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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC. 20555-0001
ATTENTION: MR. J. F. WILLIAMS

Dear Mr. Williams:

SUBJECT: PRELIMINARY STEAM GENERATOR TUBE RUPTURE ANALYSIS FOR IRIS

ATTACHMENT 1: PRELIMINARY STEAM GENERATOR TUBE RUPTURE ANALYSIS FOR IRIS

As I stated in my email to you last week, the attachment provides a preliminary steam generator tube rupture analysis for IRIS. This information is being provided to you prior to our meeting with the staff scheduled for April 15, 2004. The purpose of the attachment is to show that, while the IRIS SGTR transient is significantly different from conventional PWR SGTR transients, the results and consequences are relatively benign due to IRIS safety-by-design features (e.g., a main steam system designed for full primary pressure to the isolation valves).

We intend to use this information in our presentations to the staff on April 15. Eventually this information will be formally transmitted as a revision to WCAP-16082, "IRIS Preliminary Safety Assessment."

The information in the attachment is considered non-proprietary; therefore, there is no restriction on its distribution.

Please contact me (860) 731-6604 or Luca Oriani (412) 256-1692 if you need any clarification or additional information.

Regards,

Charles L. Kling, Licensing Manager
IRIS Project

ATTACHMENT:
cc: M. D. Carelli (W), R. A. Matzie (W, Windsor), J. E. Goossen (401 2x27A), C. B. Brinkman (W)

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Attachment 1 to STD-ES-04-09

PRELIMINARY STEAM GENERATOR TUBE RUPTURE ANALYSIS FOR IRIS

PRELIMINARY STEAM GENERATOR TUBE RUPTURE ANALYSIS FOR IRIS

1. Accident Overview

The steam generator tube rupture (SGTR) is an event for which IRIS presents a markedly different response from conventional PWRs.

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 into the steam generator.

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, since the higher pressure primary water flows outside the tubes, and the secondary feed water and steam is inside the tubes. Activity in the secondary side is subject to continual surveillance with radiation monitoring on each steam line, and an accumulation of such leaks is not permitted during operation, if they exceed the limit established in the Technical Specifications.

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 pair and eventual actuation of the emergency heat removal system (EHRS). Since the steam generators are designed for full primary design pressure up to the steam and feed water isolation valves, the isolation of the faulted steam generator does not result in the over-pressurization of any equipment, and this isolation automatically terminates the release of radioactivity. The detection of high radiation in the steam lines generates a reactor trip and isolates the steam generator pair associated with the faulted line and is a safety grade function. As a backup to the radiation monitors, the low pressurizer level and low pressurizer pressure trip signals may be reached if leakage from the reactor coolant system to the steam generators is greater than the capability of the makeup system.

The sequence of events that follows a postulated steam generator tube rupture depends on several assumptions, including the size of the break that is postulated to occur.

1. The event is initiated with a break in one or more tubes in the steam generators which leads to leakage of primary fluid into the lower pressure steam generator and to the secondary side steam discharge and feed water piping.
2. Following the break, energy will be released from the primary to the secondary system in addition to the normal heat transfer at the steam generators. The feedwater control system will reduce the feed water flow to the pair of steam generators that are removing more energy from the primary side to compensate for the mismatch. Depending on the size of the break (and in general for breaks up to and including a double ended rupture of one tube) the feedwater system will be able to compensate for the mass and energy input into the faulted steam

generator and prevent reaching any protection setpoints, other than high radiation.

