + 1 RHR

(2) The equipment in each group shall be separated from that in the other two groups.

(3) Separation barriers shall be constructed between the functional groups, as required, to assure that environmental disturbances such as fire, pipe rupture, falling objects, etc., affecting one functional group will not affect the remaining groups.

6.3.1.1.4 ECCS Environmental Design Basis

Each ECCS has a safety-related injection/isolation testable check valve located in piping within the drywell, except RCIC and RHR Division A, which connect to feedwater lines outside the drywell. In addition, the RCIC System has an isolation valve in the drywell portion of its steam supply piping. The portions of ECCS piping and equipment located outside the drywell and within the secondary containment are qualified for the environmental conditions defined in Section 3.11.

6.3.1.2 Summary Descriptions of ECCS

The ECCS injection network is comprised of a RCIC System, a HPCF System, and a RHR System. These systems are briefly described here as an introduction to the more detailed system design descriptions provided in Subsection 6.3.2. The ADS, which assists the injection network under certain conditions, is also briefly described.

6.3.1.2.1 High Pressure Core Flooder

The HPCF System pumps water through a flooder sparger mounted above the reactor core. Coolant is supplied over the entire range of system operation pressures. The primary purpose of the HPCF System is to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel.

6.3.1.2.2 Residual Heat Removal

The RHR System has three independent loops and delivers water to the core at relatively low reactor pressures. The primary purpose of the RHR System is to provide inventory makeup and core cooling during large breaks and to provide containment cooling. Following ADS initiation, the RHR System provides inventory makeup following a small break.

6.3.1.2.3 Reactor Core Isolation Cooling

The RCIC System injects water into a feedwater line, using a pump driven by a steam turbine. The RCIC steam supply line branches off one of the main steamlines leaving the reactor pressure vessel and goes to the RCIC turbine. Makeup water is supplied from the condensate storage tank (CST) or the wetwell with the preferred source being the CST.

6.3.1.2.4 Automatic Depressurization System

The ADS utilizes a number of the reactor safety/relief valves (SRVs) to reduce reactor pressure during small breaks in the event of HPCF failure. When the vessel pressure is reduced to within the capacity of the low pressure system, these systems provide inventory makeup so that acceptable post-accident temperatures are maintained.

6.3.2 System Design

A more detailed description of the individual systems, including individual design characteristics of the systems, is provided in Subsections 6.3.2.1 through 6.3.2.4.

The following discussion provides details of the combined systems; in particular, those design features and characteristics which are common to all systems.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The P&IDs for the ECCS are identified in Subsection 6.3.2.2. The process diagrams which identify the various operating modes of each system are also identified in Subsection 6.3.2.2.

6.3.2.2 Equipment and Component Descriptions

The starting signal for the ECCS comes from four independent and redundant sensors of drywell pressure and low reactor water level. The ECCS is actuated automatically and requires no operator action during the first 30 min following the accident. The time sequence for starting of the systems for the limiting feed water line break mass inventory case is provided in Table 6.3-2.

Electric power for operation of the ECCS is from regular AC power sources. Upon loss of the regular power, operation is from onsite emergency standby AC power sources. Emergency sources have sufficient capacity so that all ECCS requirements are satisfied. Each of the three ECCS functional groups identified in Subsection 6.3.1.1.3(1) has its own diesel generator emergency power source. Section 8.3 contains a more detailed description of the power supplies for the ECCS.

Regulatory Guide 1.1 and 1.82 prohibit design reliance on pressure and/or temperature transients expected during a LOCA for assuring adequate NPSH. The requirements of these Regulatory Guides are applicable to the HPCF, RCIC and RHR pumps.

The BWR design conservatively assumes 0 kPaG containment pressure and maximum expected temperatures of the pumped fluids. Thus, no reliance is placed on pressure and/or temperature transients to assure adequate NPSH.

Requirements for NPSH are given in Tables 6.2-2c(HPCF), 5.4-1a (RCIC) and 6.2-2b (RHR). Vessel pressure versus system flow curves are given in Figures 6.3-4 (HPCF), 6.3-5 (RCIC) and 6.3-6 (RHR).

The design parameters for the HPCF and RHR System components are provided in Tables 6.3-8 and 6.3-9, respectively.

6.3.2.2.1 High Pressure Core Flooder (HPCF) System

The HPCF System is composed of two HPCF loops (B and C) flooding water to the RPV above the core. Each of the two loops belongs to a separate division; electrical and mechanical separation between the two divisions is complete. Physical separation is also assured by locating each division in a different area of the Reactor Building. The two loops are both high pressure pumping systems (i.e., they are capable of injecting water into the reactor vessel over the entire operating pressure range). Rated flow at both high and low pressure is the same for each loop. The piping and instrumentation diagram and process flow diagram are given in Figures 6.3-7 and 6.3-1, respectively.

The reference pressure for the operating performance of the system at high pressure is the lowest spring (safety) setpoint of the SRVs.

Both HPCF divisions take primary suction from the CST and secondary suction from the suppression pool. In the event CST water level falls below a predetermined setpoint or suppression pool water level rises above a predetermined setpoint, the pump suction will automatically transfer from the CST to the suppression pool. Both HPCF System loops have suction lines that are separate from RHR loops.

The HPCF pumps are located at an elevation which is below the water level in the suppression pool. This assures a flooded pump suction. The motor-operated valve in the suction line from the suppression pool on each division is normally closed, since primary suction is taken from the CST. This valve automatically opens on receipt of either of the suction transfer signals noted above. The suppression pool suction valves on each loop are capable of being closed from the control room if a leak develops in the system piping downstream of the isolation valves. Overpressure protection of the pump suction line is provided by a relief valve to the pump minimum flow line.

Each of the two high pressure flooder loops discharges water into the core via a separate flooder sparger. Internal piping connects each sparger to the vessel nozzle.

Each HPCF discharge line to the reactor is provided with two isolation valves in series. One of these valves is a testable check valve located inside the drywell as close as practical to the reactor vessel. HPCF injection flow causes this valve to open during LOCA conditions; thus, no power is required for valve actuation during LOCA. If an HPCF line should break outside the drywell, the check valve in that line inside the drywell will prevent loss of reactor water outside the drywell. The other isolation valve (which is also referred to as a HPCF injection valve) is a motor-operated gate valve located outside the drywell and as close as practical to the HPCF discharge line drywell penetration. This valve is capable of opening with the maximum pressure differential across the valve expected for any system operating mode including HPCF pump shutoff head. This valve is normally closed as a backup to the inside testable check valve for containment integrity purposes. A vent line is provided between the two valves. A normally open manual isolation valve inside the drywell is provided for HPCF loop maintenance during a plant refueling or maintenance outage.

For each loop, a full flow line is provided with discharge to the suppression pool to allow for full flow test of the system during normal operation. The valves in these lines are closed during normal operation. A full flow test return line is consistent with established BWR practice. There is no Regulatory Guide requiring this feature, but all BWRs have a 100% capacity test return line, and the Chapter 16 Technical Specifications specify periodic full flow system functional tests. There are no specific requirements for testing at runout flow; however, the system does have this capability.

For each loop, a pump minimum flow bypass line is also provided to return water to the suppression pool to prevent pump damage due to overheating when the injection valves on the main discharge lines are closed. The bypass line connects to the main discharge line between the main pump and the discharge check valve. A motor-operated valve on the bypass line automatically closes when sufficient flow in the main discharge line has been established. A flow element in the main discharge line measures system flow rate during LOCA and test conditions and automatically controls the motor-operated valve on the minimum flow bypass line.

The HPCF is designed to operate from normal offsite auxiliary power or from emergency diesel generators if offsite power is not available. If normal auxiliary power is lost, the onsite power source (diesel generator) for that division is started. The onsite power source for any division is capable of carrying all of the division emergency loads, including the HPCF pump and valve motors. Manually operated remote controls for system components (such as HPCF pumps, valves, etc.) and diesel generators are provided in the plant control room.

Full flow functional tests of the HPCF System can be performed during normal plant operation or during plant shutdown by manual operation of the HPCF System from the control room. For testing during normal plant operation, the pump suction is transferred to the suppression pool, the pump is started, and the test discharge line to the suppression pool is opened. A reverse sequence is used to terminate this test. Upon receipt of an automatic initiation signal while in the flow testing mode, the system will automatically return to injection mode and flow will be directed to the reactor vessel.

Appendix 6D outlines the HPCF flow analysis.

6.3.2.2.2 Automatic Depressurization System (ADS)

If the RCIC and HPCF Systems cannot maintain the reactor water level, the ADS, which is independent of any other ECCS, reduces the reactor pressure so that flow from the RHR System operating in the low pressure flooder mode enters the reactor vessel in time to cool the core and limit fuel cladding temperature.

The ADS employs nuclear system pressure relief valves to relieve high pressure steam to the suppression pool. The design number, location, description, operational characteristics and evaluation of the pressure relief valves are discussed in detail in Subsection 5.2.2. The instrumentation and controls for ADS are discussed in Subsection 7.3.1.1.1.2.

6.3.2.2.3 Reactor Core Isolation Cooling System (RCIC)

The RCIC System consists of a steam-driven turbine integral with a pump assembly. The system also includes piping, valves, and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the main steamlines (leaving the reactor pressure vessel) and goes to the RCIC turbine with drainage provision to the main condenser. The turbine exhausts to the suppression pool with vacuum breaking protection. Makeup water is supplied from the CST and the suppression pool with the preferred source being the CST. RCIC pump discharge lines include the main discharge line to the feedwater line, a test return line to the suppression pool, a minimum flow bypass line to the suppression pool. The piping configuration and instrumentation is shown in Figure 5.4-8. The process diagram is given in Figure 5.4-9.

Following the reactor scram, steam generation in the reactor core will continue at a reduced rate due to the core fission product decay heat. The turbine bypass system will divert the steam to the main condenser, and the feedwater system will supply the makeup water required to maintain reactor vessel inventory.

In the event that the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat. Upon reaching a predetermined low level, the RCIC System is initiated automatically. The turbine-driven pump will supply water from the suppression pool or from the CST to the reactor vessel. The turbine will be driven with a portion of the decay heat steam from the reactor vessel, and will exhaust to the suppression pool.

In the event that there is a LOCA, the RCIC System, in conjunction with the two HPCF Systems, is designed to pump water into the vessel while it is fully pressurized. This combination of systems will provide adequate core cooling until vessel pressure drops to the point at which the Low Pressure Flooder (LPFL) Subsystems of the RHR System can be placed in operation.

During RCIC operation, the suppression pool acts as the heat sink for steam generated by reactor decay heat. This will result in a rise in pool water temperature. Heat exchangers in the RHR System are used to maintain pool water temperature within acceptable limits by cooling the pool water directly during normal plant operation.

A design flow functional test of the RCIC System may be performed during normal plant operation by drawing suction from the suppression pool and discharging through a full flow test return line back to the suppression pool. The discharge valve to the vessel remains closed during the test, and reactor operation remains undisturbed. Should an initiation signal occur during test mode operation, flow will be automatically directed to the vessel. All components of the RCIC System are capable of individual functional testing during normal plant operation.

6.3.2.2.4 Residual Heat Removal System (RHR)

The RHR System is a closed system consisting of three independent pump loops which inject water into the vessel and/or remove heat from the reactor core or containment. Each of the pump loops contains the necessary piping, pumps valves, and heat exchangers. The piping and instrumentation diagram and process diagram are given in Figures 5.4-10 and 5.4-11, respectively. In the core cooling mode, each loop draws water from the suppression pool and injects the water into the vessel outside the core shroud (via the feedwater line on one loop and via the core cooling subsystem discharge return line on two loops). In the heat removal mode, pump suction may be taken either from the suppression pool or the reactor pressure vessel. With the pump suction being taken from the suppression pool, the pump discharge within these loops provides a flow path to the following points:

(1) Suppression pool

(2) Reactor pressure vessel (via feedwater on one loop and via the core cooling subsystem return lines on the other two loops)

(3) Wetwell and drywell spray spargers (on two loops only)

In the shutdown cooling mode, with the pump suction being taken from the reactor pressure vessel (via the shutdown cooling lines), the pump discharge within these loops provides a flow path back to the reactor vessel via the core cooling discharge return lines, and feedwater line, or to the upper reactor well via the fuel cooling system.

With the pump suction being taken from the skimmer surge tanks of the fuel pool cooling system, the pump discharge is returned to the fuel pool.

Each loop is in a single quadrant of the Reactor Building and receives its electric power from a bus separate from those serving the other two loops. Each bus is supplied from both onsite and offsite power sources.

For each loop, a full flow line is provided with discharge to the suppression pool to allow for full flow test of the system during normal operation. The valves in these lines are closed during normal operation.

For each loop, a minimum flow bypass line is also provided to return water to the suppression pool to prevent pump damage due to overheating when the injection valves on the main discharge lines are closed. The bypass line connects to the main discharge lines between the main pump and the discharge check valve. A motor-operated valve on the bypass line automatically closes when flow in the main discharge line is sufficient to provide the required pump cooling. A flow element in the main discharge line measures system flow rate during LOCA and test conditions and automatically controls the motor-operated valve on the bypass lines. The motor-operated valve does not receive an automatic signal to open unless the associated pump indicates a high discharge pressure.

Each loop contains instruments necessary to maintain a ready condition, to evaluate loop performance, and to operate the minimum flow valve.

Each RHR pump discharge line is maintained in a filled condition to minimize the time lag between a starting signal and initiation of flow into the reactor vessel and to minimize momentum forces associated with accelerating fluid into an empty pipe.

Each division is provided with a discharge line fill pump, which takes suction from the suppression pool suction line. A check valve is located in the discharge line at an elevation lower than the suppression pool minimum water level line to prevent backflow from emptying the lines into the suppression pool.

Full flow functional tests of the RHR System can be performed during normal plant operation or during plant shutdown by manual operation of the RHR System from the control room. For plant testing during normal plant operation, the pump is started and the return line to the suppression pool is opened. A reverse sequence is used to terminate this test. Upon receipt of an automatic initiation signal while in the flow testing mode, the system is returned to automatic control.

6.3.2.2.5 ECCS Discharge Line Fill System

A requirement of the core cooling systems is that cooling water flow to the reactor vessel be initiated rapidly when the system is called on to perform its function. This quick-start system characteristic is provided by quick-opening valves, quick-start pumps, and standby AC power source. The lag between the signal to start the pump and the initiation of flow into the RPV can be minimized by keeping the core cooling pump discharge lines full. Additionally, if these lines were empty when the systems were called for, the large momentum forces associated with accelerating fluid into a dry pipe could cause physical damage to the piping. Therefore, the ECCS discharge line fill system is designed to maintain the pump discharge lines in a filled condition.

Since the ECCS discharge lines are elevated above the suppression pool, check or stop-check valves are provided near the pumps to prevent back flow from emptying the lines into the suppression pool. Past experience has shown that these valves will leak slightly, producing a small back flow that will eventually empty the discharge piping. To ensure that this leakage from the discharge lines is replaced and the lines are always kept filled, a water leg pump is provided for each of the three RHR loops. The power supply to these pumps is classified as essential when the main ECCS pumps are deactivated. Indication is provided in the control room as to whether these pumps are operating, and alarms indicate low discharge line level. The RCIC loop and the two HPCF loops are maintained full by connection to the makeup water (condensate).

6.3.2.3 Applicable Codes and Classifications

The applicable codes and classification of the ECCS are specified in Section 3.2. The edition of the codes applicable to the design are provided in Table 1.8-21. The piping and components of each ECCS within containment and out to and including the pressure retaining injection valve are Safety Class 1. The remaining piping and components are Safety Class 2, 3, or noncode, as indicated in Section 3.2 and on the individual system P&ID. The equipment and piping of the ECCS are designed to the requirements of Seismic Category I. This seismic designation applies to all structures and equipment essential to the core cooling function. IEEE codes applicable to the controls and power supply are specified in Section 7.1.

6.3.2.4 Materials Specifications and Compatibility

Materials specifications and compatibility for the ECCS are presented in Section 6.1. Nonmetallic materials such as lubricants, seals, packings, paints and primers, insulation, as well as metallic materials, etc., are selected as a result of an engineering review and evaluation for compatibility with other materials in the system and the surroundings with concern for chemical, radiolytic, mechanical and nuclear effects. Materials used are reviewed and evaluated with regard to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the ECCS.

6.3.2.5 System Reliability

A single-failure analysis shows that no single failure prevents the starting of the ECCS, when required, or the delivery of coolant to the reactor vessel. No individual system of the ECCS is single-failure proof with the exception of the ADS; hence, it is expected that single failures will disable individual systems of the ECCS. The most severe effects of single failures with respect to loss of equipment occur if the LOCA occurs in combination with an ECCS pipe break coincident with a loss of offsite power. The consequences of the most severe single failures are shown in Table 6.3-3.

6.3.2.6 Protection Provisions

Protection provisions are included in the design of the ECCS. Protection is afforded against missiles, pipe whip and flooding. Also accounted for in the design are thermal stresses, loadings from a LOCA, and seismic effects.

The ECCS piping and components located outside the drywell are protected from internally and externally generated missiles by the reinforced concrete structure of the ECCS pump rooms. The watertight construction of these ECCS pump rooms also protects the equipment against flooding.

The ECCS is protected against the effects of pipe whip, which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints, and energy absorbing materials. One of these three methods is applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level (see Section 3.6 for criteria on pipe whip).

The component supports which protect against damage from movement and from seismic events are discussed in Subsection 5.4.14. The methods used to provide assurance that thermal stresses do not cause damage to the ECCS are described in Subsection 3.9.3.

6.3.2.7 Provisions for Performance Testing

Periodic system and component testing provisions for the ECCS are described in Subsection 6.3.2.2 as part of the individual system description.

6.3.2.8 Manual Actions

The ECCS is actuated automatically and requires no operator action during the first 30 min following the accident, although operator action is not prevented. During the long-term cooling period (after 10 min), containment cooling occurs as a normal consequence of RHR LPFL operation because the RHR heat exchangers are in series with the pumps. Although not prevented from doing so earlier, the operator is not required to select another RHR mode, such as suppression pool cooling, until after the 30 minutes.

The operator has multiple instrumentation available in the control room to assist him in assessing the post-LOCA conditions. This instrumentation provides reactor vessel pressures, water levels, containment pressure, temperature and radiation levels, as well as indicating the operation of the ECCS. ECCS flow indication is the primary parameter available to assess proper operation of the system. Other indications, such as position of valves, status of circuit breakers, and essential power bus voltage, are available to assist him in determining system operating status. The electrical and instrumentation complement to the ECCS is discussed in detail in Section 7.3. Other available instrumentation is listed in the P&IDs for the individual systems. Much of the monitoring instrumentation available to the operator is discussed in more detail in Chapter 5 and Section 6.2.

6.3.3 ECCS Performance Evaluation

Performance of the ECCS is determined by evaluating the system response to an instantaneous break of a pipe. This evaluation is performed using models either approved by the USNRC or which have met the change criteria in 10CFR50.46.

The analyses included in this subsection demonstrate the ABWR ECCS performance for the entire spectrum of postulated break sizes. The analyses are based upon the core loading shown in Figure 4.3-1 and were performed with the NRC-approved GOBLIN and CHACHA models. Plants with different core loadings, including those with some blank fuel bundles similar to that shown in Figure 4.3-2, will show the same system sensitivities. The MAPLHGR, peak cladding temperature (PCT) and oxidation fraction results will be provided for the limiting break (for each bundle design) and this meets the criteria documented in Appendix 4B. These results will be provided by the COL applicant to the USNRC for information. See Subsection 6.3.6.1 for COL license information.

The accidents, as listed in Chapter 15, for which ECCS operation is required are:

Subsection	Title
15.2.8	Feedwater Line Break
15.6.4	Spectrum of BWR Steam System Piping Failures Outside Containment
15.6.5	Loss-of-Coolant Accidents

Chapter 15 provides the radiological consequences of the above listed events for the core loading in Figure 4.3-1.

6.3.3.1 ECCS Bases for Technical Specifications

The MAPLHGRs calculated in this performance analysis provide the basis for the Chapter 16 Technical Specifications designed to ensure conformance with the acceptance criteria of 10CFR50.46. Minimum ECCS functional requirements are specified in Subsections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed in Subsection 6.3.4. Limits on minimum suppression pool water level are discussed in Section 6.2.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from 10CFR50.46 are listed, and, for each criterion, applicable parts of Subsection 6.3.3 (where conformance is demonstrated) are indicated.

Criterion 1: Peak Cladding Temperature (PCT)
"The calculated maximum fuel element cladding temperature shall not exceed 2200°F." Conformance to Criterion 1 (≤1204°C) is shown for the system response analyses in Subsections 6.3.3.7.3 (Break Spectrum), 6.3.3.7.4 (Line Breaks Inside Containment), 6.3.3.7.7 (Outside Containment Breaks) and specifically in Table 6.3-4 (Summary of LOCA Analysis Results). Conformance for each plant will be assured for the limiting break. See Subsection 6.3.6 for COL license information.

Criterion 2: Maximum Cladding Oxidation
"The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformance to Criterion 2 is shown in Table 6.3-4 (Summary of LOCA Analysis Results) for the system response analysis. This limit will be assured for the limiting break. See Subsection 6.3.6 for COL license information.

Criterion 3: Maximum Hydrogen Generation
"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is shown in Table 6.3-4 (Summary of LOCA Analysis Results) for the system analysis.

Criterion 4: Coolable Geometry
"Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 6.3-1, Section III.A, conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2.

Criterion 5: Long-Term Cooling
"After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is demonstrated generically for the ABWRs in Reference 6.3-1, Section 4.3.3.2. Briefly summarized, for any LOCA, the water level can be restored to a level above the top of the core and maintained there indefinitely.

6.3.3.3 Single-Failure Considerations

The functional consequences of potential operator errors and single failures (including those which might cause any manually controlled electrically operated valve in the ECCS to move to a position which could adversely affect the ECCS) and the potential for submergence of valve motors in the ECCS are discussed in Subsection 6.3.2. There it was shown that all potential single failures are no more severe than one of the single failures identified in Table 6.3-3.

It is therefore only necessary to consider each of these single failures in the ECCS performance analyses. The worst failure for any LOCA event is the failure of one of the diesel generators which provide electrical power to one HPCF and one RHR/LPFL. This failure results in the elimination of the greatest amount of flooding capability at both high and low reactor pressures.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as:

(1) Receiving an initiation signal

(2) A small lag time (to open all valves and have the pumps up to rated speed)

(3) The ECCS flow entering the vessel

Key ECCS actuation setpoints and time delays for all the ECCS systems are provided in Table 6.3-1. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on accelerating the diesel-generators and pumps. The delay time due to valve motion in the case of the high pressure system provides a suitably conservative allowance for valves available for this application. In the case of the low pressure system, the time delay for valve motion is such that the pumps are at rated speed prior to the time the vessel pressure reaches the pump shutoff pressure.

The ADS actuation logic includes a 30-second delay timer (including a 1 second instrument response time) to confirm the presence of Low Water Level 1 (LWL 1) initiation signal. This timer is initiated upon receipt of a high drywell pressure signal (which is sealed-in) and a LWL 1 signal. The timer setting is consistent with the startup time of the ECCS which also must be running before ADS operation can occur. Once the ADS timer is initiated, it is automatically reset if the reactor water level is restored above the LWL 1 setpoint before ADS operation occurs. For defense-in-depth protection against inventory decreasing events where a high drywell pressure is not present, the ADS actuation logic also includes an 8-minute high drywell pressure bypass timer. This timer is initiated upon receipt of a LWL 1 signal and is automatically reset if the reactor water level is restored above the LWL 1. After this timer runs out, the need for a high drywell pressure signal to initiate the ADS 30-second delay timer is bypassed (i.e., the 30-second delay timer would require only a LWL 1 signal to initiate). The ADS control system also provides the operator with an ADS inhibit switch which can be used to prevent automatic ADS operation as covered by the engineering operating procedures (refer to Subsection 7.3).

The flow delivery rates analyzed in Subsection 6.3.3 can be determined from the vessel pressure versus system flow curves in Figures 6.3-4, 6.3-5 and 6.3-6 and the pressure versus time plots discussed in Subsection 6.3.3.7. Simplified piping and instrumentation and process diagrams for the ECCS are referenced in Subsection 6.3.2. The operational sequence of ECCS for the limiting minimum mass inventory case is shown in Table 6.3-2.

Operator action is not required, except as a monitoring function, during the short-term cooling period following the LOCA. During the long-term cooling period, the operator may need to take action as specified in Subsection 6.2.2.2 to place the containment cooling system into operation for some LOCA events.

6.3.3.5 Use of Dual Function Components for ECCS

With the exception of the LPFL systems, the systems of the ECCS are designed to accomplish only one function: to cool the reactor core following a loss of reactor coolant. To this extent, components or portions of these systems (except for pressure relief) are not required for operation of other systems which have emergency core cooling functions, or vice versa. Because either the ADS initiating signal or the overpressure signal opens the safety-relief valve, no conflict exists.

The LPFL Subsystem is configured from the RHR pumps and some of the RHR valves and piping. When the reactor water level is low, the LPFL Subsystem (line up) has priority through the valve control logic over the other RHR Subsystems for containment cooling. Immediately following a LOCA, the RHR System is directed to the LPFL mode. When the RHR shutdown cooling mode is utilized, the transfer to the LPFL mode must be remote manually initiated.

6.3.3.6 Limits on ECCS Parameters

Limits on ECCS parameters are given in the sections and tables referenced in Subsections 6.3.3.1 and 6.3.3.7.1. Any number of components in any given system may be out of service, up to the entire system. The maximum allowable out-of-service time is a function of the level of redundancy and the specified test intervals.

6.3.3.7 ECCS Analyses for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

The methods used in the analysis have been approved by the NRC or meet the change criterion in 10CFR50.46. For the system response analysis, the GOBLIN and CHACHA models approved by the NRC were used. The significant input variables used for the response analysis are listed in Table 6.3-1.

6.3.3.7.2 Accident Description

The operation sequence of events for the limiting case is shown in Table 6.3-2.

6.3.3.7.3 Break Spectrum Calculations

A complete spectrum of postulated break sizes and locations were evaluated to demonstrate ECCS performance. For ease of reference, a summary of figures presented in Subsection 6.3.3.7 is shown in Table 6.3-5.

A summary of results of the break spectrum calculations is shown in tabular form in Table 6.3-4 and graphically in Figure 6.3-10. Conformance to the acceptance criteria (PCT= 1204°C, local oxidation = 17% and core-wide metal-water reaction = 1%) is demonstrated for the core loading in Figure 4.3-1. Results for the limiting break for each bundle design in a plant will be given for information to the USNRC by the COL applicant. See Subsection 6.3.6.3 for COL license information. Details of calculations for specific breaks are included in subsequent paragraphs

The ABWR design does not result in the hot assembly uncovering for any break in piping connected to the reactor pressure vessel (RPV), even when the limiting single active failure of ECCS equipment is assumed. The PCT in the ABWR design occurs during the rapid coastdown of the RIPs within the first few seconds, immediately after loss of power. Subsequent heatup for any break, while considering a single failure, remains bounded by the initial peak. As a result, the hot assembly power in the ABWR LOCA analysis may be established in a conservative manner as discussed in Reference 6.3-2. The process used is as follows:

The methodology assumes a symmetrical axial power shape (chopped cosine) with a 1.5 axial peaking factor. It is recognized that the axial power shape at the beginning of cycle is bottom peaked and that the peak power location generally moves upward during the cycle to become slightly top-peaked at the end of cycle. However, the axial shape used in the analysis is conservative and representative. The power at the hottest axial node is set as discussed in Reference 6.3-2. The resulting initial assembly power is considerably higher than any that would occur during operation. This is confirmed for each reload.