3. While the feedwater control system maintains the heat balance, radiation in the steam line will increase, the radiation monitors 1) will detect high radiation and 2) will initiate reactor trip and main steam system isolation. From this point the sequence will proceed as described in 6, below.
4. For smaller leaks, where the loss of primary fluid mass is within the capability of the makeup system, the 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 this postulated break size where the secondary activity does not exceed the Technical Specification limit, protection is provided by the operator detecting the continuous leakage from the primary system.
5. If the break is beyond the capacity of the makeup system, <u>and</u> assuming that the high radiation reactor trip and related main steam system isolation is not available, the event will evolve as follows:
   a. As primary inventory is continuously lost to the secondary side through the break, the low pressurizer level or low pressurizer pressure trip setpoints will be reached (the larger the break, the faster the reactor trip setpoint will be reached). It is conservative to assume that the normal control grade controls (feedwater flow and pressurizer pressure control) are operating, since this will delay the time of reactor trip (preventing or delaying a reactor trip on low pressurizer pressure, for example) and will maximize the loss of primary inventory.
   b. Following reactor trip, the turbine will be tripped and the feedwater control system will be switched to level control on each steam generator pair. The startup feedwater system will maintain level near its setpoint. The pressure in the steam generators will increase until the steam dump system is actuated to relieve steam to the condenser. The assumption that the start-up feedwater system and steam dump system are available is conservative, since the operation of these two systems will act to delay the main steam system isolation and maximize the loss of mass from the reactor coolant system. In principle, a loss of offsite power could be assumed to follow the turbine trip, thus leading to an unavailability of the condenser and thus of the steam dump system. In this case, the steam pressure will rapidly rise to the high pressure EHRS actuation signal, after which the sequence will proceed as discussed in 6, below.
   c. As level decreases in the reactor coolant system, the shut-off setpoint for the pressurizer heaters will be reached, and the pressurizer pressure will start to decrease more rapidly. Eventually the pressurizer pressure will decrease until the S-signal setpoint is reached. Upon reaching the S-signal setpoint, an actuation signal for the EHRS will be generated, and this will in turn generate an isolation signal for the main steam system.
6. Once an isolation signal for the main steam system is obtained, the reactor coolant system will rapidly fill the faulted steam generator pair up to the steam isolation valves. The loss of mass from the primary system will terminate when the faulted steam generator pair fills, and an equilibrium condition will be reached with the EHRS cooling down and depressurizing the plant. It should be noted that the break may be located in a steam generator pair for which an EHRS subsystem is actuated (2 of 4 subsystems are actuated on a S-SIGNAL). This would have only a limited effect on the transient response since it has been demonstrated in the feedline break analysis that a single subsystem of the EHRS

is sufficient to remove the core decay heat. The total volume of a steam generator pair, including the connected EHRS subsystem and the feed and steam lines up to the isolation lines, is about 13 m^3 (460 ft^3). Since the reactor coolant system, excluding the pressurizer, has an inventory of over 420 m^3 (almost 15,000 ft^3) it is evident that the loss of mass from the primary system required to fill a steam generator pair will not significantly impact the reactor coolant system inventory.

The most conservative scenario in terms of loss of mass is identified in 5 and 6 above, and has been considered in a preliminary quantitative assessment of a SGTR for IRIS. The evaluation model used in this assessment is described in the following section.

It should be stressed that the sequence described above is completely automatic and no operator action is required to terminate a steam generator tube rupture for which the level of radioactivity in the steam line exceeds the technical specification limits.

2. Method of Analysis

A preliminary analysis is provided for the steam generator tube rupture, to support the conclusions reached in the previous section.

A modified version of the RELAP 5 Mod3.3 code (see Section 2.0.11.1 of WCAP-16082-NP) has been used to develop a model of the IRIS primary and secondary system to study the overall thermal-hydraulic plant behavior. The program simulates the neutron kinetics, reactor coolant system, reactor protection system, steam generators and safety systems. The program computes pertinent plant variables including temperatures, pressures, and power level.

The major assumptions used in the analysis are summarized below.

- **Initial Operating Conditions.** The plant is assumed to initially be operating at nominal, full power conditions. Given the plant response, it is not expected that any difference in initial conditions will significantly affect the transient.
- **Reactivity Coefficients.** Reactivity coefficients are not expected to significantly impact the analysis. To minimize the power reduction during the depressurization, minimum feedback coefficients are assumed. A conservatively large absolute value of the Doppler-only power coefficient is used (see Table 2.0-5). This is equivalent to a total integrated Doppler Reactivity from 0- to 100-percent of 0.016 Δk.
- **Reactor Control.** Assumptions relative to the rod control system are not expected to impact the system response during the event. The reactor is assumed to be in automatic control before the time of reactor trip.
- **Steam Relief.** As discussed above, the steam dump system should be considered available since its availability will delay the time of steam line isolation and maximize the loss of inventory from the primary system.
- **Pressurizer Pressure Control System.** The pressurizer pressure control system is assumed to be operable during the transient since this will slow the primary system pressure decrease due to the loss of inventory. Both proportional and backup

heaters are available until the water level in the pressurizer drop below the heater *shut-off setpoint.*

- **Feedwater Flow.** The feedwater control system is assumed to operate as designed during the event. The feedwater control system will in fact compensate for the increase energy released from the primary to the secondary side at the break thus delaying the time of actuation of the protection system. Since the design of the feedwater control system has not yet been completed, a simplified approach was used in this analysis, whereby the feedwater flow to the faulted steam generator pair is manually reduced until the feedwater flow rate that minimizes the perturbation on plant parameters is determined with an iterative process.

- **Reactor Trip.** Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the high radiation monitors. This is done to provide an overly conservative scenario and provide confirmation that the sequence of events that follows a SGTR for IRIS is very mild. Trip signals are expected due to low pressurizer pressure (the reactor trip setpoint) or low pressurizer level.

- **Safety Systems and Single Failure.** By design, no single failure prevents operation of the safety systems required to function. The EHRS actuation and main steam system isolation may be required to terminate the loss of mass from the primary side and to cool-down the reactor coolant system and the steam generators. EHRS actuation is expected on low pressurizer pressure (the safety actuation setpoint), not on any secondary system signal since the steam dump and startup feedwater system are assumed to operate to control the secondary side pressure and inventory thus preventing the SGs from initiating an EHRS actuation.

- **Availability of Offsite Power.** The analysis is provided only for a case with offsite power available. The reason for this is that shortly after the reactor trip, the low pressurizer pressure setpoint is reached resulting in the trip of the reactor coolant pumps. Therefore, this case is not very different from a case where a loss of offsite power is assumed to follow the turbine trip. Therefore, no additional analysis for a case with loss of offsite power assumed was considered necessary

- **Break.** A guillotine rupture of one of the steam generator tubes is assumed in the analysis. Note that a spectrum of smaller and larger break sizes were considered to demonstrate how the transient is only affected in a limited way by the break size. The larger breaks lead to a faster transient evolution, while smaller breaks result in longer, milder transients. It is important to notice how the end result (total inventory in the reactor coolant system) at the end of the analysis time, is not significantly affected by the size of the break. This is an expected IRIS result since the final reactor coolant system inventory is essentially the inventory corresponding to the low pressurizer level setpoint, minus the inventory required to fill the faulted steam generator pair (up to the main steam and feed isolation valves) and to fill the associated EHRS subsystem.

  The break can either be assumed to occur on a steam generator that is connected to an EHRS subsystem that is actuated during the transient, or to a steam generator connected to a steam generator whose connected EHRS subsystem is not actuated during the transient. Both locations were considered in the analysis, and results for the limiting case (i.e. maximum loss of mass from the reactor coolant system) are presented. In this analysis this limiting case occurs with the break located on steam generator #1 (connected to one of the EHRS subsystems automatically actuated during the transient).

3. Results

The sequence of events following a postulated single steam generator tube rupture event is provided in Table 1. Figures 1 through 7 illustrate the transient behavior of the key plant parameters. Figure 2 shows the break flow rate and the integrated reactor coolant system loss of mass. The loss of reactor coolant system fluid is slowed following the trip of the turbine and the subsequent increase in the steam generator pressures, and is finally terminated once the reactor coolant system and steam generator pressures equalize. This pressure equalization occurs after about 1700 seconds, as shown in Figure 3. After this time, the loss of mass from the reactor coolant system to the faulted steam generator pair is due only to the cooldown of the SG and RCS, with the steam generator system isolated. The plant cooldown by a single EHRS subsystem is confirmed by the reactor coolant system temperature shown in Figure 4.

As shown in Figure 5, the pressurizer level decreases during the transient, with the reactor trip setpoint reached about 12 minutes into the transient, and the pressurizer finally empties at approximately 20 minutes after the initiation of the transient.

Figures 6 and 7 show the steam generator levels (averaged for each pair) and pressures during the event.

These results show that a single steam generator tube rupture is effectively and automatically mitigated in IRIS without any requirement for operator action. Also, the loss of mass from the reactor coolant system is terminated as soon as the steam system is isolated, and the plant is then cooled down and depressurized by the emergency heat removal system.

Note that only a thermal hydraulic analysis of this event is provided in this preliminary study, but on the basis of the total steam released prior to SG isolations it can be concluded that this release would have negligible impact on the site boundary dose.

Finally, a sensitivity study was performed for different break sizes, ranging from a 40% split break to 50 tubes. Note that the case corresponding to a break with a flow area equivalent to a guillotine rupture of 50 tubes is only provided for scoping purposes to show the mild influence of the break size on the total loss of mass. Figure 8 shows the time to reactor trip and to steam line isolation for this spectrum of postulated ruptured tubes, and, as expected, the larger the break size, the faster the transient evolves. Also, for the larger breaks, the reactor trip occurs on low pressurizer pressure rather than on low pressurizer level, since the pressurizer pressure control system is not capable of mitigating faster RCS depressurization that occurs. While the difference in the time at which reactor trip and main steam system isolation occurs is significantly different between the different cases, Figure 9 shows that the final total integrated break flow for all the break size cases tends to be similar. For example, the hypothetical break of 50 tubes (50 times the break flow area of a single tube rupture), only results in a 50% increase in integrated break mass loss from the primary system, which is still a very small fraction of the primary inventory. This confirms that the IRIS response to a steam generator tube rupture is not significantly impacted by the postulated number of faulted tubes and that the total loss of mass from the primary system is always small compared to the reactor coolant system water inventory.