The ECCS performance MAPLHGR is calculated so that the hot rod does not violate the TMOL. Therefore, the MAPLHGR generated by the LOCA Analysis is limited by the TMOL and not by the ECCS performance.

The break spectrum analysis is performed using the predicted PCT in the GOBLIN hot assembly to identify the limiting break size/location case that will be subsequently evaluated in the CHACHA heatup calculation. The single failures of ECCS equipment have no impact on this analysis since the PCT occurs before any of the ECCS equipment has actuated. The break spectrum/single failure analysis results are also used to confirm that the performance of the ECCS equipment is sufficient to ensure that any uncovery is minimal and there is no appreciable cladding heat-up.

When the limiting break size/location has been determined, the boundary conditions from the hot assembly node applicable to the lattice being evaluated are applied to a CHACHA heatup calculation. The nodal power in the CHACHA heatup calculation is set as discussed in Reference 6.3-2. The resulting conditions from the analysis are shown to be less than the 10 CFR 50.46 criteria, with regard to PCT and maximum local oxidation.

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6.3.3.7.4 Line Breaks Inside Containment

Since the ABWR has no external recirculation lines, the largest line breaks are those postulated in a steam line, a feedwater line, or the RHR shutdown suction line. In comparison to external recirculation loop BWRs, the maximum ABWR line break size is only ~15% of the maximum double-ended recirculation line break. In addition, all of the large ABWR lines are located above the core. Breaks in smaller lines such as the RHR injection line, the HPCF injection line, and the bottom head drain line are also considered along with smaller breaks in the larger connecting lines. Table 6.3-4 lists the breaks considered inside the containment and the corresponding limiting single failure. Note that the limiting single failure is based on minimum inventory predicted during the event because failures in ECCS equipment do not impact the predicted PCT which occurs before any ECCS equipment is activated. The limiting single failure is the one that provides the least makeup when considering the impact of the break location on the ECCS. The breaks listed in Table 6.3-4 are piping systems with RPV penetrations. There are other branch lines that connect to these lines such as reactor water clean-up (RWCU), RCIC injection line, and the RCIC turbine steam supply line. Breaks in these lines are bounded by the break spectrum of lines that connect directly to the RPV.

A conservative assumption made in the ABWR LOCA analysis is that all offsite AC power is lost simultaneously with the initiation of the LOCA. In this case, four of the Reactor Internal Pumps (RIPs) will automatically trip off as a result of the turbine trip or the generator load rejection. The remaining six RIPs powered by M/G sets are capable of operating. (See Section 15.2.6). However, in the analysis, all of the RIPs are assumed to coast down rapidly as a result of the loss of power. The resulting rapid core flow coastdown produces a calculated departure from nucleate boiling in the hot assemblies within a fraction of a second after the accident. If offsite power were available, the four RIPs would continue to run until a LWL-3 signal is generated, the six RIPs would operate until a LWL-2 signal is generated and the pump coastdown would not begin until after the reactor scram.

Table 6.3-4a gives the summary results from the Reference 6.3-1 sensitivity studies. Figures 6.3-12 through 6.3-45, with legends provided in Table 6.3-4a, show the following key output variables from those studies:

- Dome Pressure
- System Response PCT
- System Mass
- ECCS Flow Rate
- Upper Plenum Void
- Hot Assembly Exit Quality
- Fuel Assembly Exit Cladding Temperature vs. Channel Peaking Factor

- System Response PCT vs. Channel Peaking Factor

6.3.3.7.5 Not Used

6.3.3.7.6 Small Line Breaks Inside Containment

A break in a reactor internal pump would involve either the welds or the casing. If the weld from the pump casing to the RPV stub tube breaks, the stretch tube will prevent the pump casing from moving. The stretch tube clamps the diffuser to the pump casing, where its nut seats. The land is located below the casing attachment weld and therefore the stretch tube forms a redundant parallel strength path to the pump casing restraint which is designed to provide support in the event of weld failure. In case the pump casing and the stretch tube break, the pump and motor will move downward until stopped by the casing restraints. The pump is part of the stretch tube. In either case, the break flow would be much less than the drain line break case. Therefore, the drain line break analysis is also bounding for any credible break within the reactor internal pump recirculation system and its associated motor housing and cover.

6.3.3.7.7 Line Breaks Outside Containment

Breaks outside of containment are characterized by isolation of the break by the MSIVs. Since the main steam line break outside the containment produces more vessel inventory loss before isolation than other breaks in the category, the results of this case are bounding for breaks in this group.

A postulated guillotine break of one of the four main steam lines outside the containment results in mass loss from each end of the break until the MSIVs close. The MSIVs receive a close signal due to high steam flow rate through the integral flow restrictors or due to LWL 1.5. Closure of the MSIVs limits the amount of flow that will be discharged outside the containment. Once the MSIVs close, the RPV will pressurize until SRVs open. The SRVs will control system pressure and discharge steam to the suppression pool. The RCIC system or one of the HPCF systems can provide adequate flow to maintain core cooling.

The limiting single failure is that one of the EDGs that provides power to HPCF pump does not start. This results in one RCIC pump, one HPCF pump, and two LPFL pumps available to mitigate the event. Although the RCIC turbine takes suction from one the of the steam lines, closure of the MSIVs ensures that there is a long-term supply of steam for the RCIC turbine. However, a more limiting case is evaluated where the RCIC is assumed unavailable. In this case the assumed available equipment is:

1HPCF + 2LPFL + 8 ADS

In this case, the LPFL systems would not actuate as the system inventory is stabilized before ADS actuation.

Important output variables from the sensitivity study of these events are shown in Figures 6.3-46 and 6.3-47.

6.3.3.7.8 Bounding Calculations

Case with Minimum Inventory

The case having the minimum system inventory was the feedwater line break initiated with maximum core flow rate. Although the steam line break has a greater flow area than the feedwater line break, the feedwater line break has the smallest minimum inventory due to the lower quality fluid flowing out the break. In spite of having the least minimum inventory, the core remains cooled by a two phase mixture throughout the transient.

Case with Maximum Peak Cladding Temperature

There is no clear case having the highest peak cladding temperature from the GOBLIN hot assembly analysis. As shown in Table 6.3-4, two cases have a PCT of 710°C as predicted by GOBLIN. The small feedwater line break initiated from 90% core flow rate, is selected for the heatup analysis using CHACHA.

Figure 6.3-48 shows the results of a calculation for a typical lattice design over a range of burnups. The PCT of 850°C remains well below the licensing limit for ECCS performance analysis of 1204°C.

Due to the relatively low PCT, the maximum local oxidation and core wide oxidation are below the licensing limits.

A similar evaluation is performed for each reload to confirm that the 10 CFR 50.46 criteria are met when the nodal power is selected as discussed in Reference 6.3-2.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Subsection 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the criteria in Appendix 4B.

6.3.3.9 LOCA Analyses to Support ECCS Technical Specifications for Allowable Outage Times

Core cooling LOCA analyses covering the complete spectrum of postulated breaks were performed with only one RHR system in the low pressure flooder (LPFL) mode and 5 ADS valves available. These calculations are also bounding for the case with one HPCF subsystem and 5 ADS valves available since, compared to the RHR/LPFL, the HPCF has the additional capacity to inject at high reactor system pressures.

For these analyses, the Appendix K LOCA evaluation assumptions and models were used consistent with the LOCA analyses presented in Subsection 6.3.3.7, with one exception. This exception is the availability of only 1 RHR/LPFL and 5 ADS valves, which goes beyond the single failure criteria and represents multiple failures or systems out of service and provides the analytical basis for the allowable outage times for ECCS equipment in Technical Specification 3.5.1. The results for these analyses met all 10CFR50.46 acceptance criteria.

Tables 6.3-10 through 6.3-15 provide the basis for the selection of the emergency core cooling systems remaining assuming various combinations of single failures and systems out of service.

6.3.3.10 Severe Accident Considerations

In the unlikely event that the ECCS does not prevent core damage, its operation (recovery if necessary) can be beneficial in mitigating the consequences of core damage. The analysis of core damage events was performed using best-estimate methods rather than design basis codes.

The primary injection path for the RHR System during a severe accident is into the vessel via the LPFL header. The conditions under which the LPFL should be used are described in the Appendix 18A, Emergency Procedure Guidelines. For injection to occur, the RPV must be at low pressure.

If the LPFL is not initiated in time to prevent core damage, LPFL is still beneficial by enhancing cooling and preventing radioactive heating from the core debris. If injection is initiated prior to vessel failure, melt progression may be arrested in-vessel. However, if vessel failure occurs, debris will relocate from the vessel breach into the lower drywell. Water flowing into the lower drywell will cover the core debris and enhance debris cooling.

6.3.4 Tests and Inspections

6.3.4.1 ECCS Performance Tests

All systems of the ECCS are tested for their operational ECCS function during the preoperational and/or startup test program. Each component is tested for power source, range, direction of rotation, setpoint, limit switch setting, torque switch setting, etc. Each pump is tested for flow capacity for comparison with vendor data. (This test is also used to verify flow measuring capability). The flow tests involve the same suction and discharge source (i.e., suppression pool).

All logic elements are tested individually and then as a system to verify complete system response to emergency signals including the ability of valves to revert to the ECCS alignment from other positions.

Finally, the entire system is tested for response time and flow capacity taking suction from its normal source and delivering flow into the reactor vessel. This last series of tests is performed with power supplied from both offsite power and onsite emergency power.

See Chapter 14 for a thorough discussion of preoperational testing for these systems. See Subsection 6.3.6.2 for COL license information regarding ECCS testing requirements.

6.3.4.2 Reliability Tests and Inspections

The average reliability of a standby (nonoperating) safety system is a function of the duration of the interval between periodic functional tests. The factors considered in determining the periodic test interval of the ECCS are:

(1) The desired system availability (average reliability)

(2) The number of redundant functional system success paths

(3) The failure rates of the individual components in the system

(4) The schedule of periodic tests (simultaneous versus uniformly staggered versus randomly staggered)

All of the active components of the HPCF, ADS, RHR and RCIC Systems are designed so that they may be tested during normal plant operation. Full flow test capability is provided by a test line back to the suction source. The full flow test is used to verify the capacity of each ECCS pump loop while the plant remains undisturbed in the power generation mode. In addition, each individual valve may be tested during normal plant operation.

All of the active components of the ADS System, except the SRVs and their associated solenoid valves, are designed so that they may be tested during normal plant operation. The SRVs and associated solenoid valves are all tested during plant initial power ascension per Regulatory Guide 1.68, Appendix A. SRVs are bench tested to establish lift settings.

Testing of the initiating instrumentation and controls portion of the ECCS is discussed in Subsection 7.3.1. The emergency power system, which supplies electrical power to the ECCS in the event that offsite power is unavailable, is tested as described in Subsection 8.3.1. The frequency of testing is specified in the Chapter 16 Technical Specifications. Visual inspections of all the ECCS components located outside the drywell can be made at any time during power operation. Components inside the drywell can be visually inspected only during periods of access to the drywell. When the reactor vessel is open, the spargers and other internals can be inspected.

6.3.4.2.1 HPCF Testing

The HPCF System can be tested at full flow with suppression pool water at any time during plant operation except when a system initiation signal is present. If an initiation signal occurs while the HPCF is being tested, the system returns automatically to the operating mode. The motor-operated valve in the suction line from the condensate storage tank is interlocked closed when the suction valve from the suppression pool is open.

A design flow functional test of the HPCF System over the operating pressure and flow range is performed by pumping water from the suppression pool through the full flow test return line and back to the suppression pool.

The suction valve from the condensate storage tank and the discharge valve to the reactor remain closed. These two valves are tested separately to ensure their operability.

The HPCF test conditions are tabulated on the HPCF process flow diagram (Figure 6.3-1). Appendix 6D provides test and analysis outlines.

6.3.4.2.2 ADS Testing

An ADS logic system functional test and simulated automatic operation of all ADS logic channels are to be performed at least once per plant operating interval between reactor refuelings. Sensor and logic channels are demonstrated operable by the performance of a divisional functional test and a trip unit calibration at least once per month and a transmitter calibration at least once per operating interval.

All SRVs, which include those used for ADS, are bench tested to establish lift settings in compliance with ASME Code Section XI.

6.3.4.2.3 RHR Testing

The RHR pump and valves are tested periodically during reactor operation. With the injection valves closed and the return line open to the suppression pool, full flowing pump capability is demonstrated. The injection valve and the check valve are tested in a manner similar to that used for the HPCF valves. The system test conditions during reactor operation are shown on the RHR System process diagram (Figure 5.4-11).

6.3.4.2.4 RCIC Testing

The RCIC loop can be tested during reactor operation. To test the RCIC pump at rated flow, the test bypass line valve to the suppression pool and the pump suction valve from the suppression pool are opened and the pump is started using the turbine controls in the control room. Correct operation is determined by observing the instruments in the control room.

If an initiation signal occurs during the test, the RCIC System returns to the operating mode. The valves in the test bypass lines are closed automatically and the RCIC pump discharge valve is opened to assure flow is correctly routed to the vessel.

6.3.5 Instrumentation Requirements

Design details including redundancy and logic of the ECCS instrumentation are discussed in Section 7.3.

All instrumentation required for automatic and manual initiation of the HPCF, RCIC, RHR and ADS Systems is discussed in Subsection 7.3.1, and is designed to meet the requirements of IEEE-603 and other applicable regulatory requirements. The HPCF, RCIC, RHR and ADS Systems can be manually initiated from the control room.

The RCIC, HPCF, and RHR Systems are automatically initiated on low reactor water level or high drywell pressure. The ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus indication that at least one RHR or HPCF pump is operating. The HPCF, RCIC, and RHR Systems automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The RHR LPFL mode injection into the RPV begins when reactor pressure decreases to the RHR's pump discharge shutoff pressure.

HPCF injection begins as soon as the HPCF pump is up to speed and the injection valve is open, since the HPCF System is capable of injection water into the RPV over a pressure range from 8.12 to 0.69 MPaD or pressure difference between the vessel and drywell.

6.3.6 COL License Information

6.3.6.1 ECCS Performance Results

The exposure-dependent MAPLHGR, peak cladding temperature, and oxidation fraction for each fuel bundle design based on the limiting break size will be provided by the COL applicant to the USNRC for information (Subsection 6.3.3).

6.3.6.2 ECCS Testing Requirements

In accordance with the Technical Specifications, the COL applicant will perform a test every refueling in which each ECCS subsystem is actuated through the emergency operating sequence (Subsection 6.3.4.1).

6.3.6.3 Limiting Break Results

Results for the limiting break for each bundle design will be provided to the USNRC by the COL applicant (Subsection 6.3.3.7.3).

6.3.7 Reference

6.3-1 WCAP-17116-P and WCAP-17116-NP,"Westinghouse BWR ECCS Evaluation Model: Supplement 5 - Application to the ABWR," September 2009, to be approved by NRC.

6.3-2 "Supplemental Information for Toshiba ABWR DCD Renewal Amendment," (WCAP-17290-P, Rev. 0).

Table 6.3-1 Significant Input Variables Used in the Loss-of-Coolant Accident Analysis

Variable	Units	Value
A. Plant Parameters		
Core Thermal Power	MWt	4005
Vessel Steam Output	kg/h	7.82×10^6
Corresponding Percentage of Rated Steam Flow	%	102.4
Vessel Steam Dome Pressure	MPaG	7.17
B. Emergency Core Cooling Systems Parameters		
B.1 Low Pressure Flooder System		
Vessel Pressure at which Flow may Commence	MPaD (vessel to drywell)	1.55
Minimum Rated Flow per system at Vessel Pressure	m^3/h MPaD (vessel to drywell)	954 0.275
Initiating signals Low Water Level or High Drywell Pressure	cm above vessel zero MPaG	≤920.25 ≥0.014
Setpoint of Low Pressure Permissive for LPFL Injection Valve Open	MPaG	3.01
Maximum Allowable Time Delay from Low Pressure Permissive Signal to Injection Valve Fully Open	s	37.0
B.2 Reactor Core Isolation Cooling System		
Vessel Pressure at which flow may commence	MPaD (vessel to pump suction)	8.12
Minimum Rated Flow at Vessel Pressure	m^3/h MPaD (vessel to the air space of the compartment containing the water source for the pump suction)	182 1.035 to 8.12
Initiating signals Low Water Level or High Drywell Pressure	cm above vessel zero MPaG	≤1148.35 ≥0.014
Maximum Allowable Time Delay from Initiating Signal to Rated Flow Available and Injection Valve Fully Open	s	30.0

Table 6.3-1 Significant Input Variables Used in the Loss-of-Coolant Accident Analysis (Continued)

Variable	Units	Value
B.3 High Pressure Core Flooder System		
Vessel Pressure at which Flow May Commence	MPaD	8.12
Minimum Rated Flow per System Available at Vessel Pressure	m^3/h MPaD (vessel to the air space of the compartment containing the water source for the pump suction)	182 to 727 0.69 to 8.12
Initiating Signals	cm above vessel zero	≤1003.65
Low Water Level or High Drywell Pressure	MPaD	≥0.014
Maximum Allowable Time Delay from Initiating Signal to Rated Flow Entering the Reactor Vessel Consistent with Figure 6.3-4	s	37.0
B.4 Automatic Depressurization System		
Total Number of Relief Valves with ADS Function		8
Total Minimum Flow Capacity	kg/h	2.903×10^6
At Vessel Pressure	MPaG	7.76
Initiating Signals Low Water Level and High Drywell Pressure or High Drywell Pressure Bypass Timer Timed Out	 cm above vessel zero MPaG s	 ≤920.25 ≥0.014 ≤480
Delay Time from All Initiating Signals Completed to the Time Valves are Open	s	≤30.25
C. Fuel Parameters*		
Fuel Type	-------	Initial Core
Fuel Bundle Geometry	-------	10x10
Lattice	-------	C
Number of Fueled Rods per Bundle	-------	96
Peak Technical Specification Linear Heat Generation Rate	kW/m	43.0
Initial Minimum Critical Power Ratio	-------	TMOL Limit
Design Axial Peaking Factor	-------	1.5

* The system response analysis is based upon the core loading in Figure 4.3-1. The sensitivities demonstrated are valid for other core loadings.

Table 6.3-2 Operational Sequence of Emergency Core Cooling System Feed Water Line Break for Minimum Mass Inventory

Time, Seconds	Events
0.00	Break occurs
0.00	Loss of Offsite Power
0.00	RIPs start to coast down
0.00	TCVs start to close
0.00	Feedwater starts to ramp down
0.08	Reactor trip on LWL-3
0.075	TCVs closed
1.00	Feedwater flow goes to zero
4.08	Maximum PCT
6.7	LWL-2 reached
15.6	LWL-1.5 reached
66.6	LWL-1 reached
88.9	HPCF starts
96.8	ADS flow starts
137.6	LPFL flow starts
165.7	Minimum Inventory
280	End of Transient

Table 6.3-3 Single Failure Evaluation*

Assumed Failure	Systems Remaining†
Emergency Diesel Generator A	All ADS, RCIC, 2 HPCF, 2 RHR/LPFL
Emergency Diesel Generator B or C	All ADS, RCIC, 1 HPCF, 2 RHR/LPFL
RCIC Injection Valve	All ADS, 2 HPCF, 3 RHR/LPFL
One ADS Valve	All ADS minus one, RCIC, 2 HPCF, 3 RHR/LPFL

* Single, active failures are considered in the ECCS performance evaluation. Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above designed failures.

† Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For the LOCA from an ECCS line break, the systems remaining are those listed, less the ECCS system in which the break is assumed.

Table 6.3-4 Summary of Results of LOCA Analysis

Break Location	Break Size (cm^2)	Systems Available	PCT (°C)	Maximum Local Oxidation* (%)	Minimum System Mass (10^3 kg)
Based on Appendix K evaluation models:					
Steamline Inside Containment	985	1HPCF + RCIC +2 RHR/LPFL + 8 ADS	657	<< 17.0	162.6
Feedwater Line	839	1 HPCF + 2 RHR/LPFL + 8 ADS	710	<< 17.0	123.5
RHR Shutdown Cooling Suction Line	792	1 HPCF + RCIC + 2 RHR/LPFL+ 8 ADS	710	<< 17.0	132.1
RHR/LPFL Injection Line	205	1 HPCF + RCIC + 1RHR/LPFL + 8 ADS	707	<< 17.0	213.1
High Pressure Core Flooder	92	RCIC+2RHR/ LPFL + 8 ADS	708	<< 17.0	132.2
Bottom Head Drain Line	20.3	1HPCF + RCIC + 2 RHR/LPFL + 8 ADS	708	<< 17.0	247
Steamline Outside Containment	3939	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	668	<< 17.0	241.4

* The core-wide metal water reaction for this analysis has been calculated to be <0.2%.

Table 6.3-4a Summary of Break Spectrum Study Results (from Ref. 6.3-1)

Case	Core Flow	Break Location	Break Size	Steam Line Isolation	PCT (GOBLIN)	Minimum Mass
hpcf3	90%	HPCF Line	100%	TCV fast closure	708°C	133.3 E3 kg
hpcf4	111%	HPCF Line	100%	TCV fast closure	692°C	132.2 E3 kg
hpcf5	90%	HPCF Line	100%	Pressure regulator	661°C	133.7 E3 kg
hpcf7	90%	HPCF Line	75%	TCV fast closure	708°C	138.1 E3 kg
hpcf8	90%	HPCF Line	50%	TCV fast closure	708°C	143.6 E3 kg
hpcf9	90%	HPCF Line	25%	TCV fast closure	708°C	151.3 E3 kg
mslb6	90%	SL – RCIC side	200%	TCV fast closure	657°C	164.1 E3 kg
mslb6a	111%	SL – RCIC side	200%	TCV fast closure	648°C	162.6 E3 kg
mslb7	90%	SL – RCIC side	150%	TCV fast closure	654°C	164.1 E3 kg
mslb8	90%	SL – RCIC side	100%	TCV fast closure	656°C	164.1 E3 kg
fwlb3	90%	FWL – RCIC side	100%	TCV fast closure	708°C	126.5 E3 kg
fwlb4	111%	FWL – RCIC side	100%	TCV fast closure	684°C	123.5 E3 kg
fwlb5	90%	FWL – RCIC side	100%	Pressure Regulator	661°C	126.0 E3 kg
fwlb6	90%	FWL – RCIC side	100%	TCV fast closure	708°C	125.9 E3 kg
fwlb7	90%	FWL – RCIC side	75%	TCV fast closure	705°C	136.9 E3 kg
fwlb8	90%	FWL – RCIC side	50%	TCV fast closure	707°C	148.4 E3 kg
fwlb9	90%	FWL – RCIC side	25%	TCV fast closure	710°C	215.6 E3 kg
rhrlb3dlb	90%	RHR Suction Line	100%	TCV fast closure	708°C	133.6 E3 kg
rhrlb4dlb	111%	RHR Suction Line	100%	TCV fast closure	691°C	132.1 E3 kg
rhrlb5dlb	90%	RHR Suction Line	100%	Pressure regulator	662°C	133.3E3 kg
rhrlb7dlb	90%	RHR Suction Line	75%	TCV fast closure	709°C	145.2 E3 kg
rhrlb8dlb	90%	RHR Suction Line	50%	TCV fast closure	710°C	189.7 E3 kg
rhrilb3	90%	RHR Injection Line	100%	TCV fast closure	707°C	215.0 E3 kg
rhrilb4	111%	RHR Injection Line	100%	TCV fast closure	655°C	213.1 E3 kg
dlb	90%	Drain Line	100%	TCV fast closure	708°C	247.0 E3 kg
mslboc6a	90%	SL Outside Containment	200%	MSIV closure	668°C	241.4 E3 kg

Note: Case numbers are shown on Figures 6.3-12 through 6.3-48

Table 6.3-5 Key to Figures

Break	Figures
High Pressure Core Flooder	6.3-12 through 6.3-21
Main Steamline Inside Containment	6.3-22 through 6.3-28
Feedwater Line	6.3-29 through 6.3-37
RHR Line	6.3-38 through 6.3-43
Bottom Head Drain Line	6.3-44 through 6.3-45
Main Steamline Outside Containment	6.3-46 through 6.3-47

Table 6.3-6 Not Used

Table 6.3-7 Not Used

Table 6.3-8 Design Parameters for HPCF System Components

(1) Main Pumps (C001)	
Number of Pumps	2
Pump Type	Centrifugal
Drive Unit	Constant speed induction motor
Flow Rate	182 m^3/h @ 8.22 MPaA reactor pressure*
	727 m^3/h @ 0.79 MPaA reactor pressure*
Developed Head	890m @ 8.22 MPaA reactor pressure
	190m @ 0.79 MPaA reactor pressure
Maximum Runout Flow	890 m^3/h @ 0.10 MPaA reactor pressure
Minimum Bypass Flow	73 m^3/h
Water Temperature Range	10° to 100°C*
NPSH Required	1.7m
(2) Strainer (D001)	
Location	Suppression Pool
Size	As required for insulation debris per Appendix 6C.
(3) Restricting Orifice (D002)	
Location	Pump discharge line
Size	Limit pump flow to values specified
(4) Condensate Storage Tank	570 m^3 reserve storage for HPCF and RCIC Systems combined
(5) Flow Elements (FE008)	
Location	Pump discharge—downstream of minimum flow bypass line
Head Loss	6.1m w.g. maximum @ 727 m^3/h
Accuracy	±2.5% combined element, transmitter and indicator at maximum rated
(6) Core Flooder Sparger	
Flow Rate	727 m^3/h minimum @ 0.79 MPaA reactor pressure
Pressure Drop	50m w.g. maximum @ 727 m^3/h
(7) Piping and Valves	
Design Pressures	0.31 MPaG—suction and discharge connected to suppression pool
	2.82 MPaG—pump suction
	10.79 MPaG—pump discharge

Table 6.3-8 Design Parameters for HPCF System Components (Continued)

Design Temperatures	66°C – condensate tank suction 100°C – pump suction and discharge 302°C – discharge to vessel
(8) Valve Operation	
Pump Suction Valve, Suppression Pool (F006)	Normally closed, opens on low water level in condensate storage tank or high water level in the suppression pool.
Pump Suction Check Valve, Suppression Pool (F007)	Prevents backflow into suppression pool.
Pump Suction Valve, Condensate Tank (F001)	Normally open, closes when F006 is fully open.
Pump Suction Check Valve, Condensate Tank (F002)	Prevents backflow into condensate storage tank.
Pump Discharge Valve, Reactor Injection Valve (F003)	Normally closed, opens within 36 seconds after initiation signal including D/G loading sequence time.
Testable Check Valve, Reactor Injection Line (F004)	Prevent loss of coolant outside drywell for line break.
Maintenance Valve, Reactor Injection Line (F005)	Normally open, used to isolate system from reactor for maintenance purposes.
Pump Test Line Valves (F008, F009)	Normally closed, throttle valves used to test system flow at rated and runout conditions.
Pump Minimum Flow Line Valve (F010)	Normally closed, opens on signal when pump discharge pressure is high and low flow through flow meter. Used to protect pump from overheating.

* The HPCF System has the capability to deliver at least 50% of these flow rates with 171°C water at the pump suction.

Table 6.3-9 Design Parameters for RHR System Components

(1) Main Pumps (C001)	
Number of Pumps	3
Pump Type	Centrifugal
Drive Unit	Constant Speed Induction Motor
Flow Rate	954 m^3/h
Developed Head	125m
Maximum Runout Flow	1130 m^3/h
Maximum Bypass Flow	148 m^3/h
Minimum Shutoff Head	195m
Maximum Pump Brake Horsepower	550 kW
Water Temperature Range	10° to 182°C
NPSH Required	2.0m
Saturated Water in Suppression Pool Pumped Over Pressure Range	0 to 0.62 MPaG
(2) Heat Exchangers (B001)	
Number of units	3
Seismic	Category I design and analysis
Types of exchangers	U-Tube/Shell
Maximum primary side pressure	3.43 MPaG
Design Point Function Cooling	Reactor Shutdown
Primary side (tube side) performance data	
(1) Flow	954 m^3/h
(2) Inlet temperature	182°C maximum
(3) Allowable pressure drop (Max)	7.0m w.g.
(4) Type water	Suppression Pool or Reactor Water
(5) Fouling factor	0.0005
Secondary side (shell side) performance data	
(1) Flow	1200 m^3/h
(2) Inlet temperature	43°C maximum
(3) Allowable pressure drop (Max)	7.0m w.g.
(4) Type water	Reactor Building Cooling Water

Table 6.3-9 Design Parameters for RHR System Components (Continued)

(5) Fouling factor	0.0005
(3) Strainer (D008)	
Location	Suppression Pool
Size	As required for insulation debris per Appendix 6C
(4) Restricting Orifices	
Location (D003)	Vessel return line
Size	Limit flow to vessel to 954 m^3/h
Location (D002)	Suppression pool return line
Size	Limit flow during suppression pool cooling to 954 m^3/h
Location (D004)	Fuel pool return line
Size	Limit flow during fuel pool cooling to 350 m^3/h
Location (D00I)	Pump minimum flow line
Size	Limit pump flow through the bypass line to 148 m^3/h
Location (D005)	Discharge line to wetwell spray
Size	Limit wetwell spray sparger flow to 114 m^3/h
Location (D006)	Discharge line to drywell sparger
Size	Limit drywell spray sparger flow to 840 m^3/h
(5) Flow Elements (FE009)	
Location	Pump discharge line, downstream of heat exchanger bypass return
Rated Flow	954 m^3/h
Head Loss	6.1m w.g. maximum @ 954 m^3/h
Accuracy	±2.5% combined element, transmitter and indicator at rated flow
(6) Vessel Flooder Sparger	
Flow Rate	954 m^3/h
Minimum Exit Velocity	11 m/s @ 954 m^3/h
(7) Wetwell Spray Sparger	
Flow Rate	114 m^3/h
(8) Drywell Spray Sparger	
Flow Rate	840 m^3/h

Table 6.3-9 Design Parameters for RHR System Components (Continued)

(9) Piping and Valves	
Design Pressures	0.31 MPaG—discharge piping connected to suppression pool
	0.31 MPaG—suction piping connected to suppression pool
	3.43 MPaG—wetwell and drywell sparger piping
	2.82 MPaG—pump suction piping
	3.43 MPaG—pump discharge piping
	8.62 MPaG—vessel suction and return piping
Design Temperatures	104°C—suppression pool piping and wetwell sparger piping
	171°C—drywell sparger piping
	182°C —pump suction and discharge piping
	302°C—vessel suction and return piping
(10) Valve Operation	
See Table 5.4-3, RHR Pump/Valve Logic	

Table 6.3-10 Not Used

Table 6.3-11 Not Used

Table 6.3-12 Not Used

Table 6.3-13 Not Used

Table 6.3-14 Not Used

Table 6.3-15 Not Used

The following figures are located in Chapter 21:

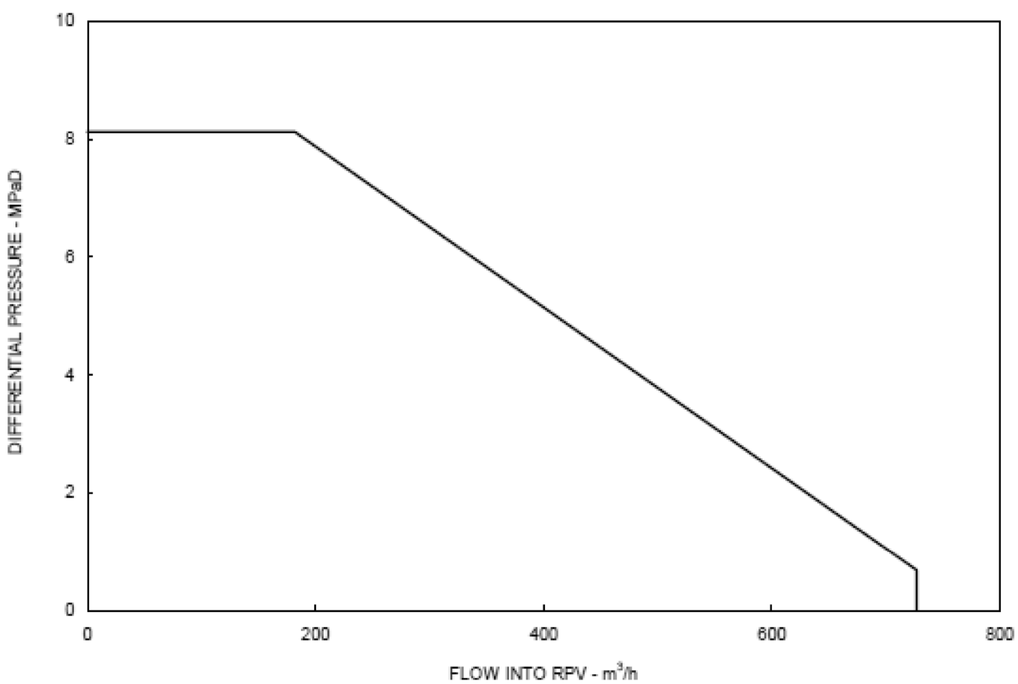


Figure 6.3-4 Pressure Versus High Pressure Core Flooder Flow (Per System) Used in LOCA Analysis

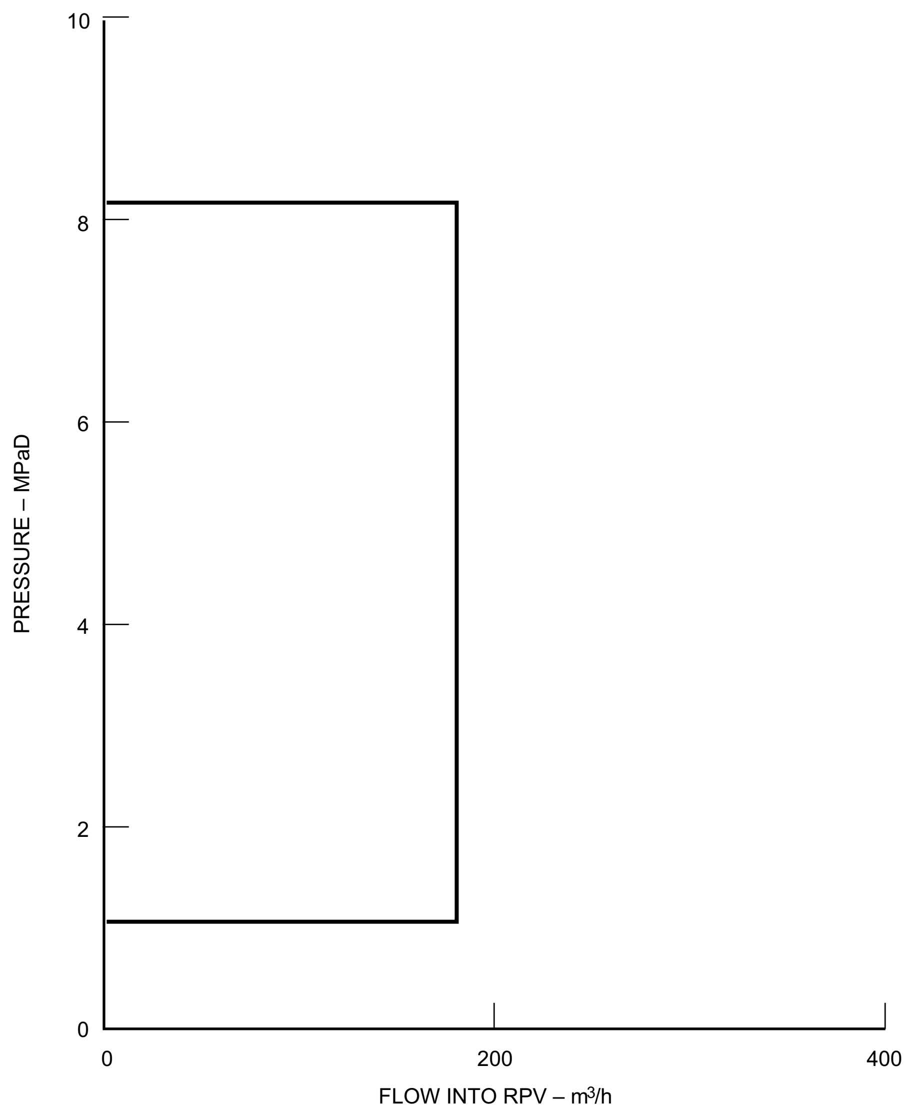


Figure 6.3-5 Pressure Versus Reactor Core Isolation Cooling Flow Used in LOCA Analysis

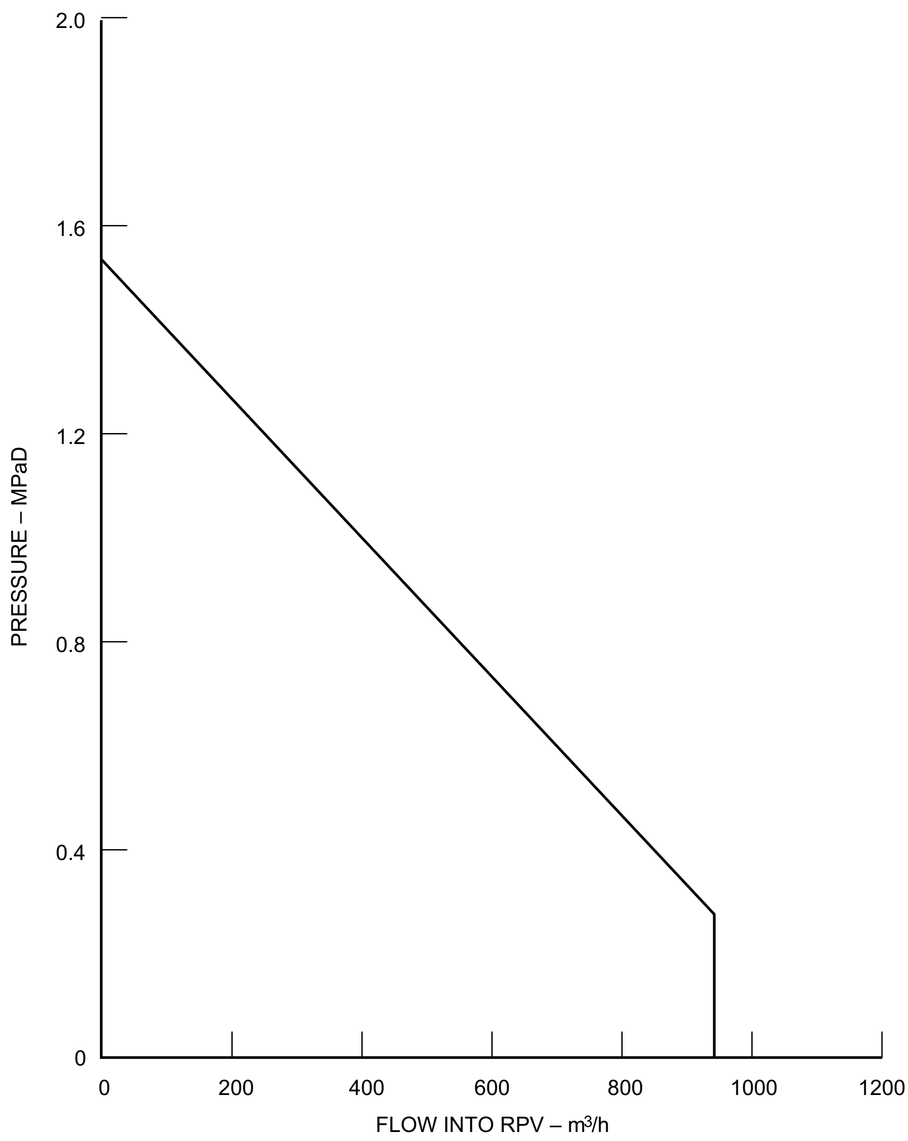


Figure 6.3-6 Pressure Versus Low Pressure Flooder Flow (Per System) Used in LOCA Analysis

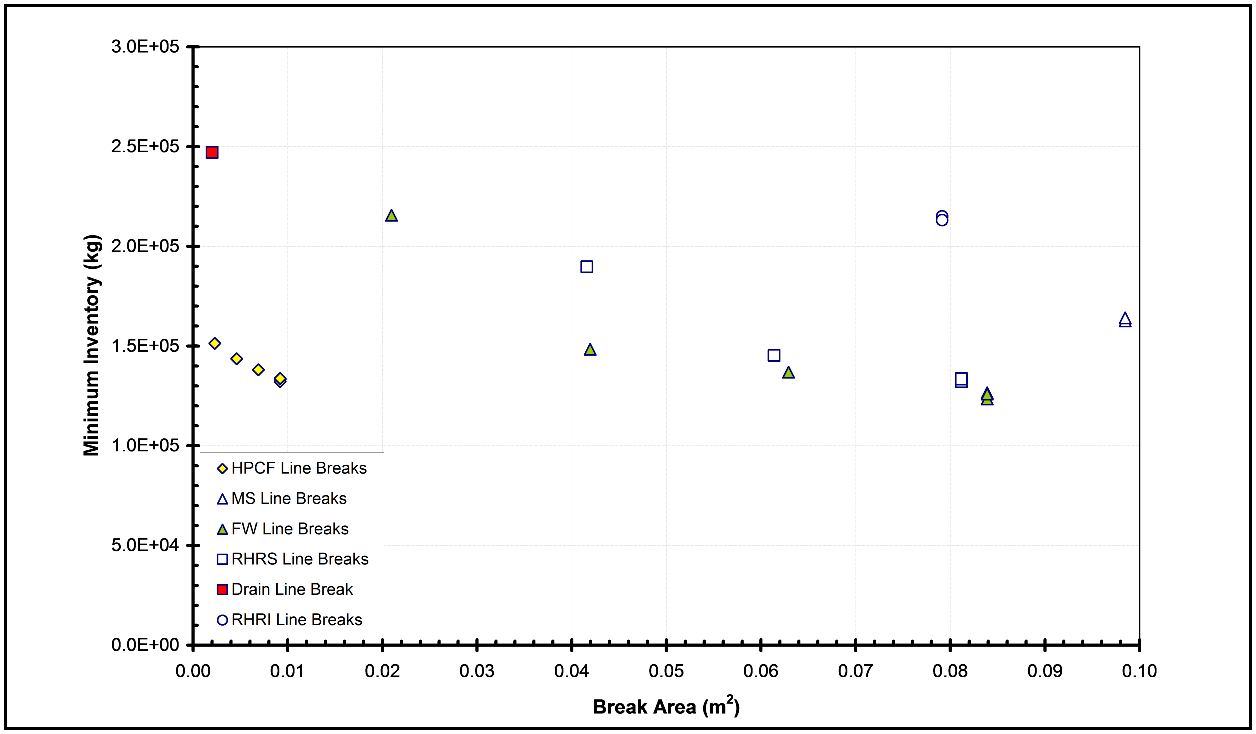
The following figure is located in Chapter 21:

Figure 6.3-7 High Pressure Core Flooder System P&ID (Sheets 1-2)

The following figures have been deleted:

Figure 6.3-8 Not Used (See Figure 5.4-8)

Figure 6.3-9 Not Used (See Figure 5.4-10)



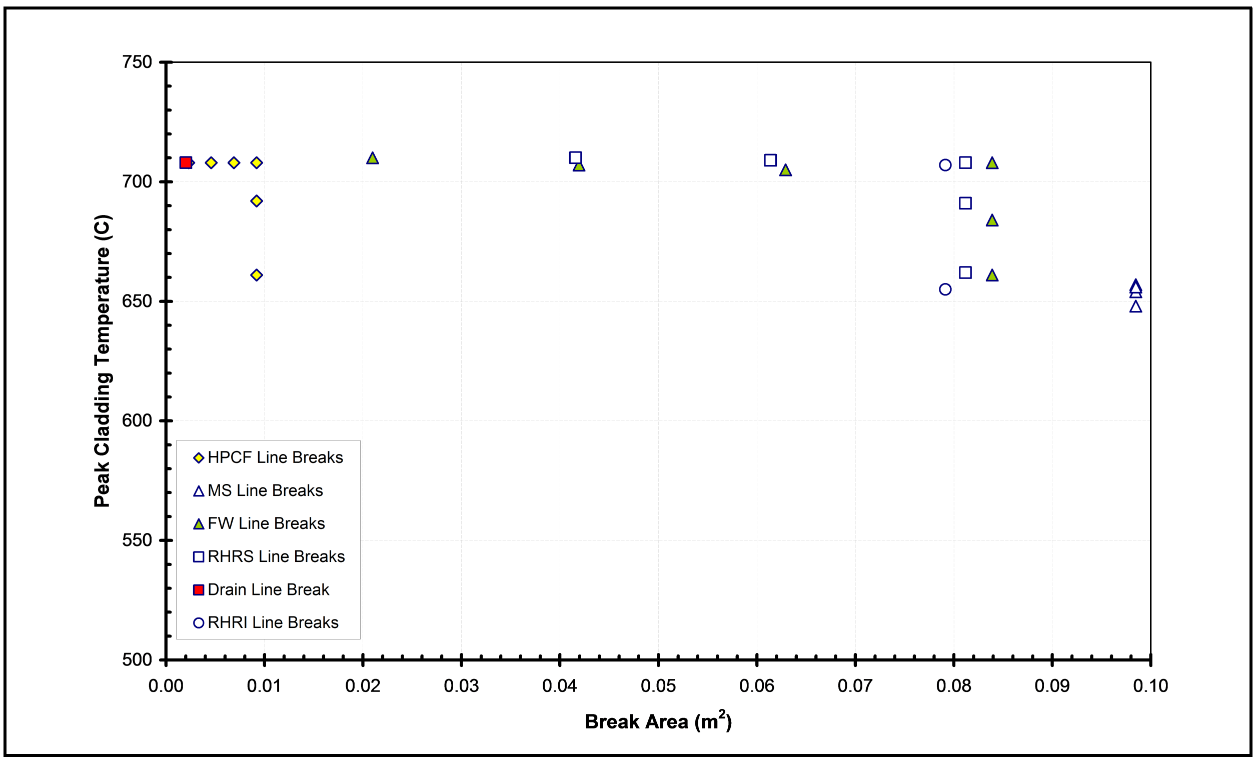


Figure 6.3-10 Summary of Peak Cladding Temperature and Minimum Inventory Results vs. Break Size

Figure 6.3-11 Not Used

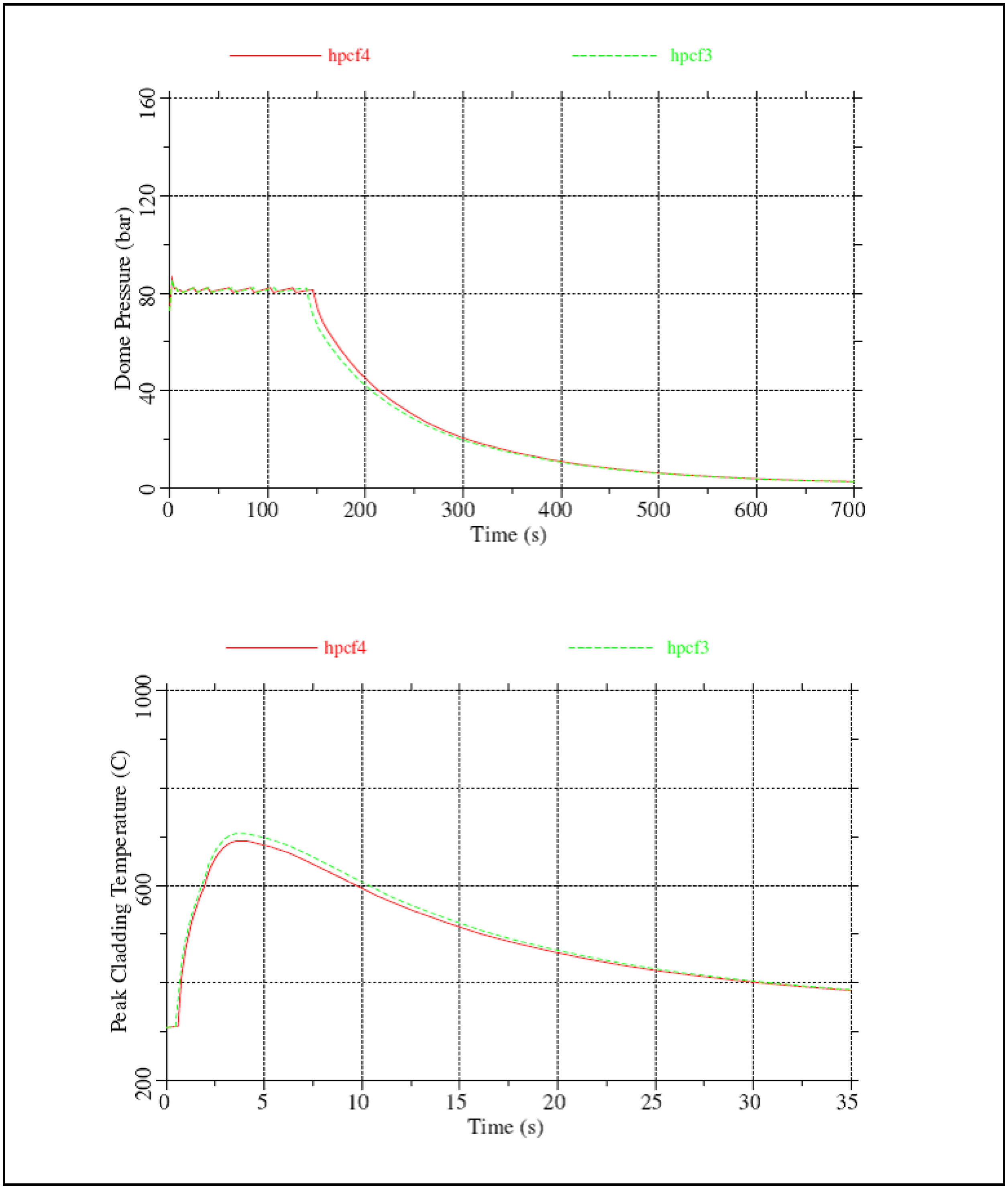


Figure 6.3-12 HPCF Core Flow Rate Sensitivity - Dome Pressure and GOBLIN PCT

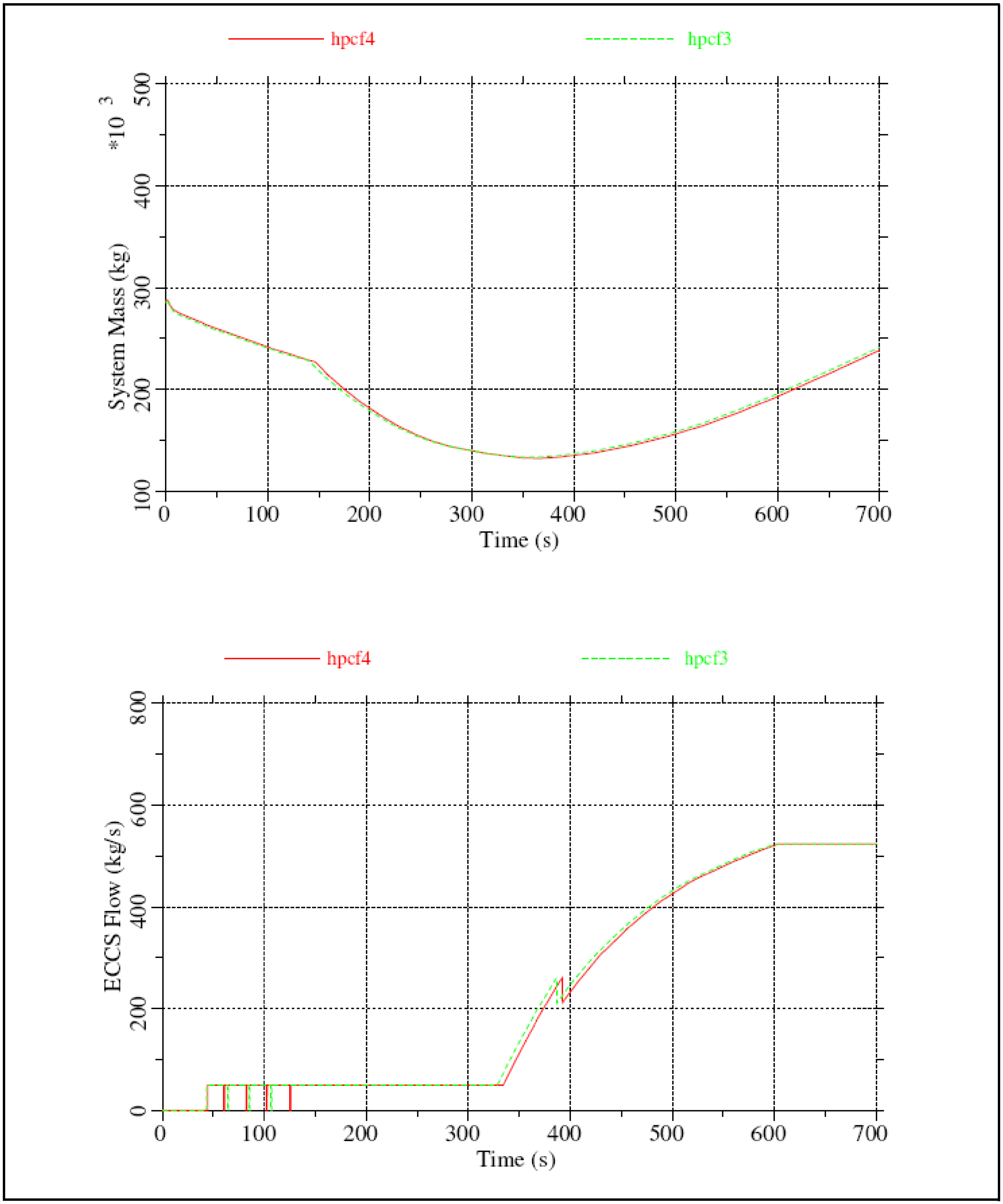


Figure 6.3-13 HPCF Core Flow Rate Sensitivity - System Mass and ECCS Flow Rate

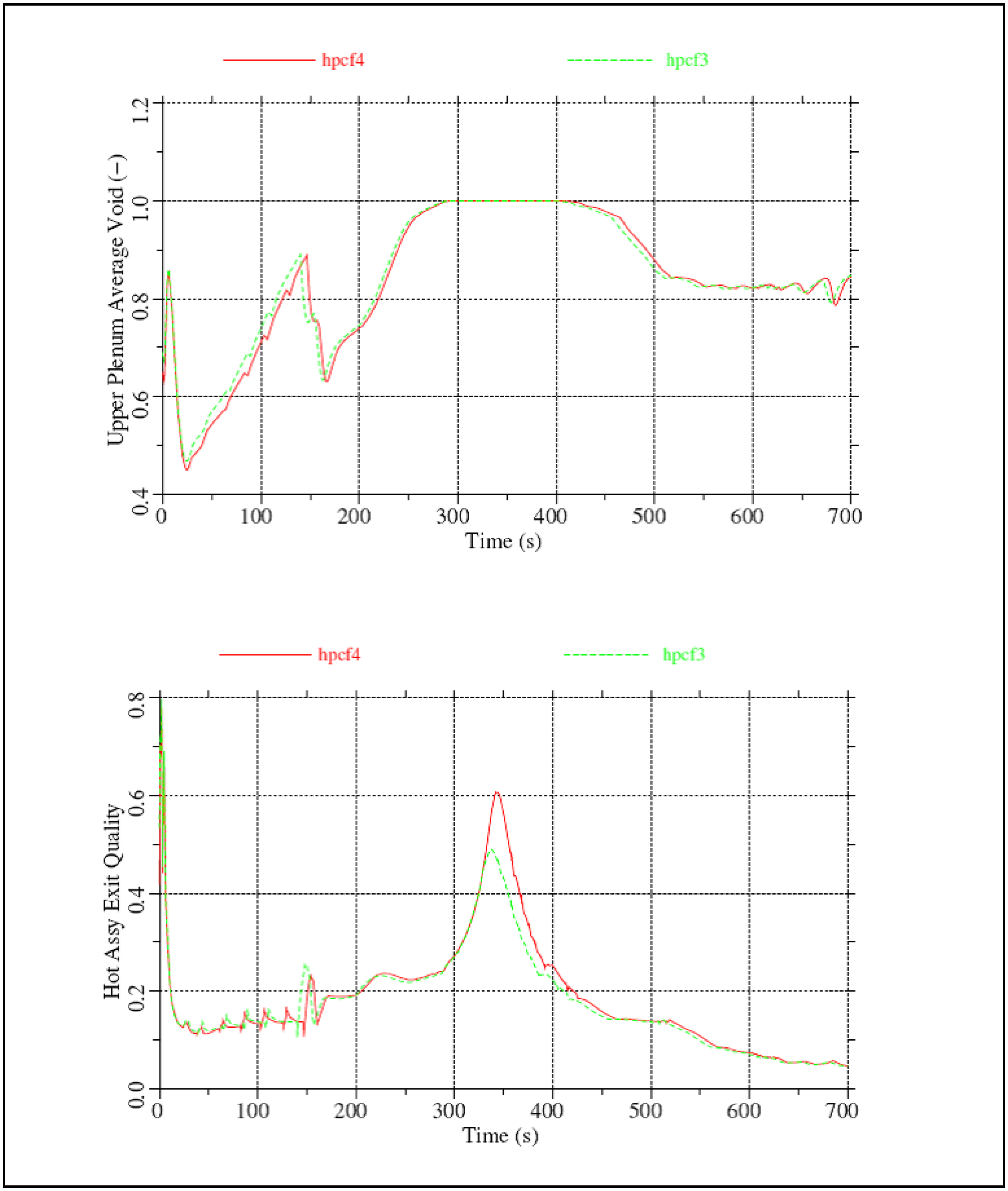


Figure 6.3-14 HPCF Core Flow Rate Sensitivity - Upper Plenum Void and Hot Assembly Exit Quality