Table 1

TIME SEQUENCE OF EVENTS FOR STEAM GENERATOR TUBE RUPTURE

Accident	Event	Time (seconds)
Steam Generator Tube Rupture		
	Double ended rupture in a single of tube of a steam generator occurs.	20.0
	Low pressurizer level reactor trip setpoint occurs.	741.9
	Rods begin to fall, turbine trip occurs.	743.9
	Reactor coolant pumps begin to coastdown.	758.9
	Steam dump system first actuation to relieve pressure from the steam generators.	~760
	Startup feedwater delivered to the steam generators.	803.8
	Low pressurizer pressure S-Signal reached.	1111.1
	Main feed isolation valves completely closed.	1113.2
	Main steam isolation valves completely closed: release of primary fluid to the condenser is terminated.	1123.2
	Partial EHRS actuation: subsystems 1 and 3 actuated.	1125.2
	Pressure in the RCS and the faulted steam generator equalizes: loss of RCS fluid to the SGs only occurs due to the cooldown as the EHRS proceeds to bring the plant to safe shutdown conditions.	~1700

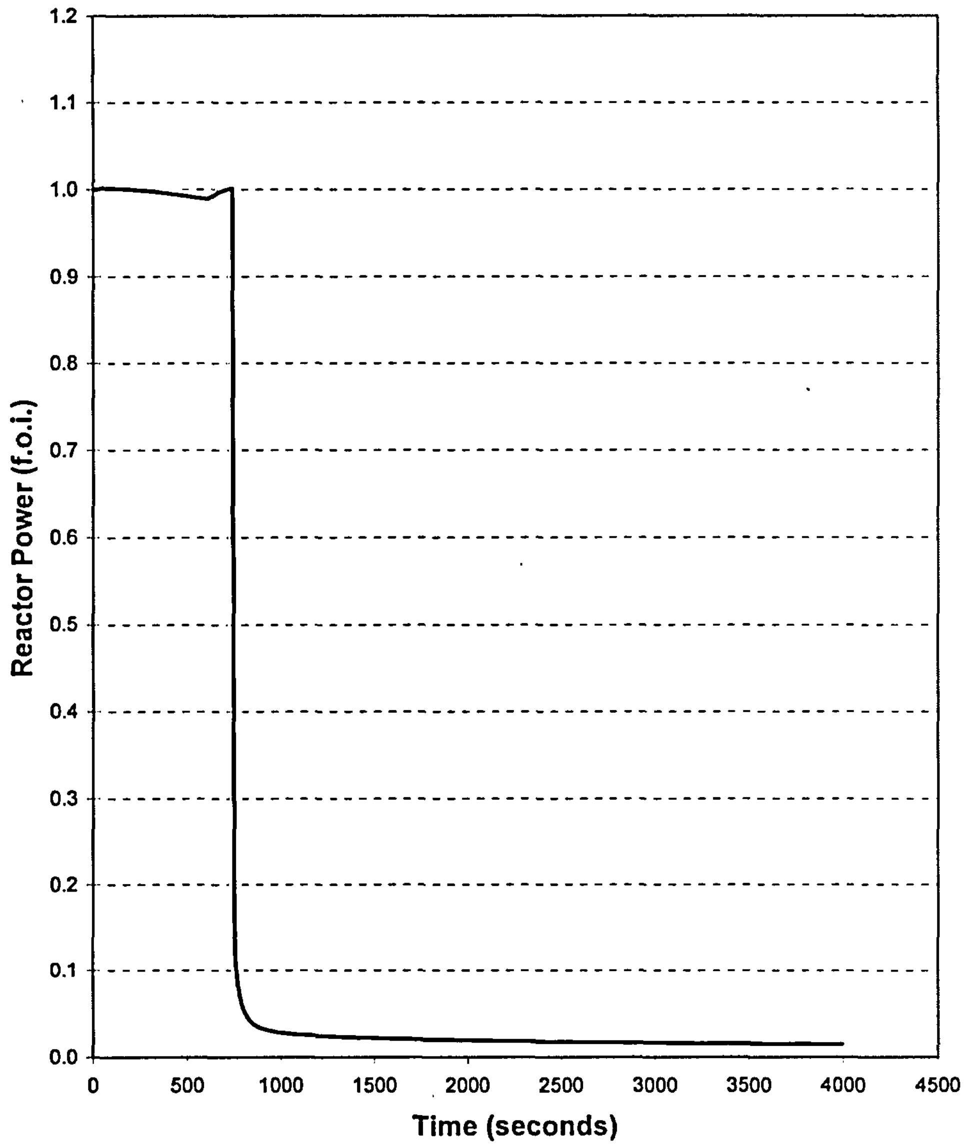

Figure 1
Nuclear Power Transient for Double Ended Rupture of a Steam Generator Tube