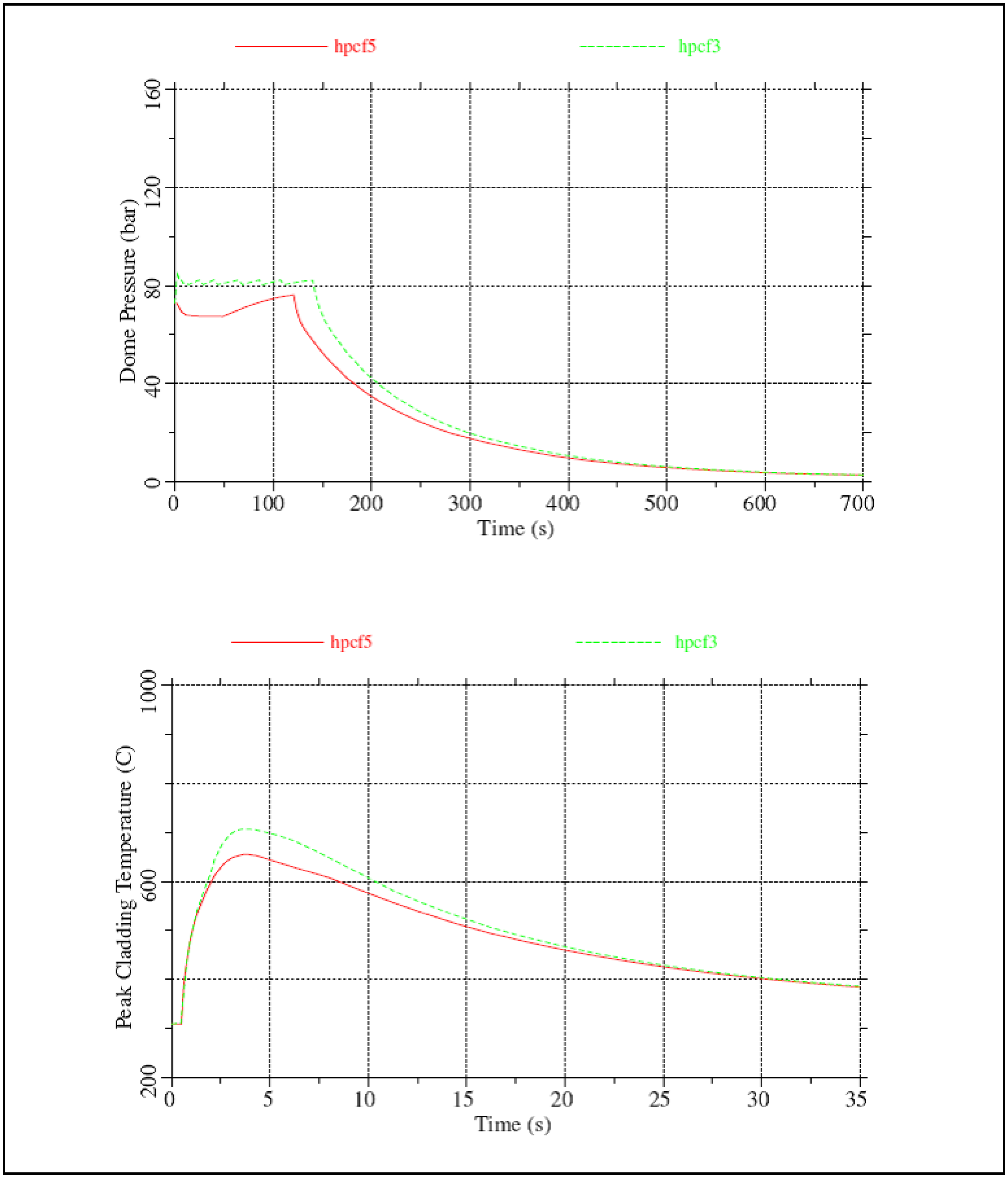


Figure 6.3-15 HPCF Steam Line Isolation Sensitivity -Dome Pressure and GOBLIN PCT

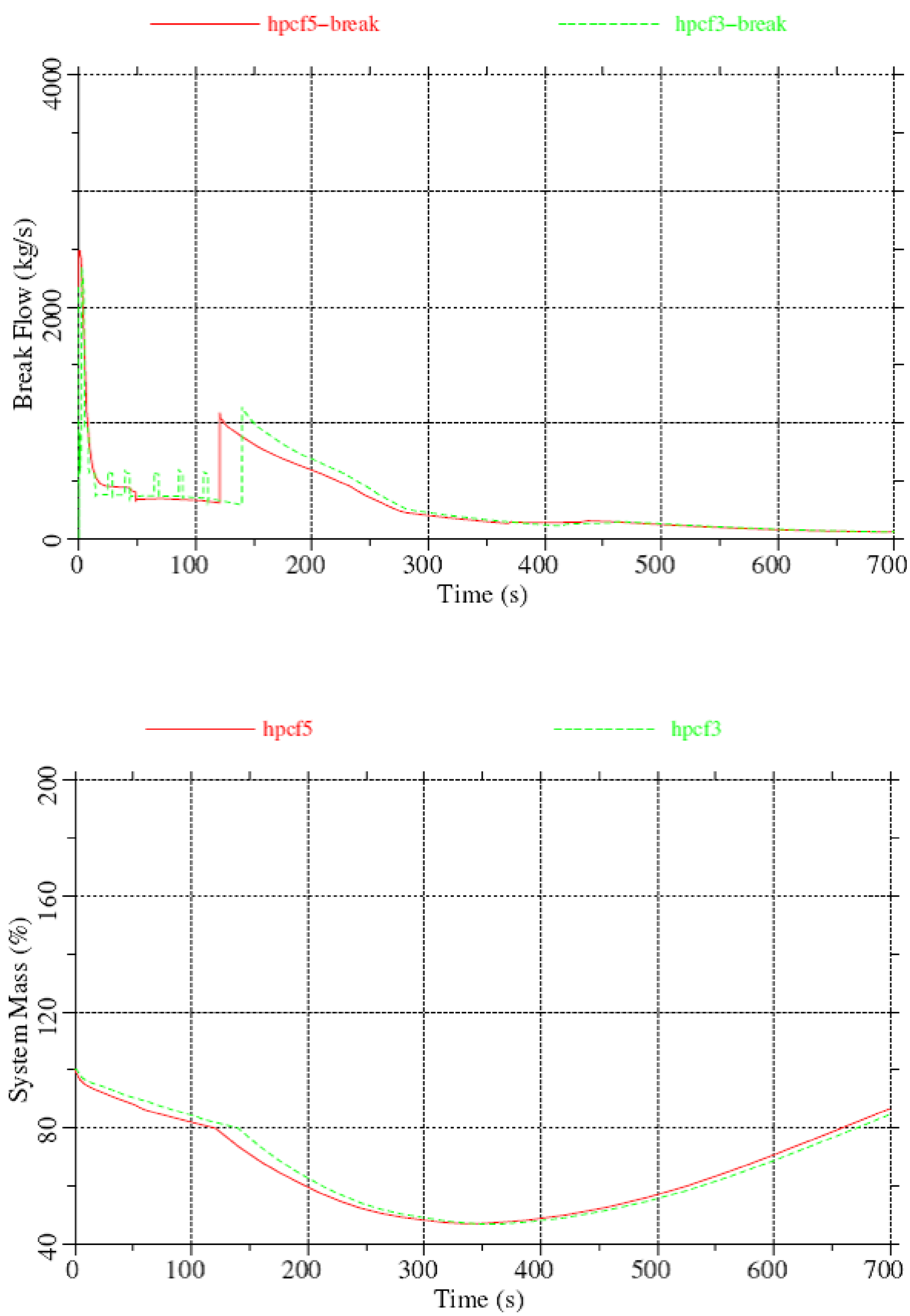


Figure 6.3-16 HPCF Steam Line Isolation Sensitivity - Break Flow Rate and System Mass

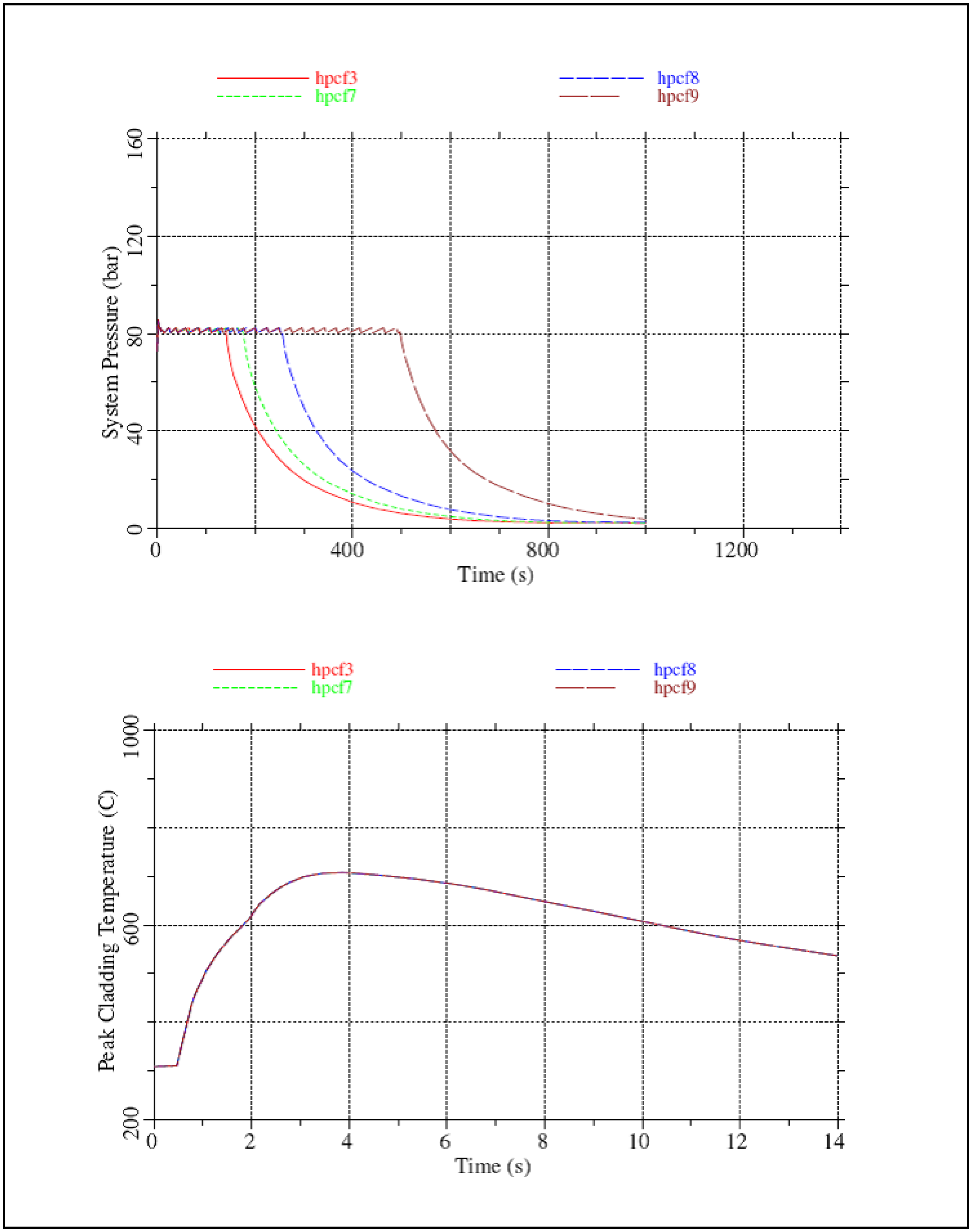


Figure 6.3-17 HPCF Break Size Sensitivity - Dome Pressure and PCT

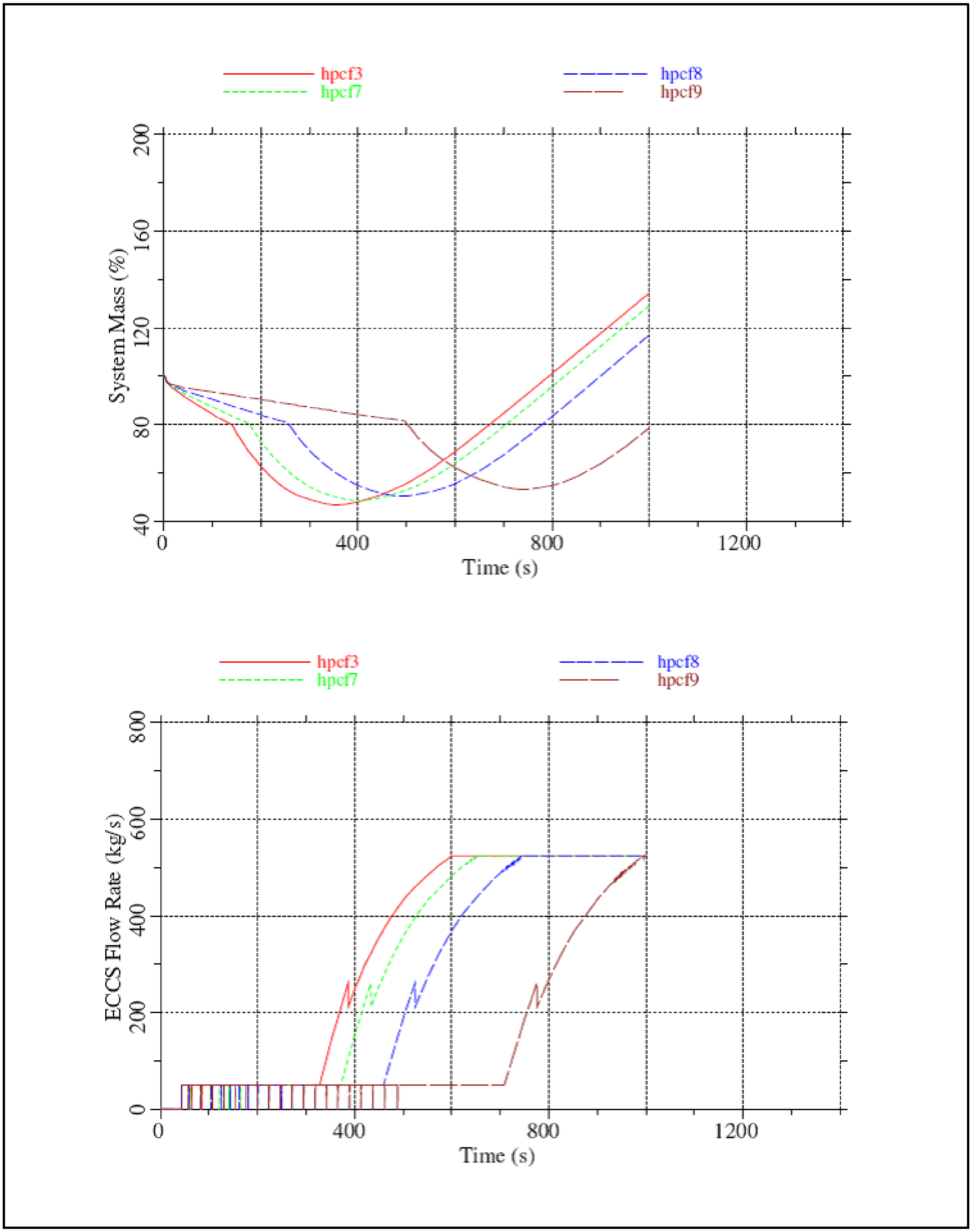


Figure 6.3-18 HPCF Break Size Sensitivity - System Mass and ECCS Flow Rate

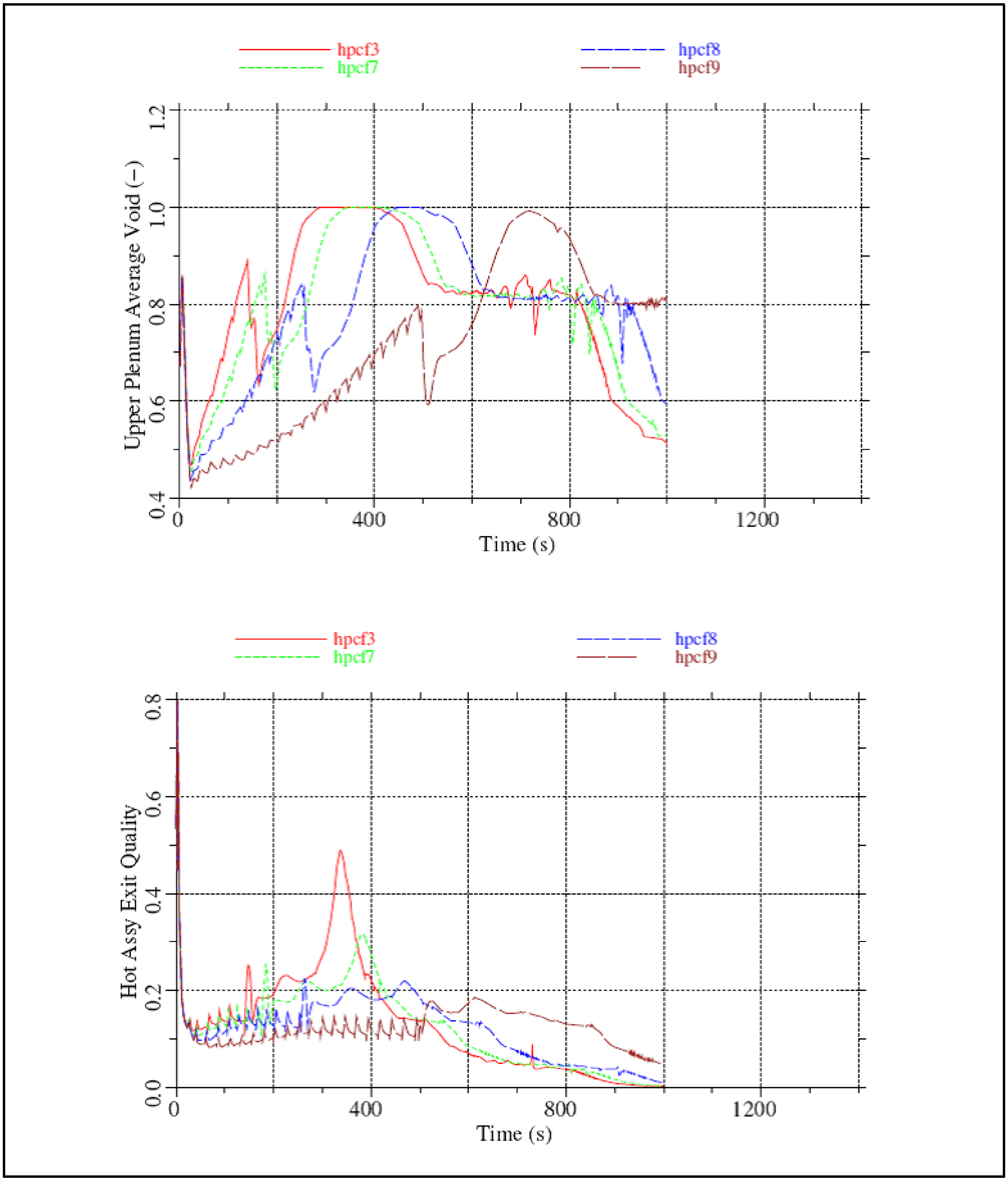


Figure 6.3-19 HPCF Break Size Sensitivity - Upper Plenum Void and Hot Assembly Exit Quality

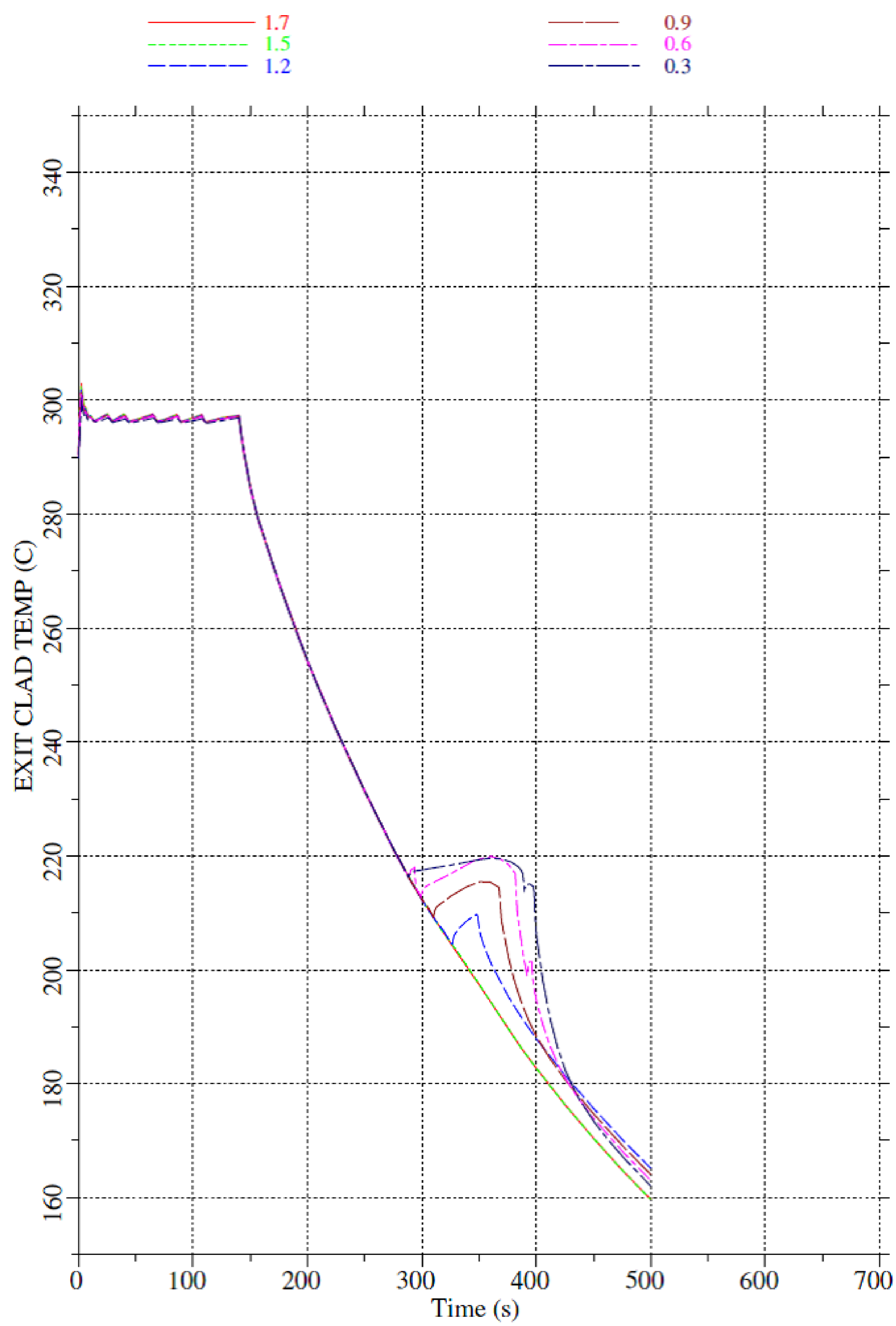


Figure 6.3-20 HPCF Assembly Power Sensitivity - Exit Cladding Temperature vs. Channel Peaking Factor

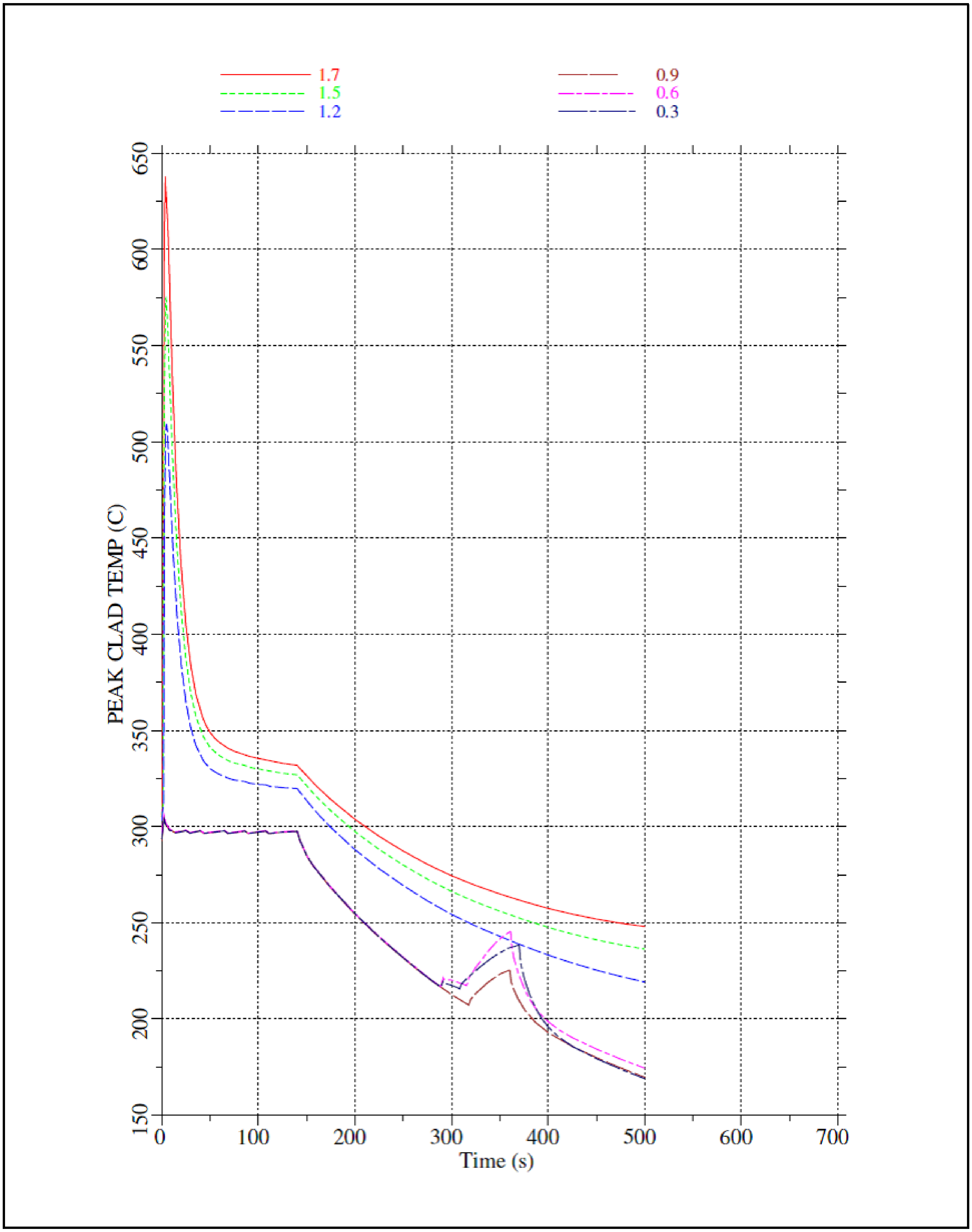


Figure 6.3-21 HPCF Assembly Power Sensitivity - GOBLIN PCT vs. Channel Peaking Factor

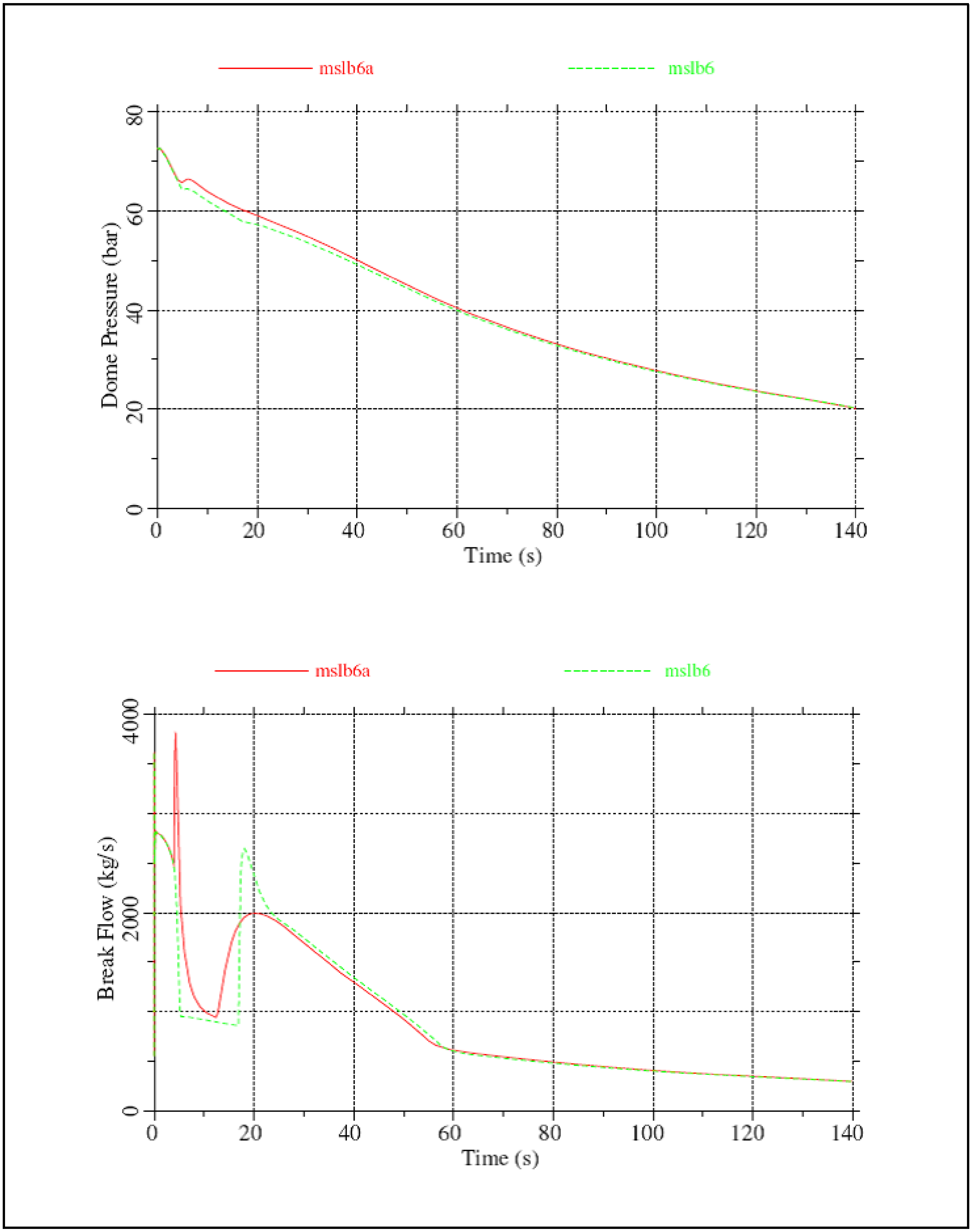


Figure 6.3-22 MSLB Core Flow Rate Sensitivity - Dome Pressure and Break Flow Rate

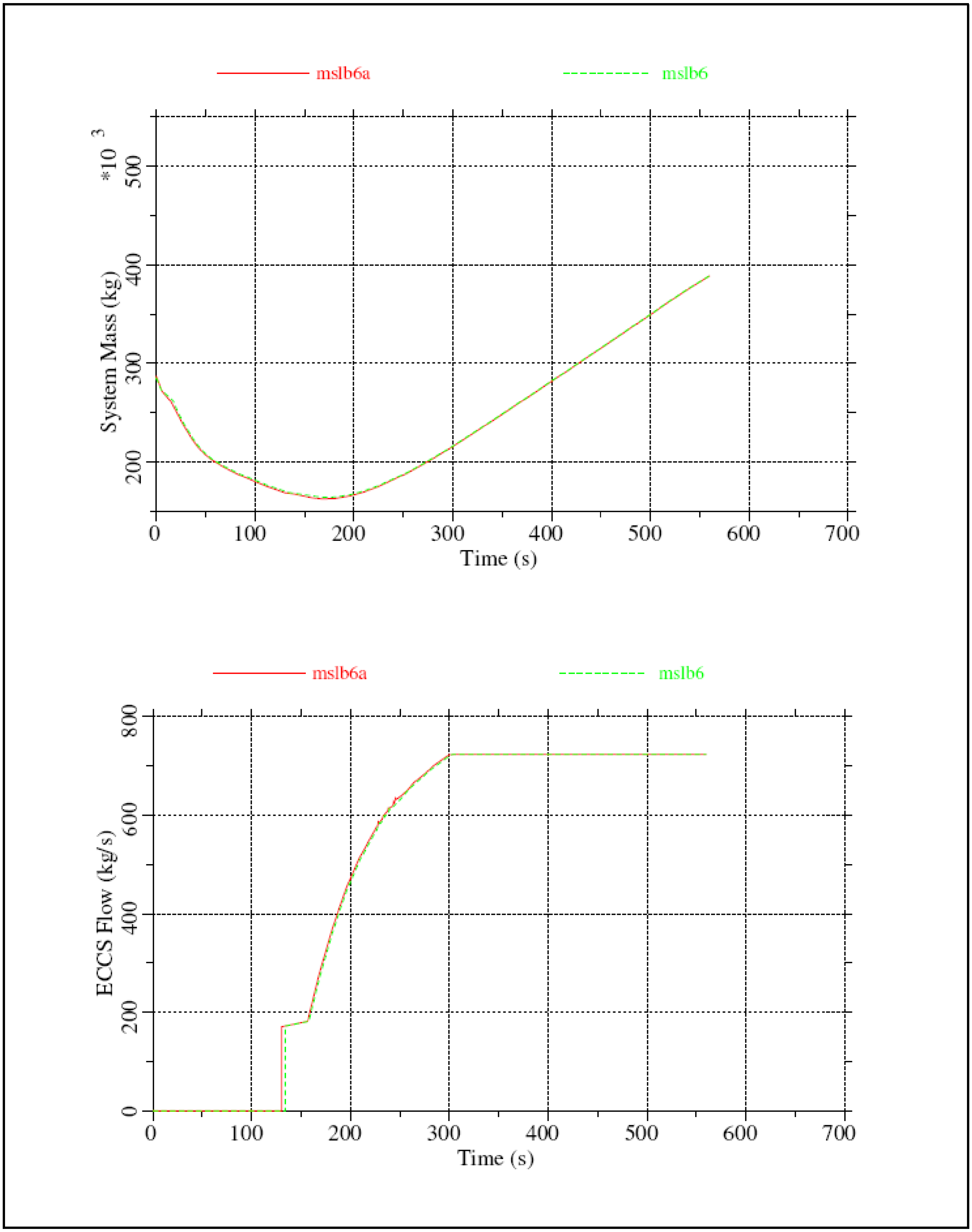


Figure 6.3-23 MSLB Core Flow Rate Sensitivity - System Mass and ECCS Flow Rate

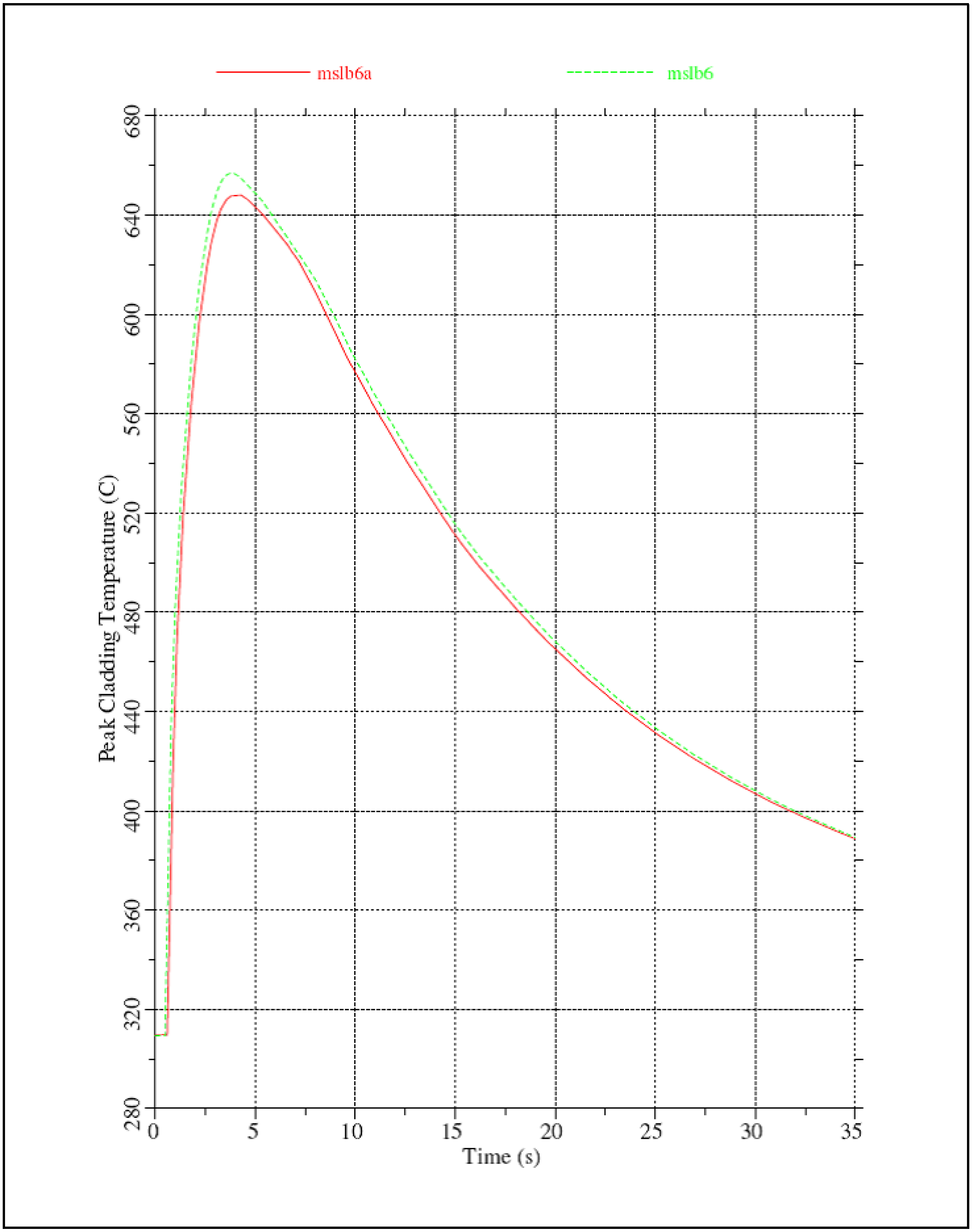


Figure 6.3-24 MSLB Core Flow Rate Sensitivity - GOBLIN PCTs

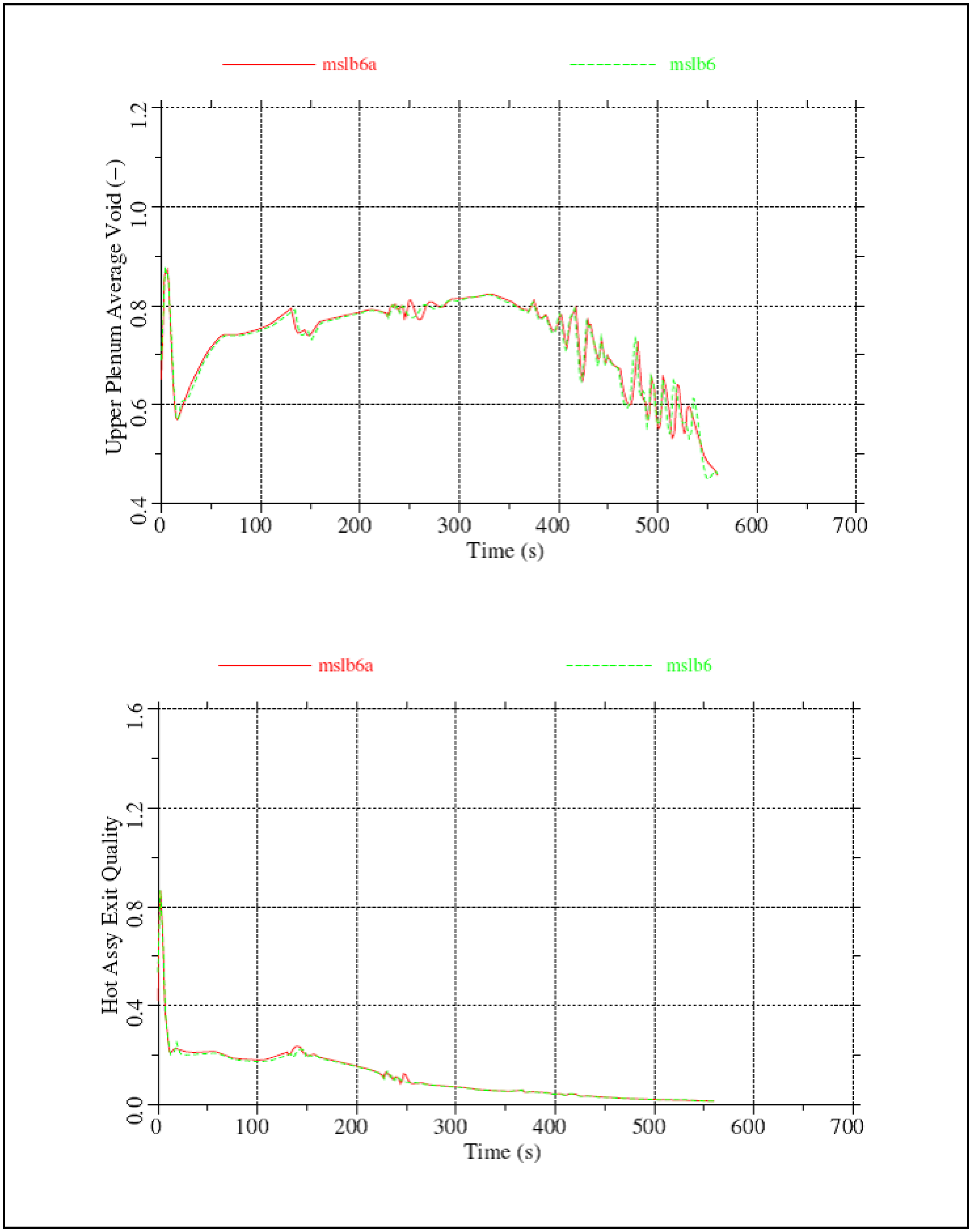


Figure 6.3-25 MSLB Core Flow Rate Sensitivity - Upper Plenum Void and Hot Assembly Exit Quality

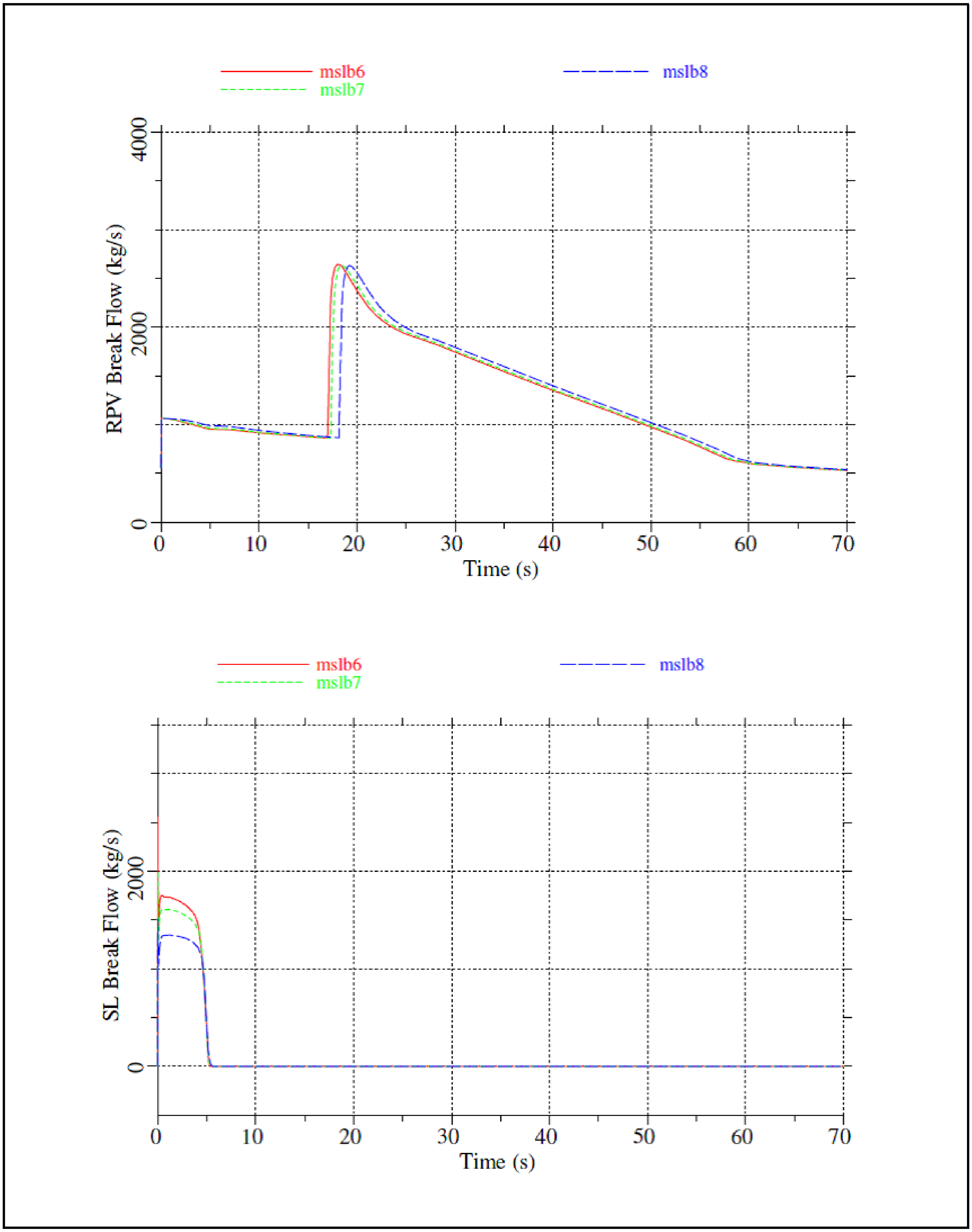


Figure 6.3-26 MSLB Break Size Sensitivity - Break Flow Rates

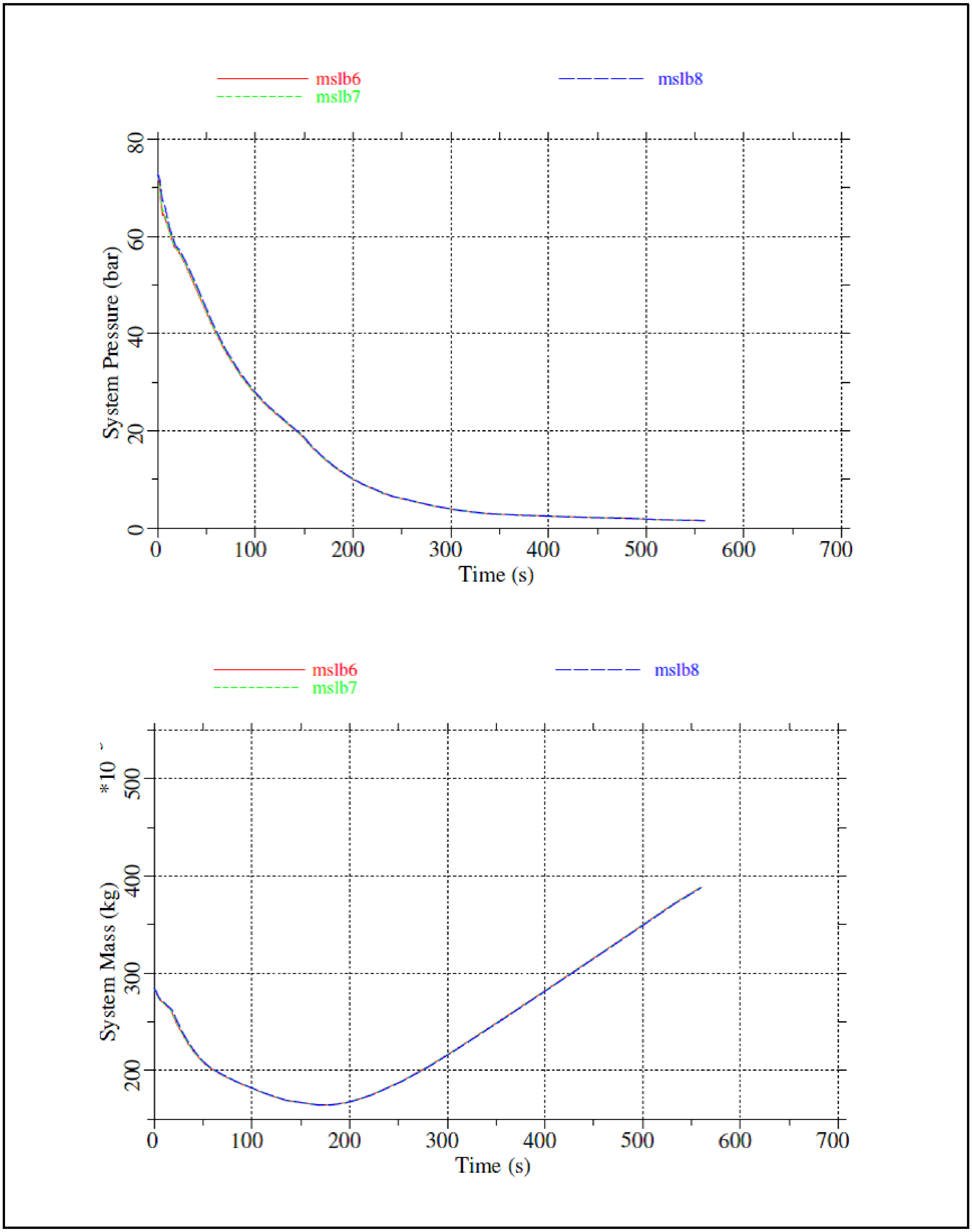


Figure 6.3-27 MSLB Break Size Sensitivity - Dome Pressure and System Mass

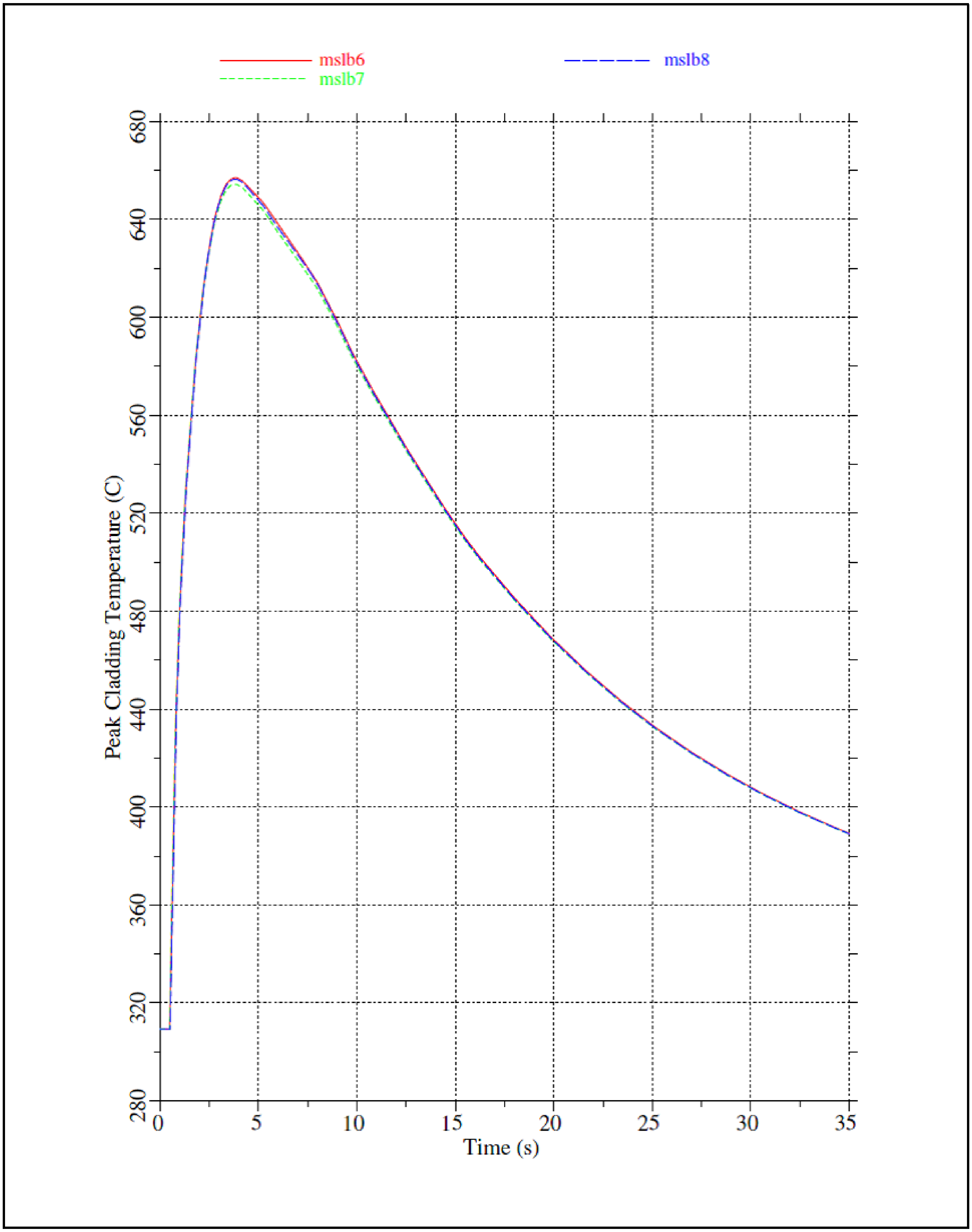


Figure 6.3-28 MSLB Break Size Sensitivity - GOBLIN PCTs

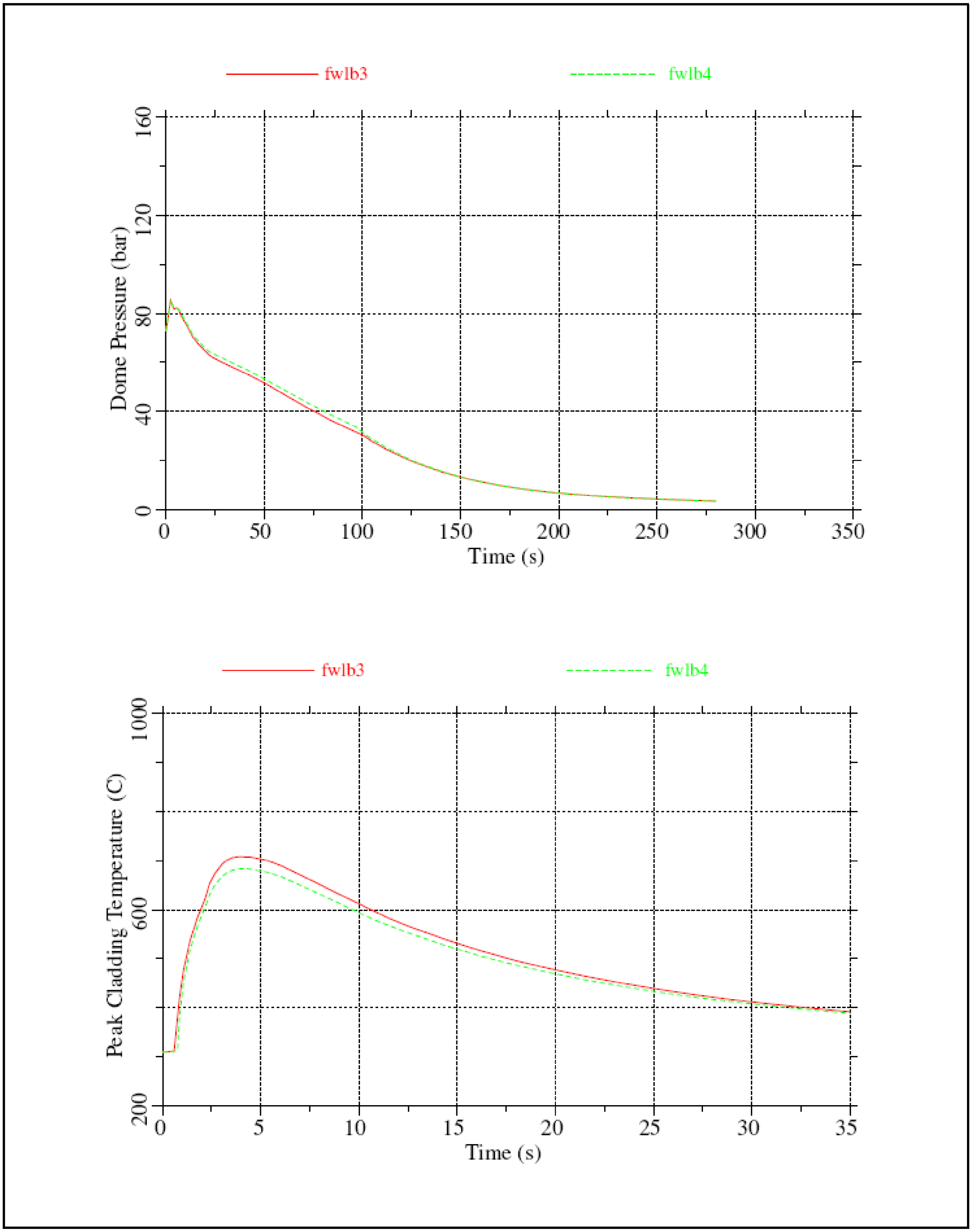


Figure 6.3-29 FWLB Core Flow Rate Sensitivity - Dome Pressure and GOBLIN PCTs

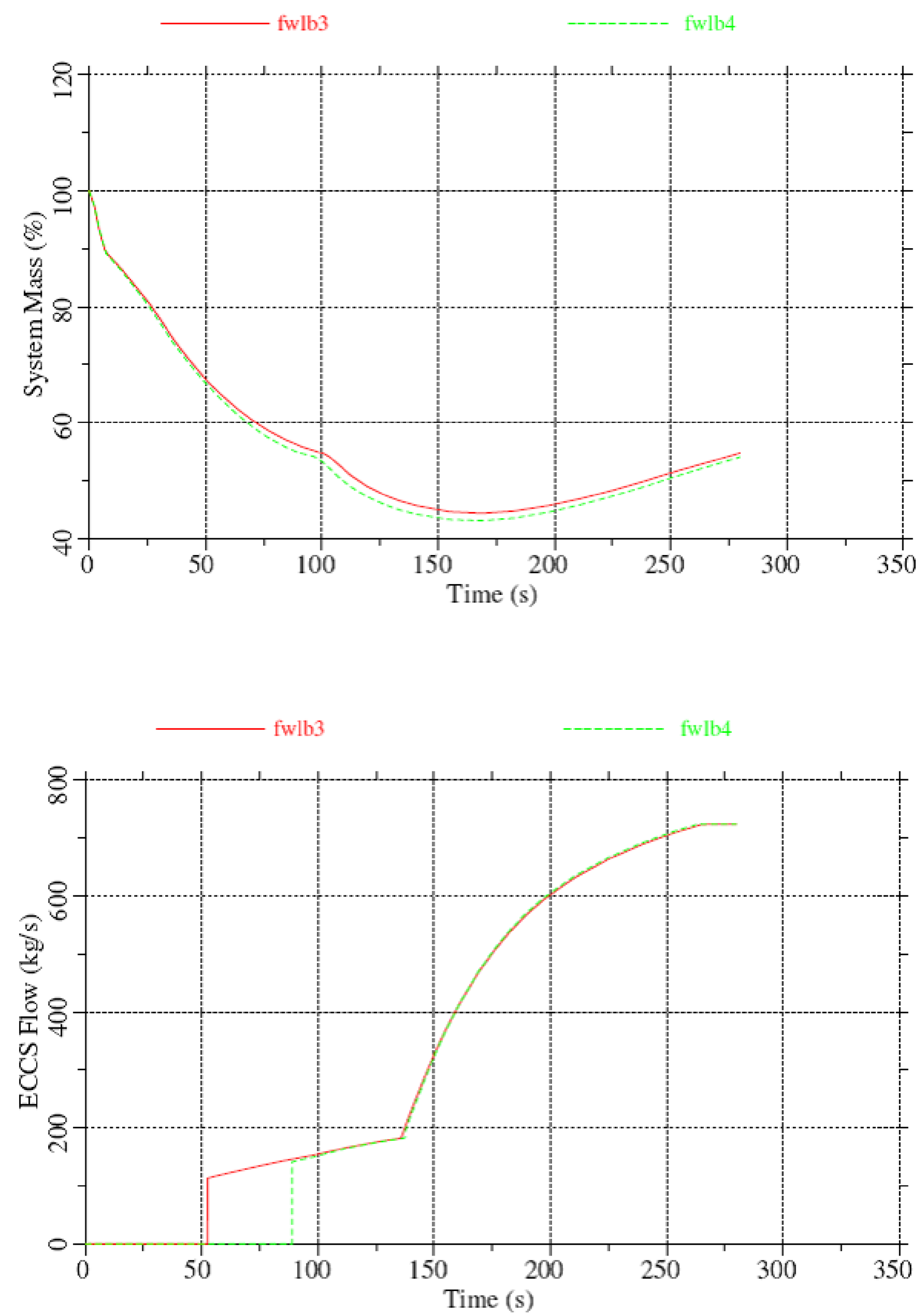


Figure 6.3-30 FWLB Core Flow Rate Sensitivity - System Mass and ECCS Flow Rates

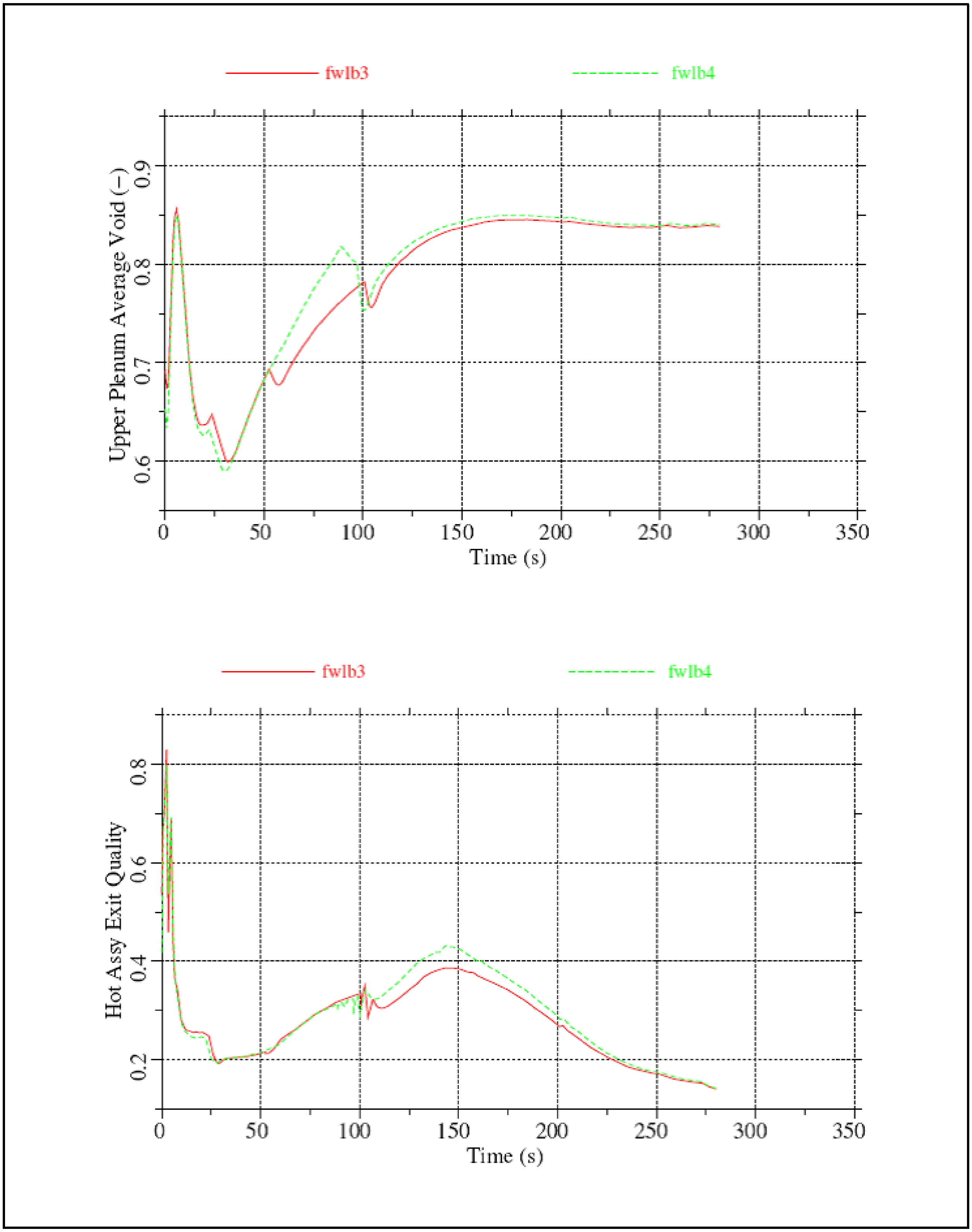


Figure 6.3-31 FWLB Core Flow Rate Sensitivity - Upper Plenum Void and Hot Assembly Exit Quality

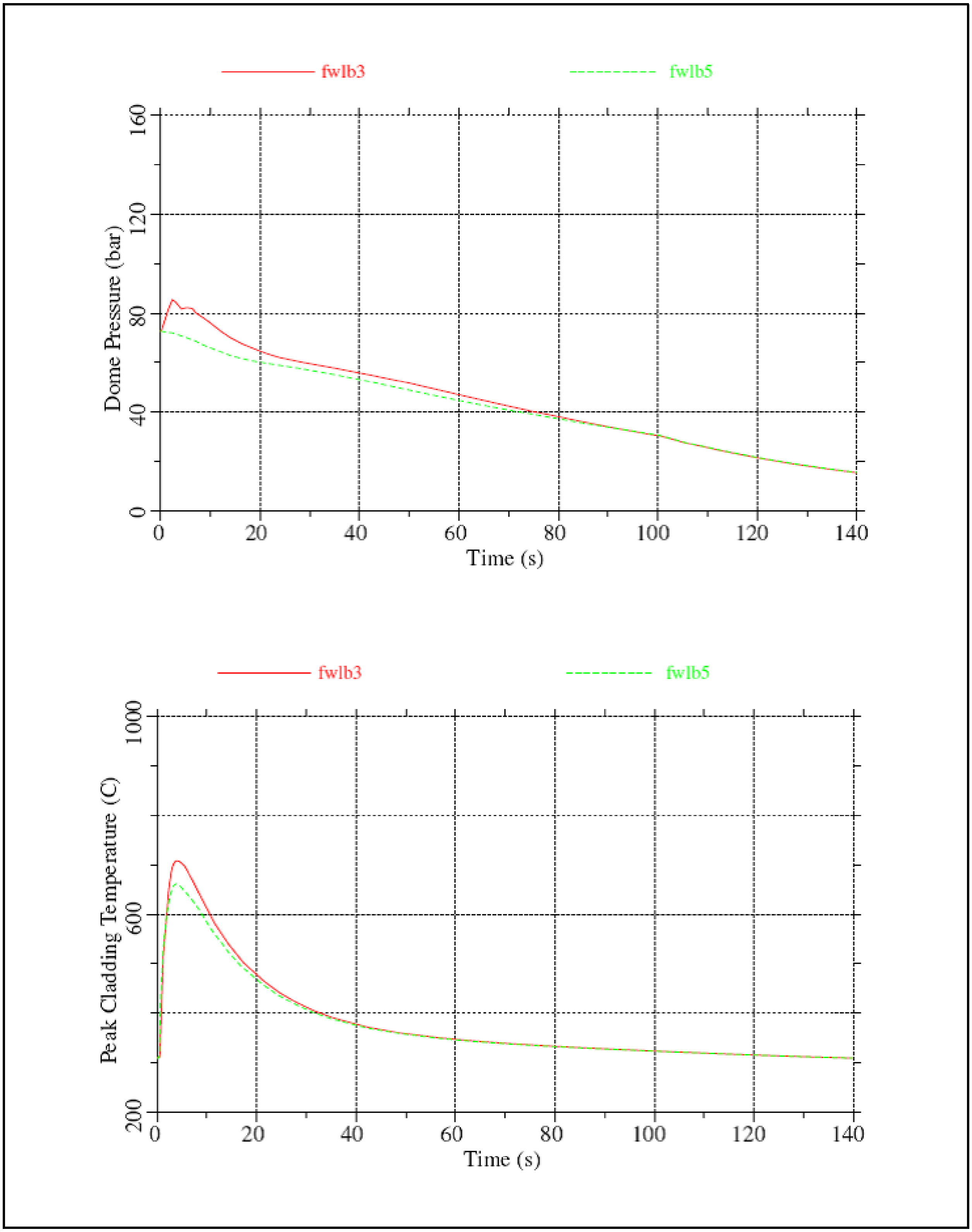
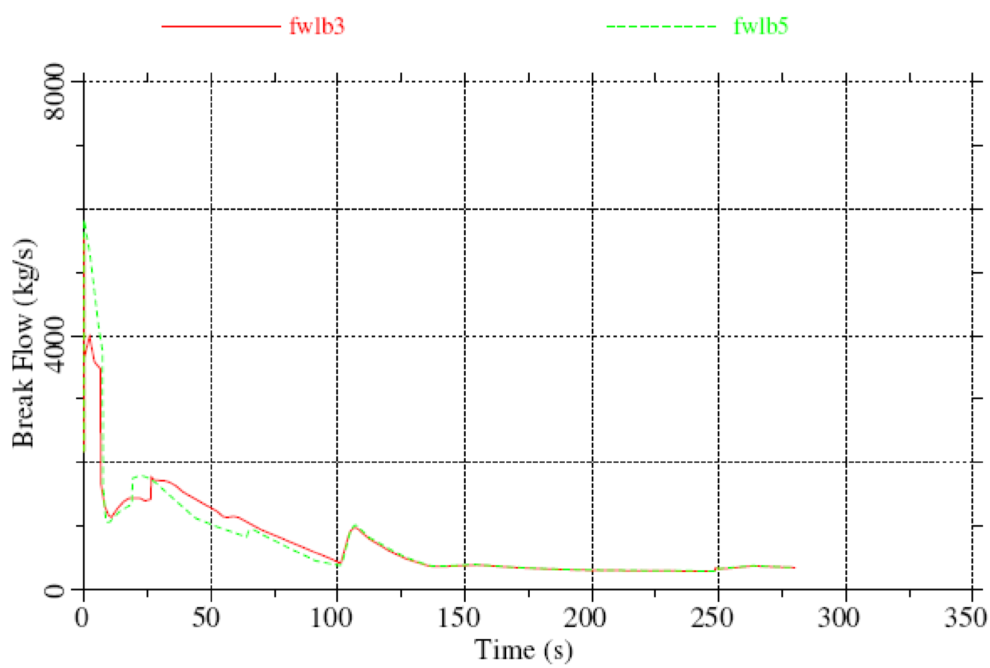


Figure 6.3-32 FWLB Steam Line Isolation Sensitivity - Dome Pressure and GOBLIN PCTs



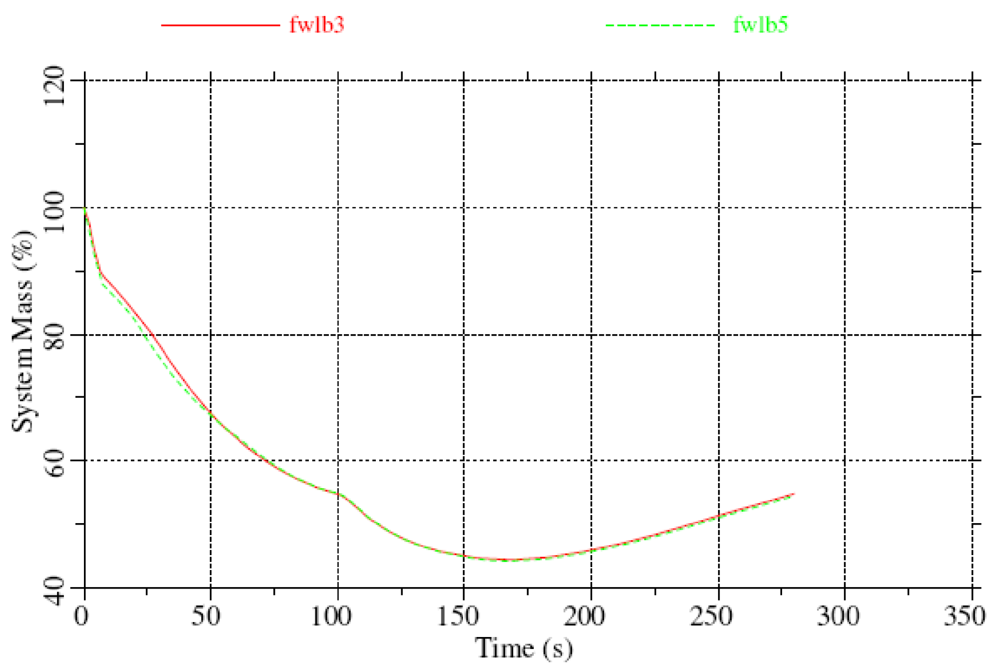


Figure 6.3-33 FWLB Steam Line Isolation Sensitivity - Break Flow Rate and System Mass

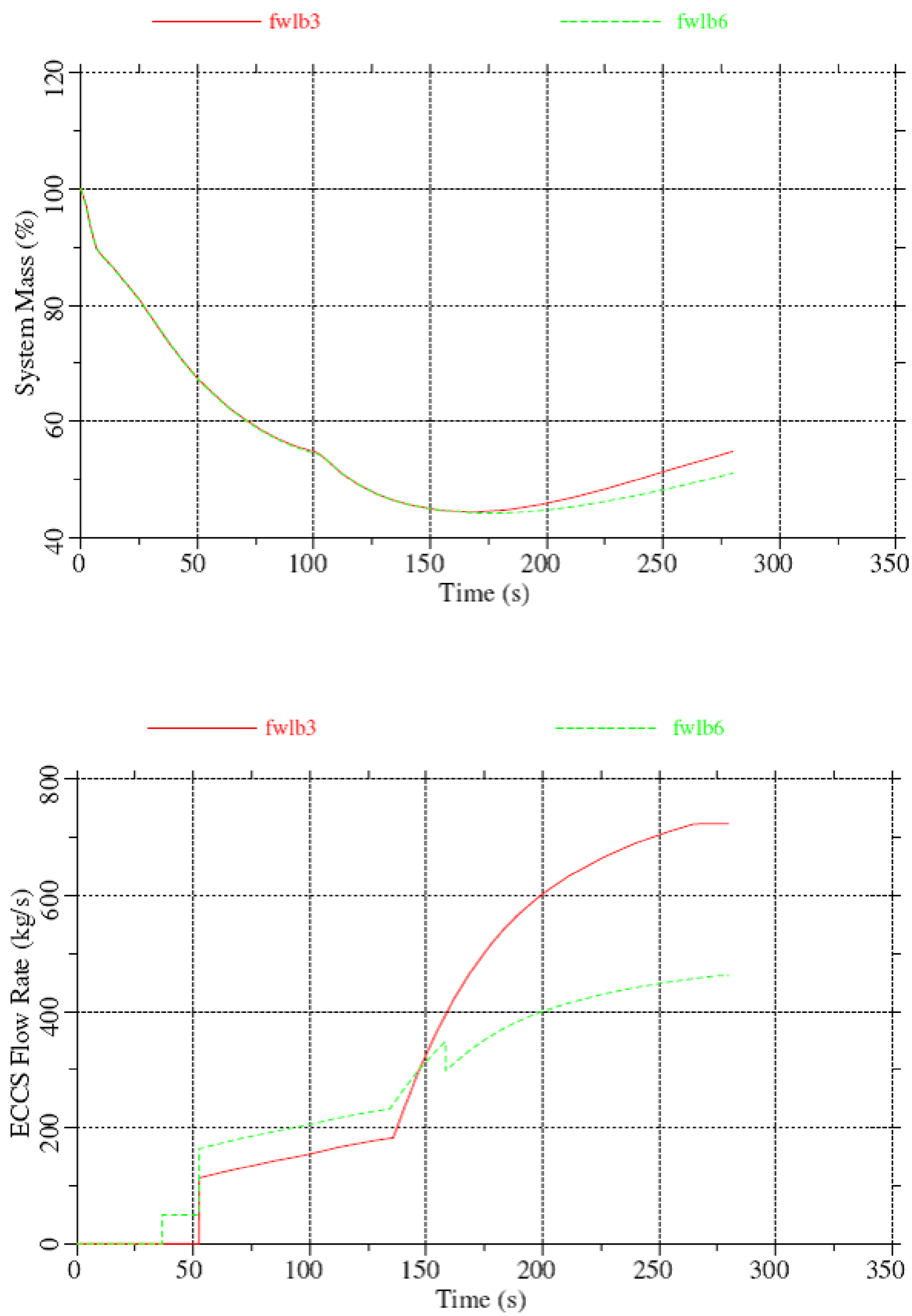


Figure 6.3-34 FWLB Break Location Sensitivity - System Mass and ECCS Flow Rates

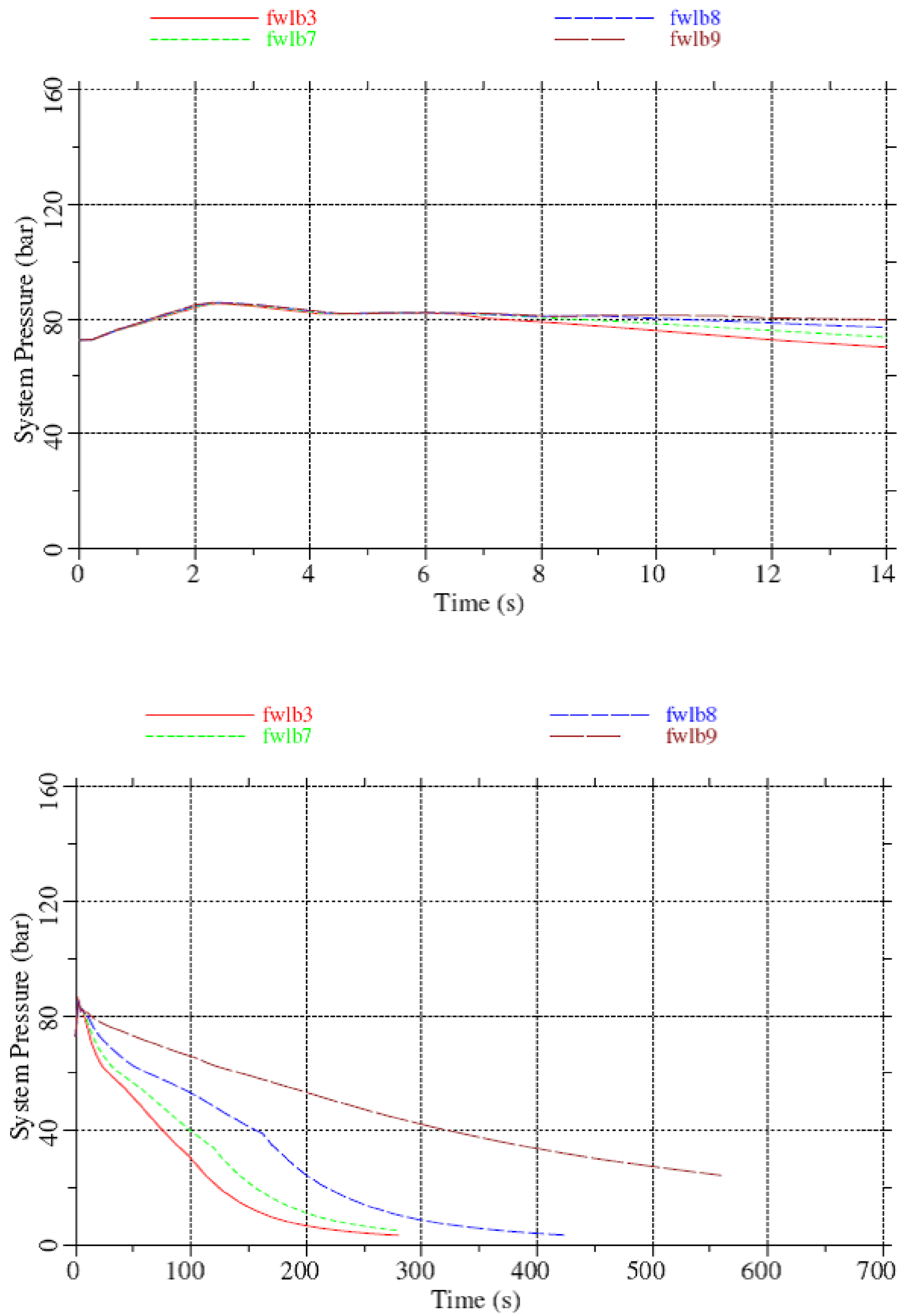


Figure 6.3-35 FWLB Break Size Sensitivity - Dome Pressure (Short-term and Long-term)

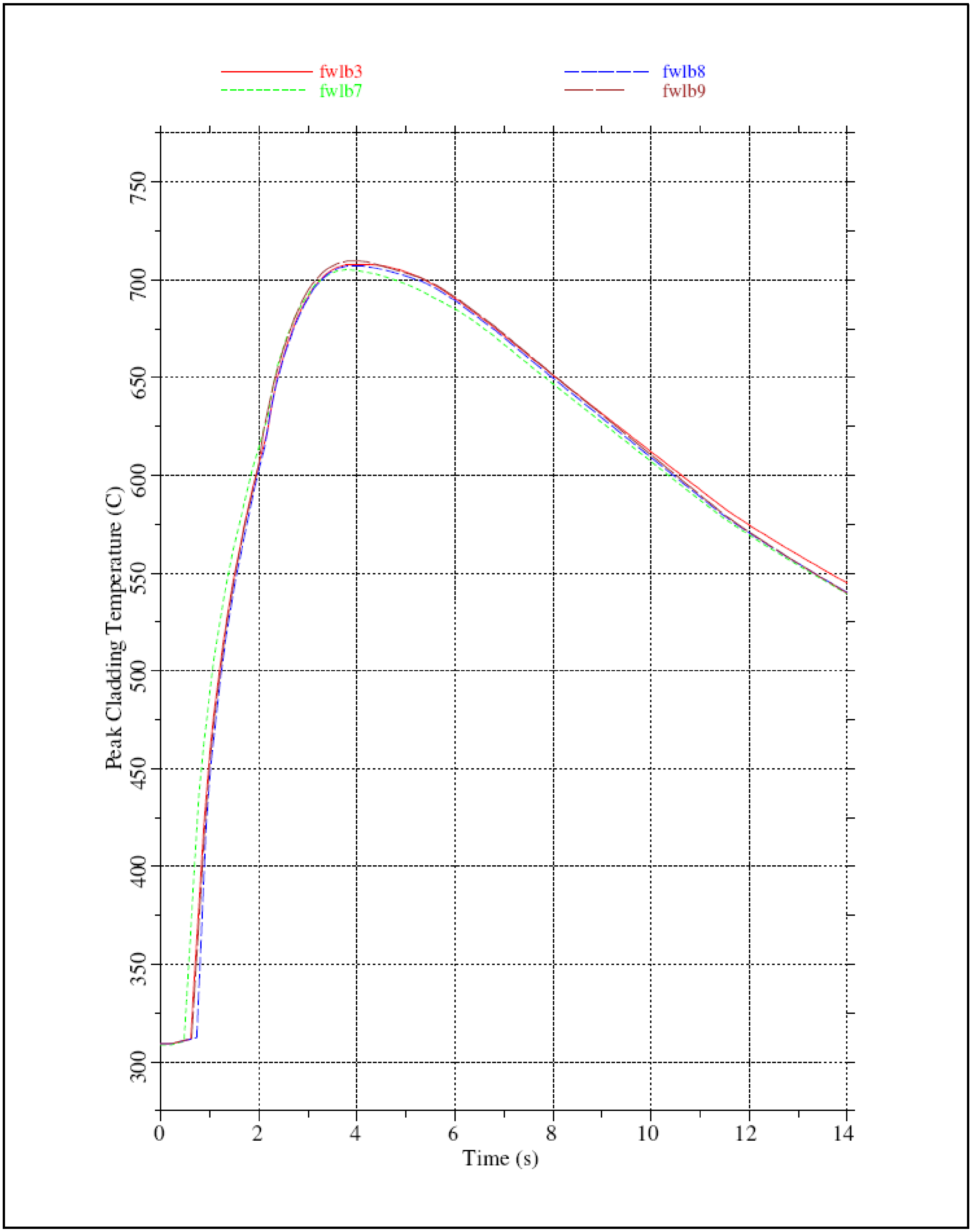


Figure 6.3-36 FWLB Break Size Sensitivity - GOBLIN PCTs

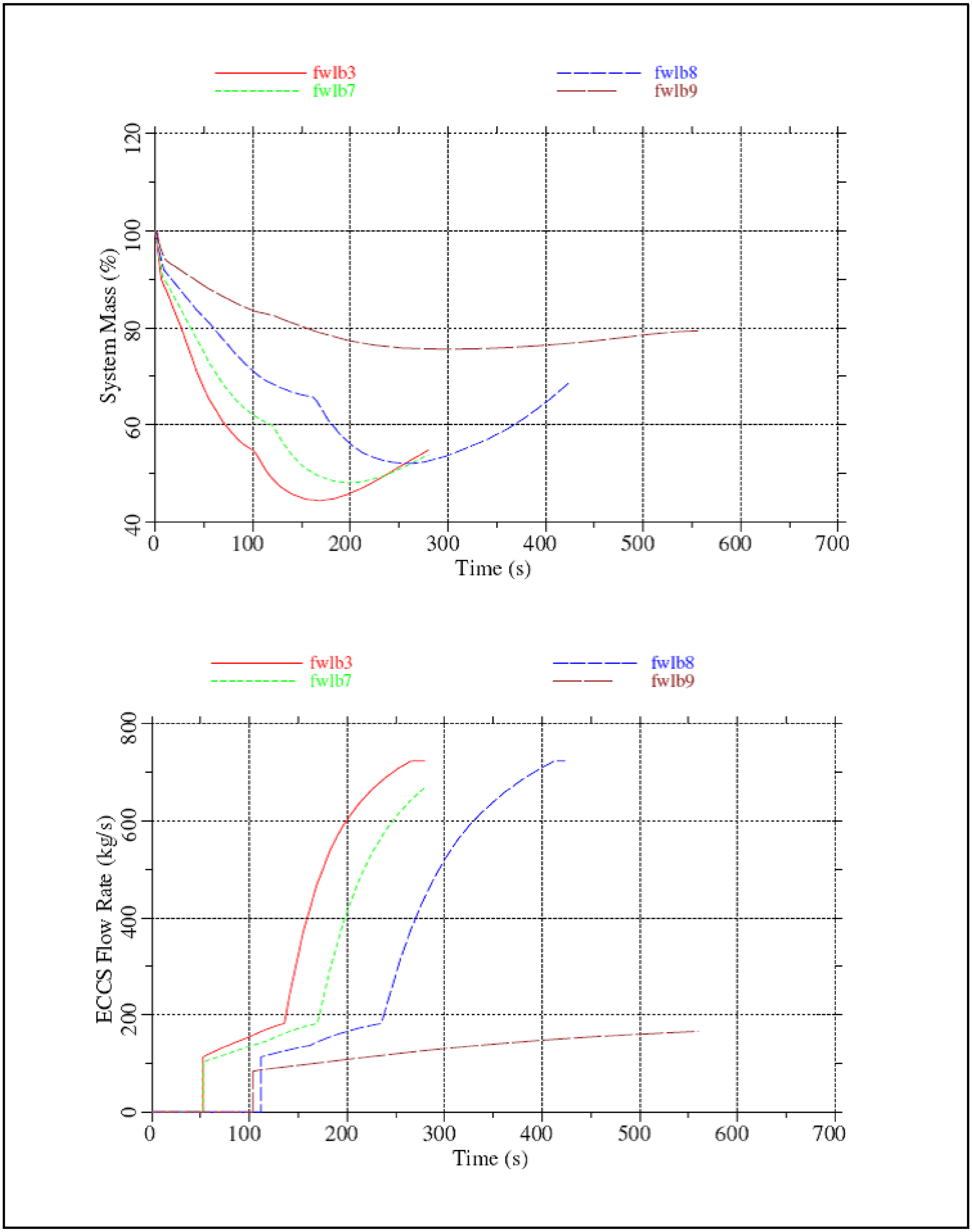


Figure 6.3-37 FWLB Break Size Sensitivity - System Mass and ECCS Flow Rates

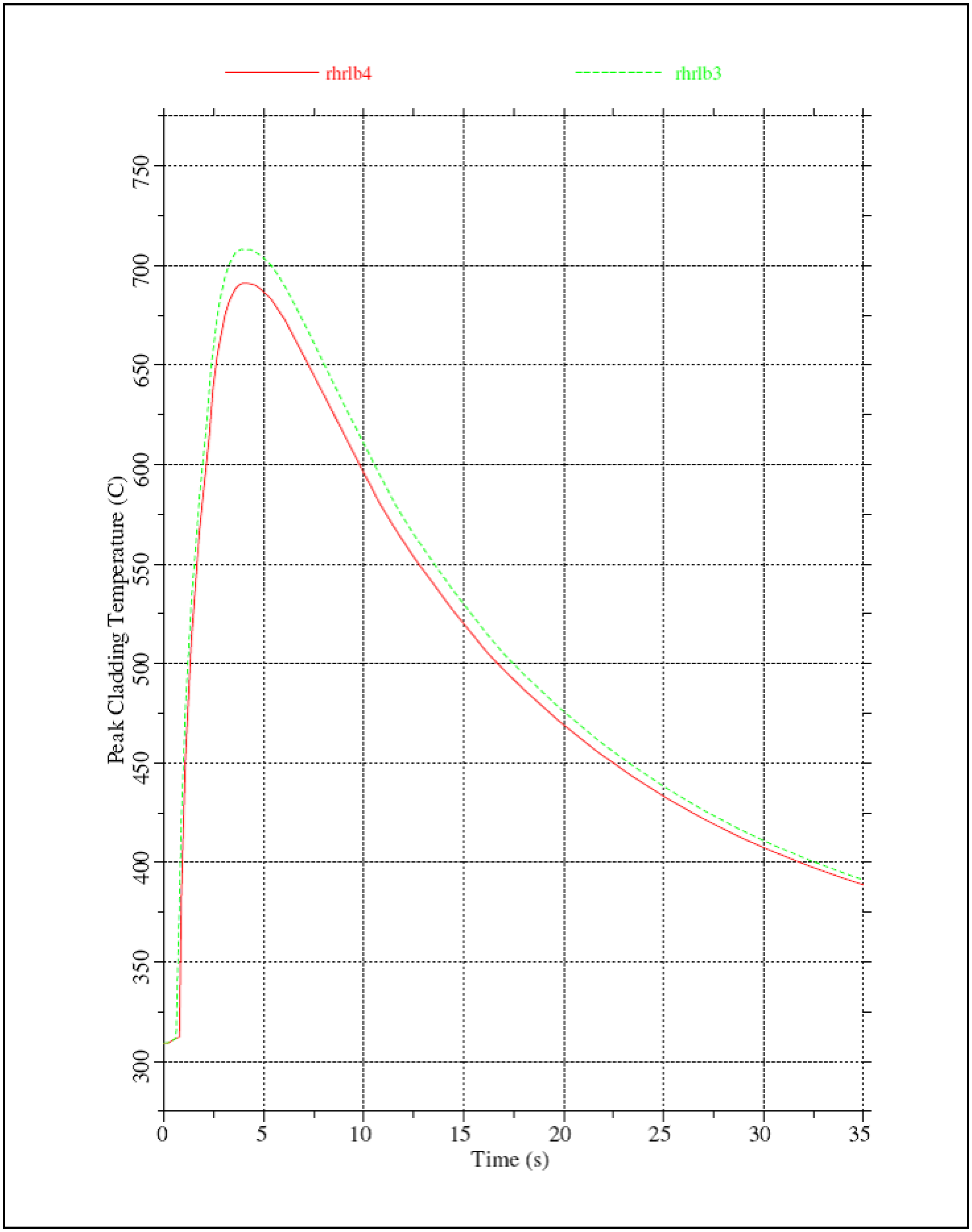


Figure 6.3-38 RHRLB Core Flow Rate Sensitivity - GOBLIN PCTs

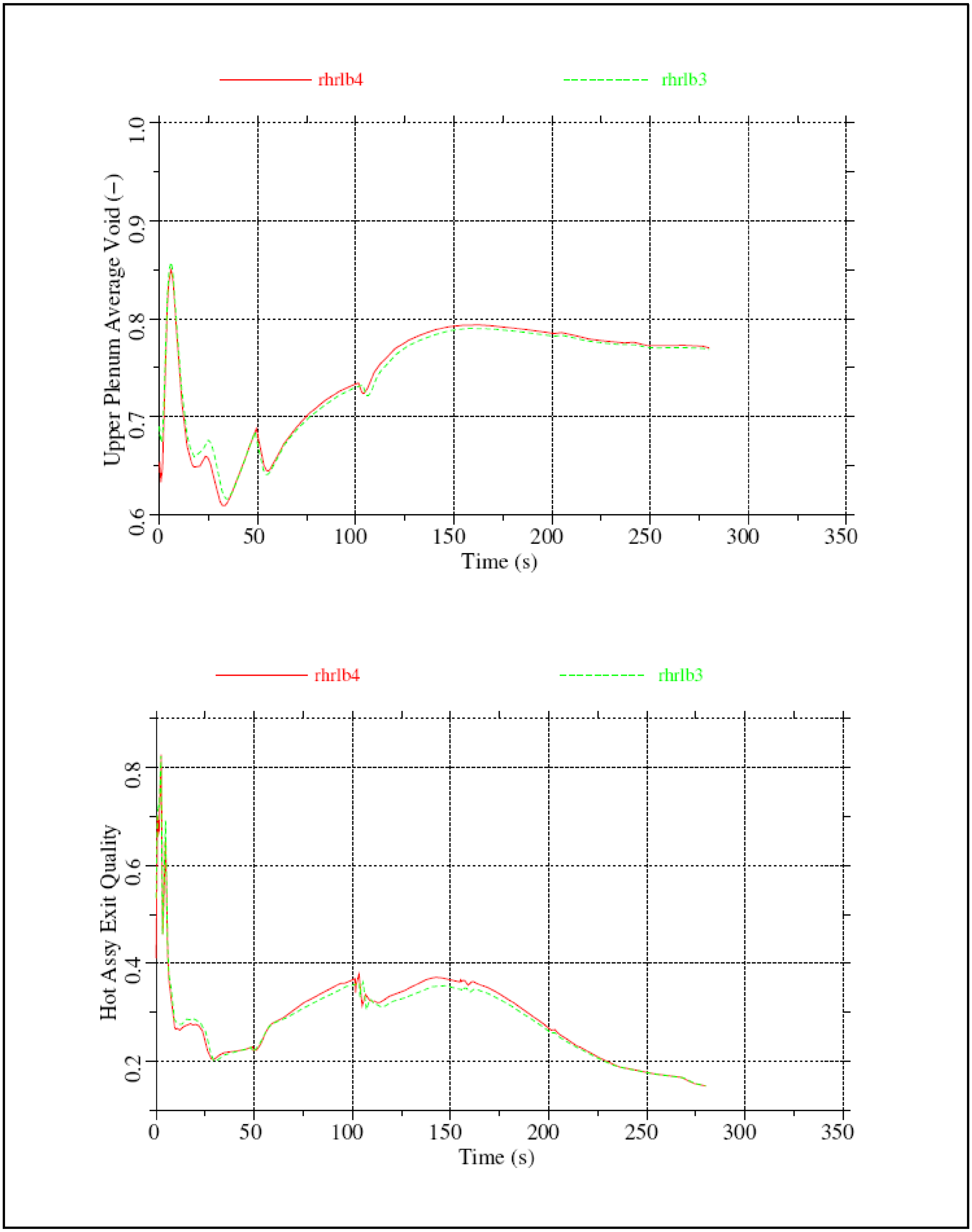


Figure 6.3-39 RHRLB Core Flow Rate Sensitivity - Upper Plenum Void and Hot Assembly Exit Quality

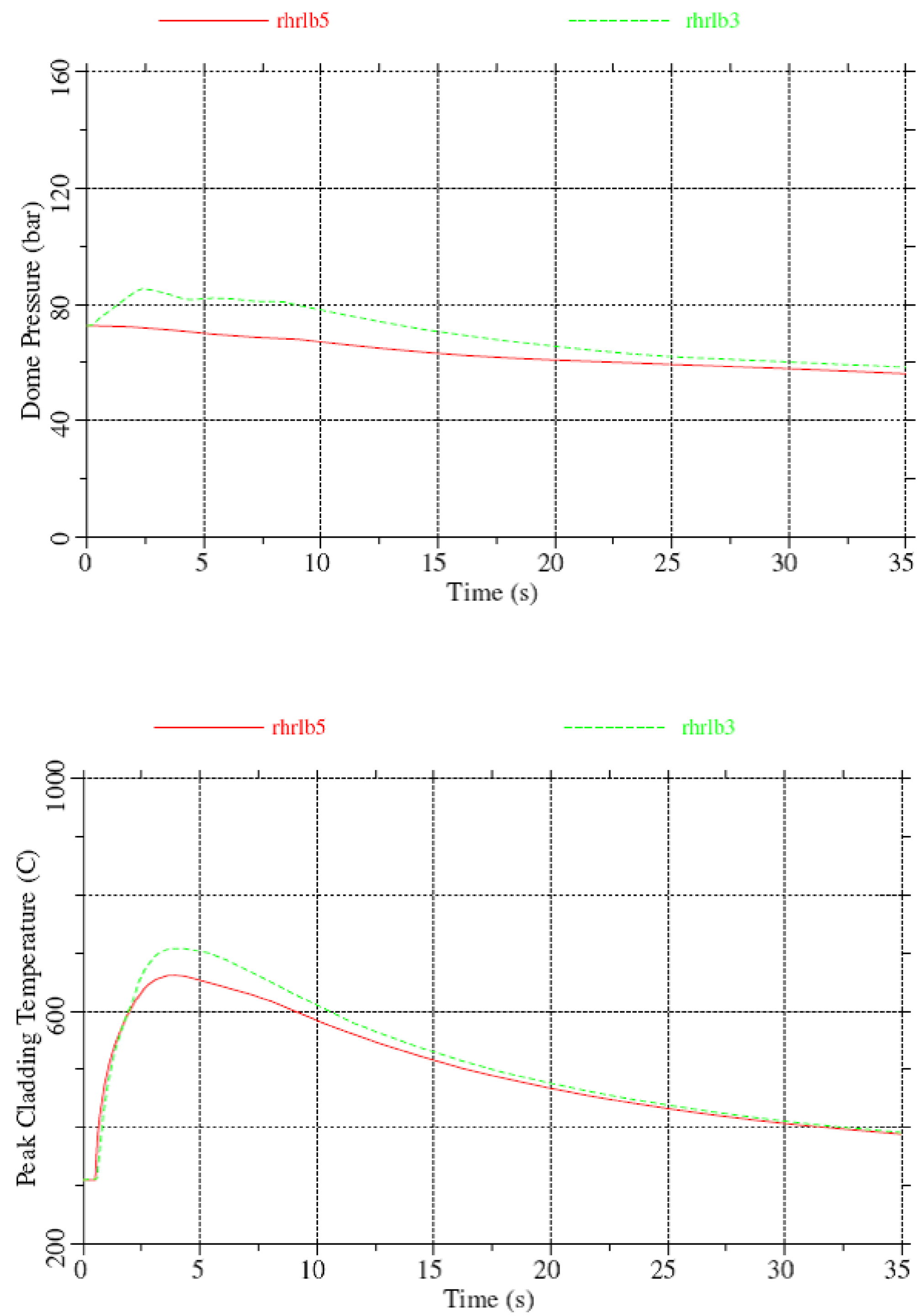


Figure 6.3-40 RHRLB Steam Line Isolation Sensitivity - Dome Pressure and GOBLIN PCT

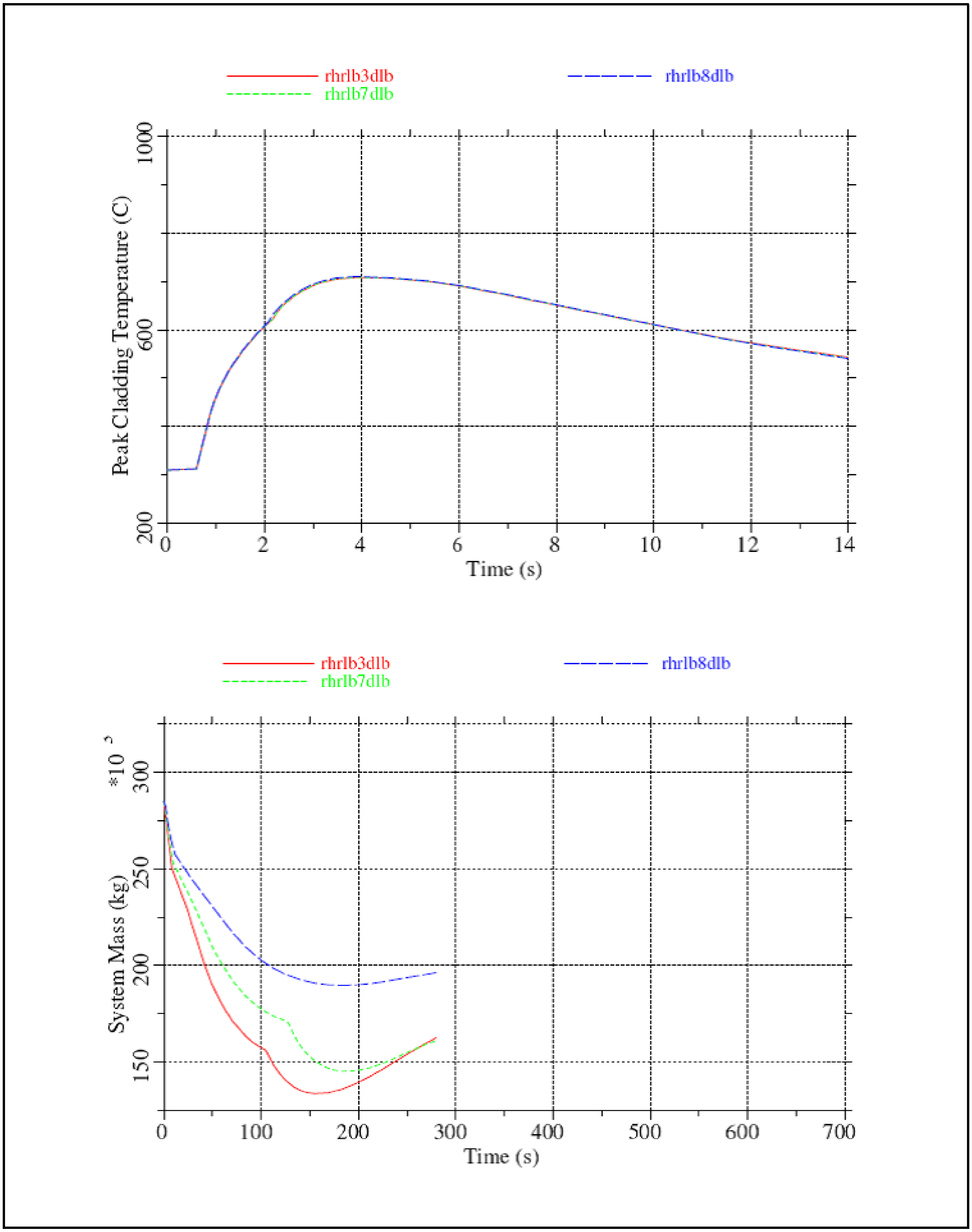


Figure 6.3-41 RHRLB Break Size Sensitivity - GOBLIN PCT and System Mass

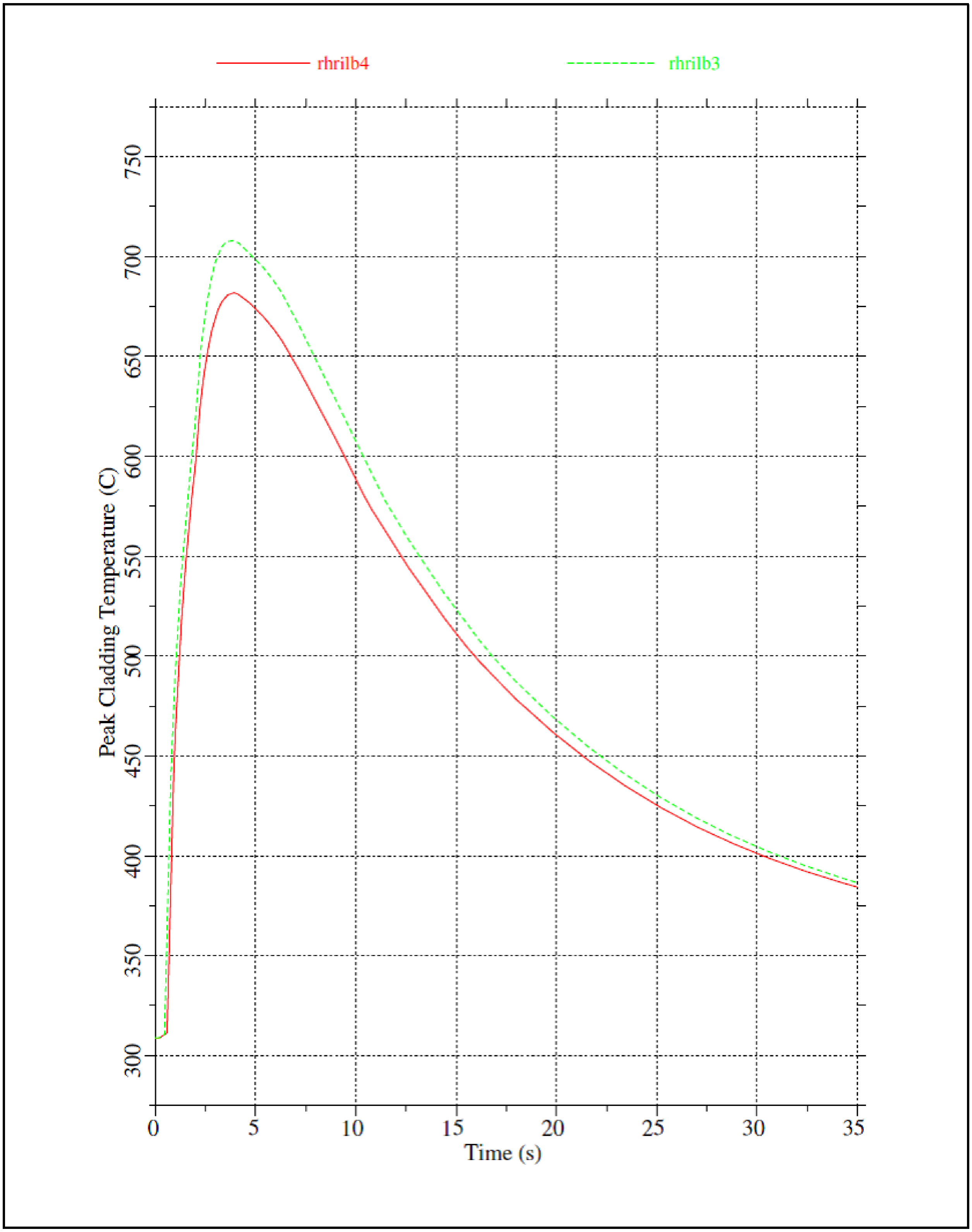
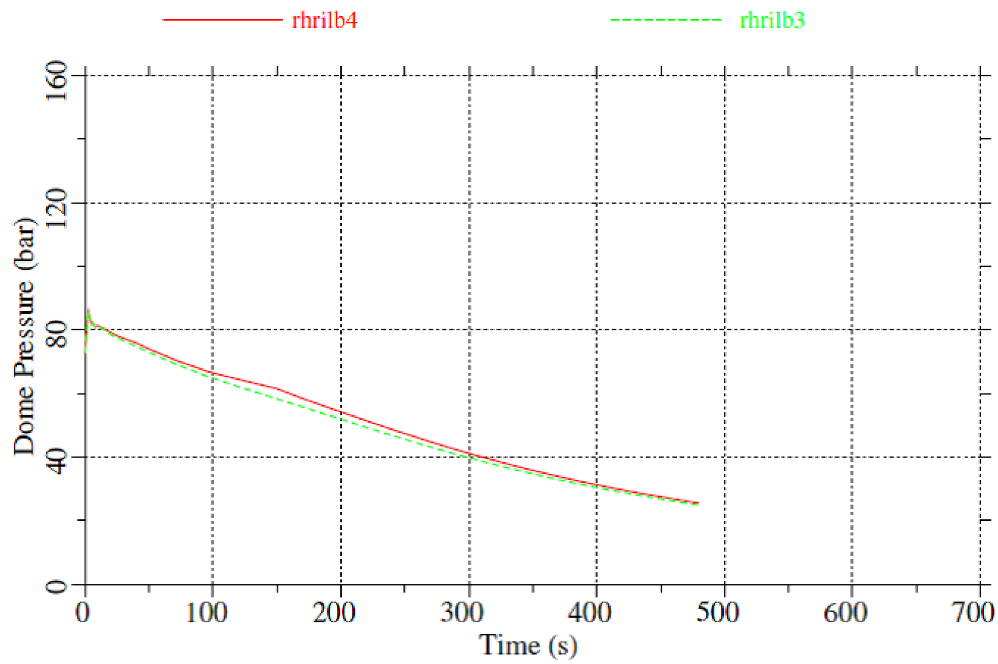


Figure 6.3-42 RHRLB Core Flow Rate Sensitivity - GOBLIN PCT



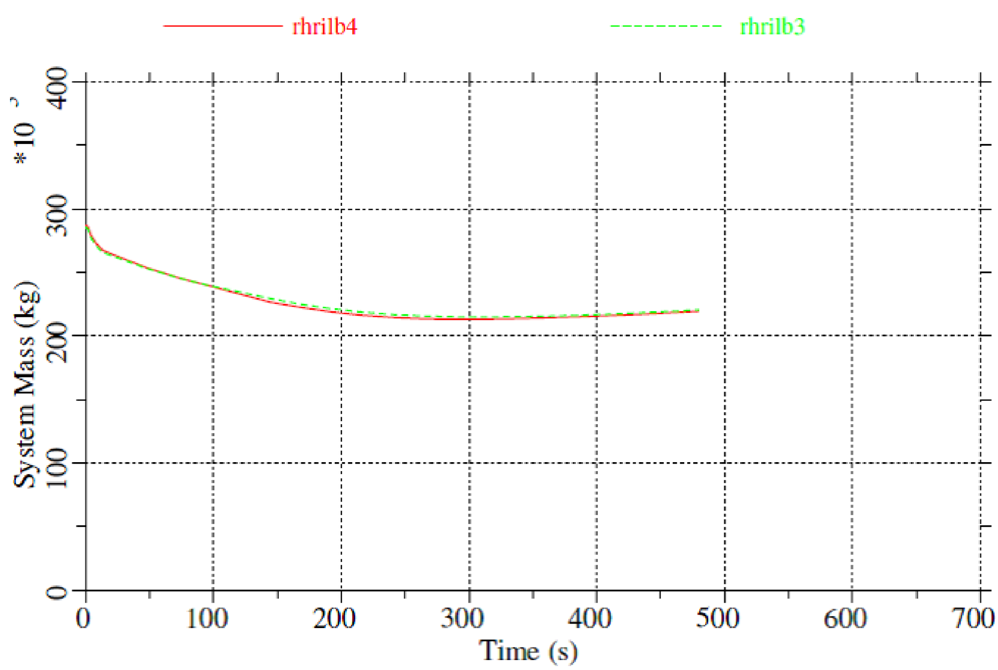


Figure 6.3-43 RHRLB Core Flow Rate Sensitivity - Dome Pressure and System Mass

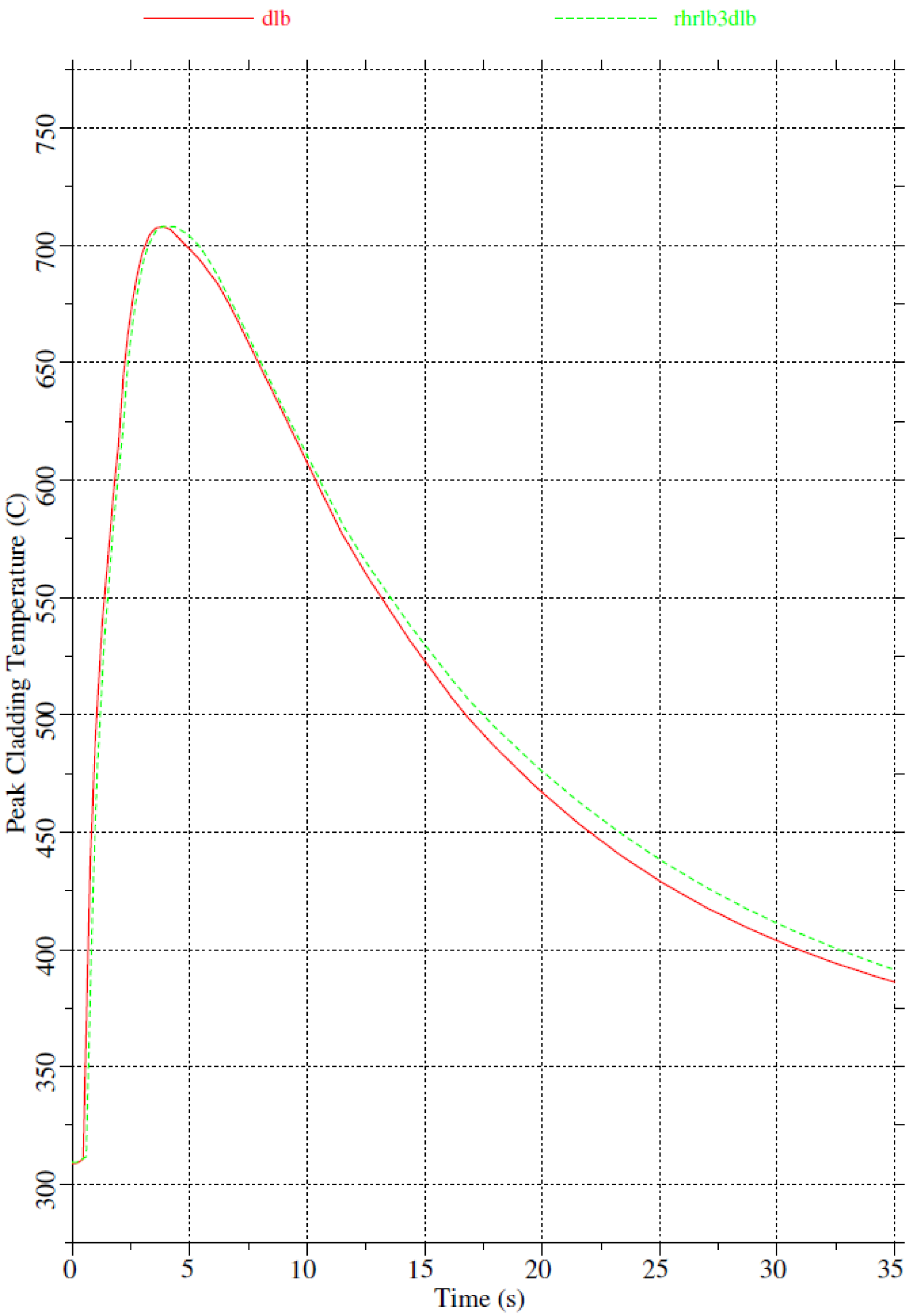


Figure 6.3-44 Comparison of DLB to RHRSLB - GOBLIN PCTs

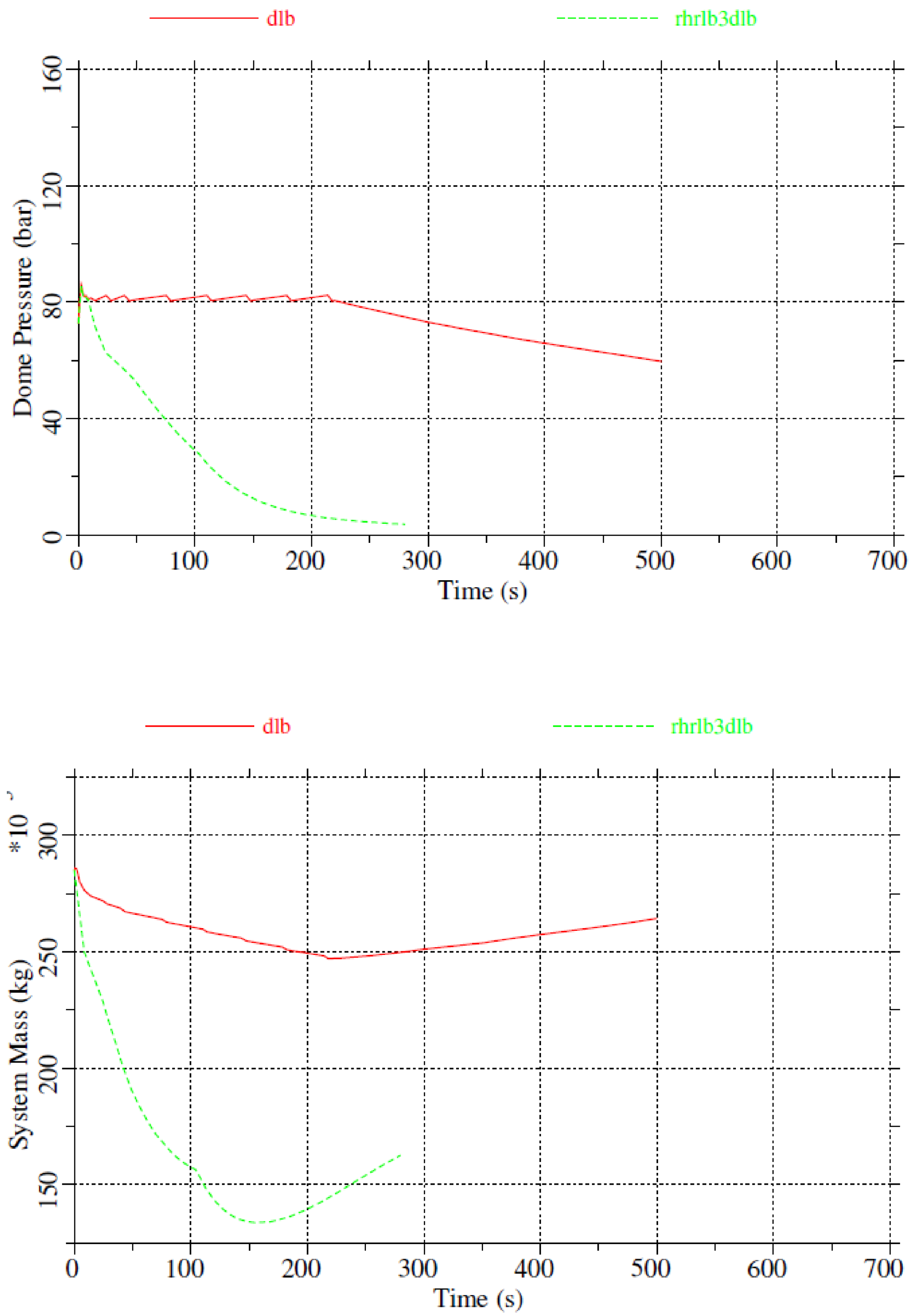


Figure 6.3-45 Comparison DLB to RHRSLB - Dome Pressure and System Mass

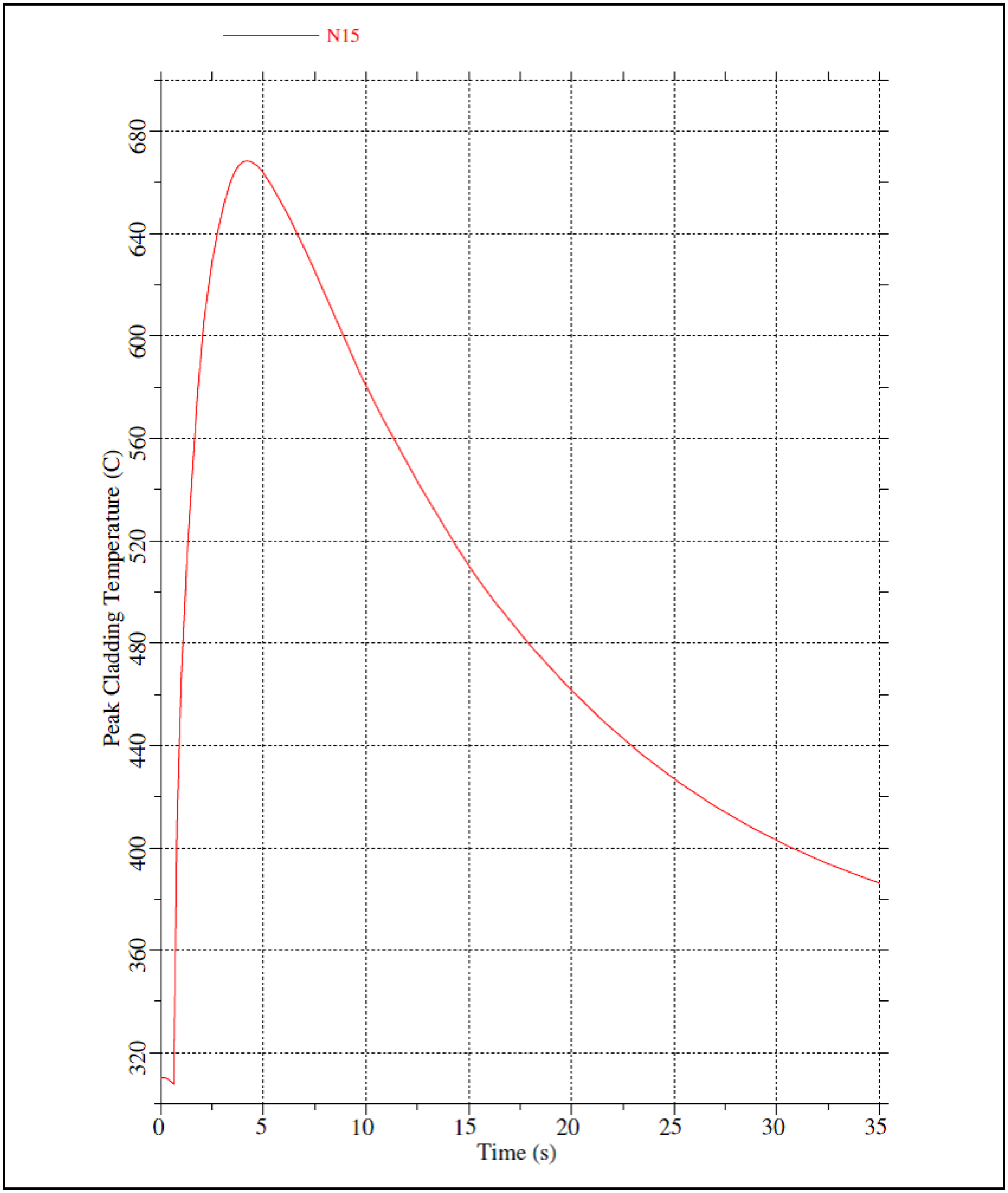


Figure 6.3-46 Steam Line Break Outside Containment - GOBLIN PCT

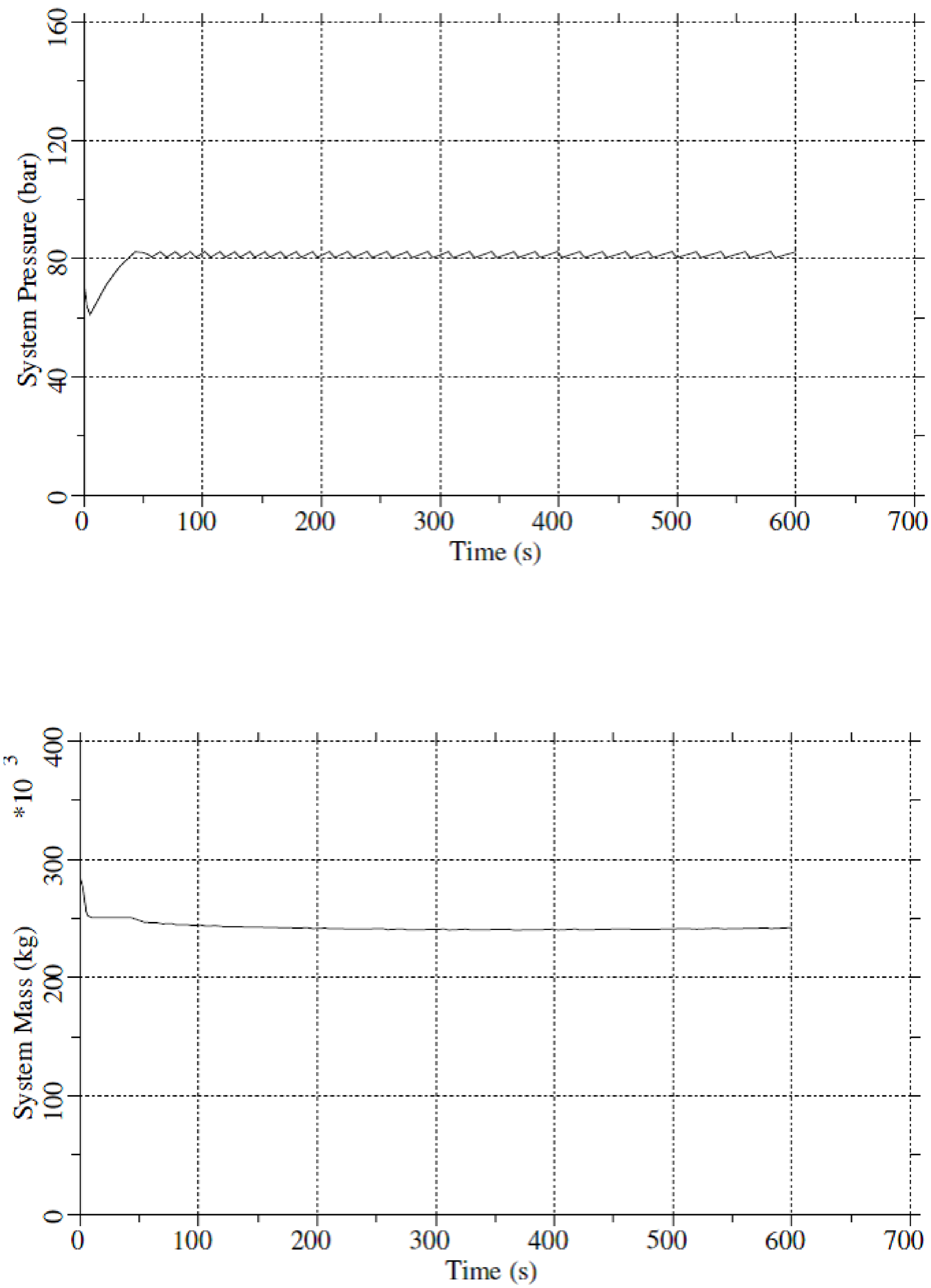


Figure 6.3-47 Steam Line Break Outside Containment - Dome Pressure and System Mass

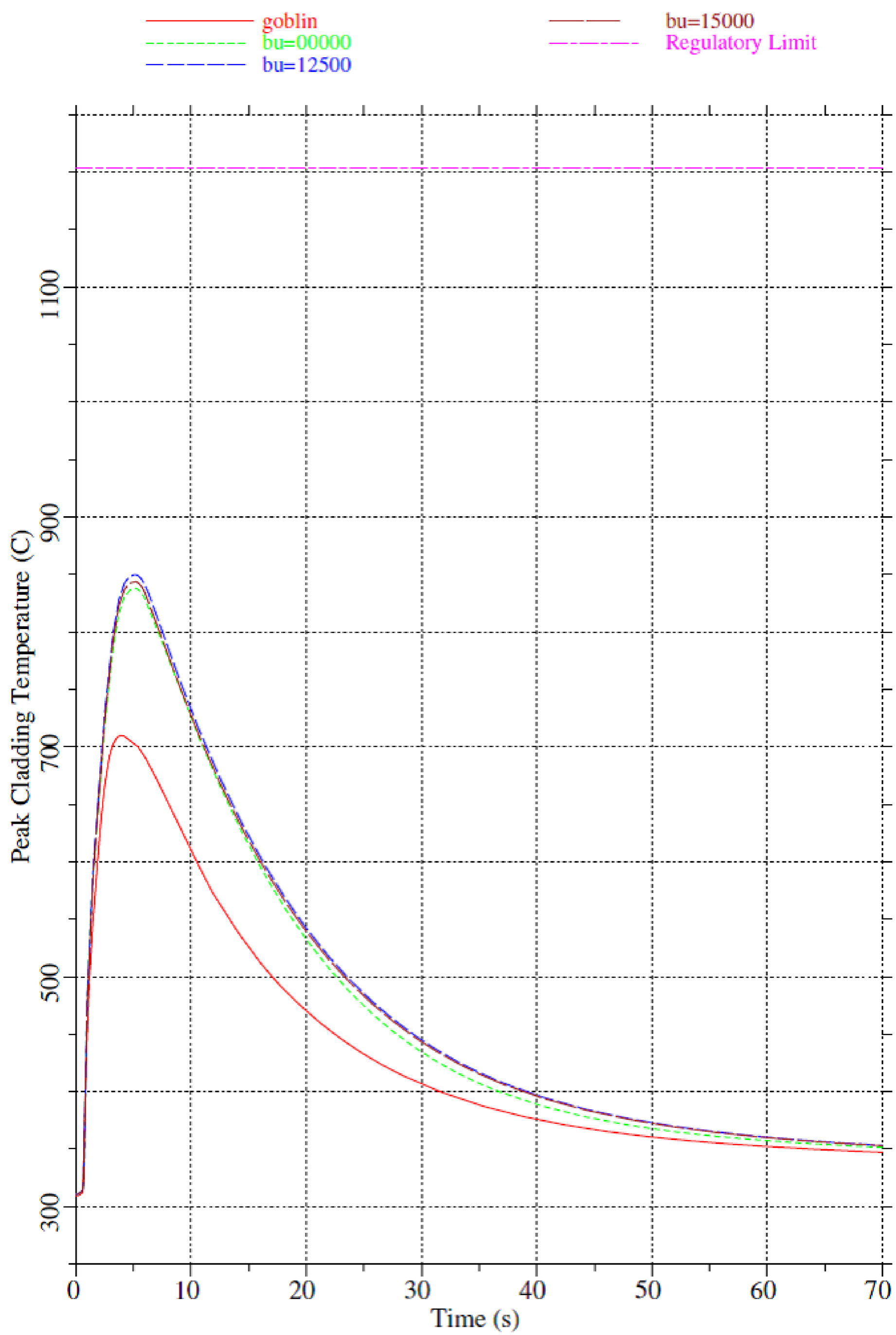


Figure 6.3-48 CHACHA Peak Cladding Temperature for Limiting Case

Figure 6.3-49 Not Used

Figure 6.3-50 Not Used

Figure 6.3-51 Not Used

Figure 6.3-52 Not Used

Figure 6.3-53 Not Used

Figure 6.3-54 Not Used

Figure 6.3-55 Not Used

Figure 6.3-56 Not Used

Figure 6.3-57 Not Used

Figure 6.3-58 Not Used

Figure 6.3-59 Not Used

Figure 6.3-60 Not Used

Figure 6.3-61 Not Used

Figure 6.3-62 Not Used

Figure 6.3-63 Not Used

Figure 6.3-64 Not Used

Figure 6.3-65 Not Used

Figure 6.3-66 Not Used

Figure 6.3-67 Not Used

Figure 6.3-68 Not Used

Figure 6.3-69 Not Used

Figure 6.3-70 Not Used

Figure 6.3-71 Not Used

Figure 6.3-72 Not Used

Figure 6.3-73 Not Used

Figure 6.3-74 Not Used

Figure 6.3-75 Not Used

Figure 6.3-76 Not Used

Figure 6.3-77 Not Used

Figure 6.3-78 Not Used

Figure 6.3-79 Not Used