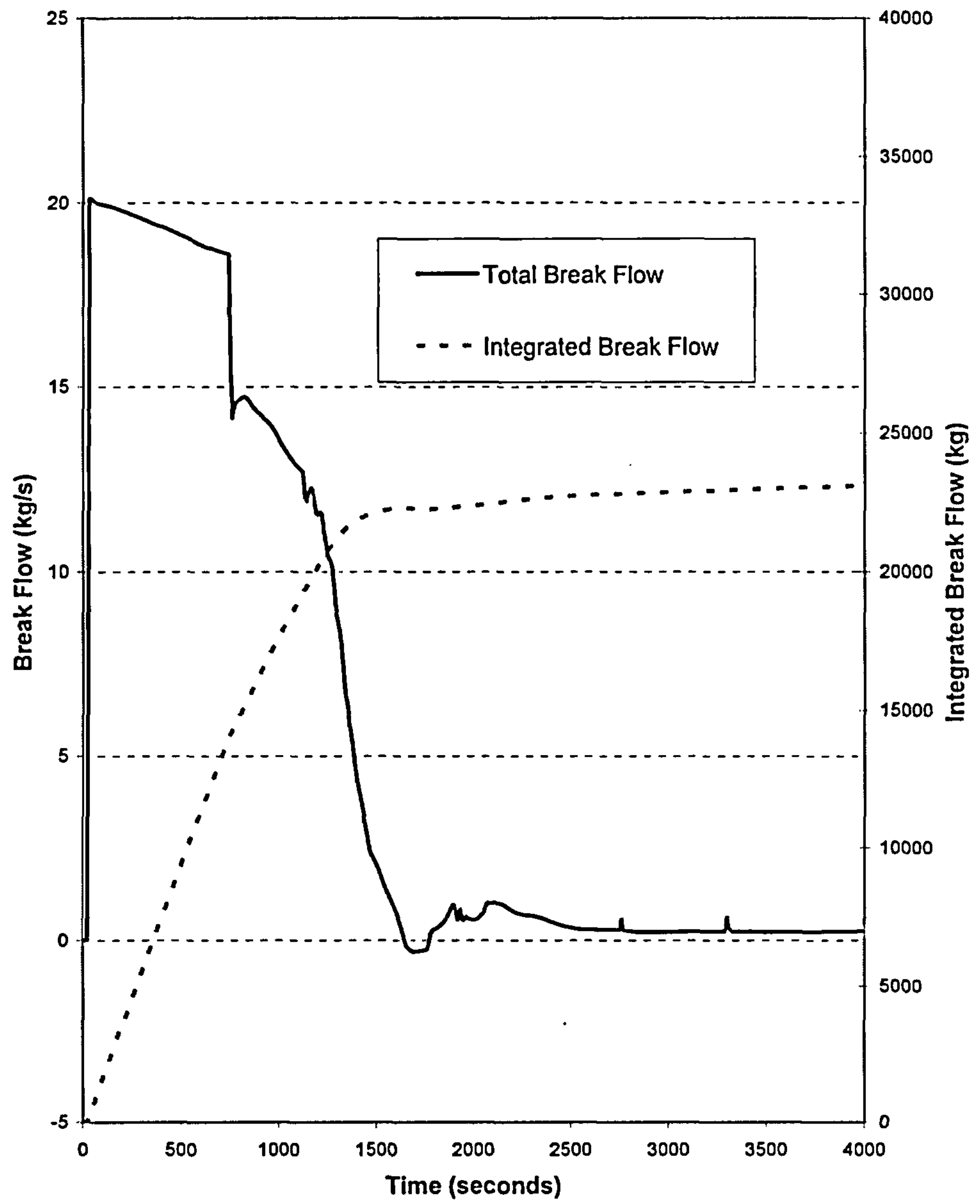

Figure 2
Break Flow Rate Transient for Double Ended Rupture of a Steam Generator Tube

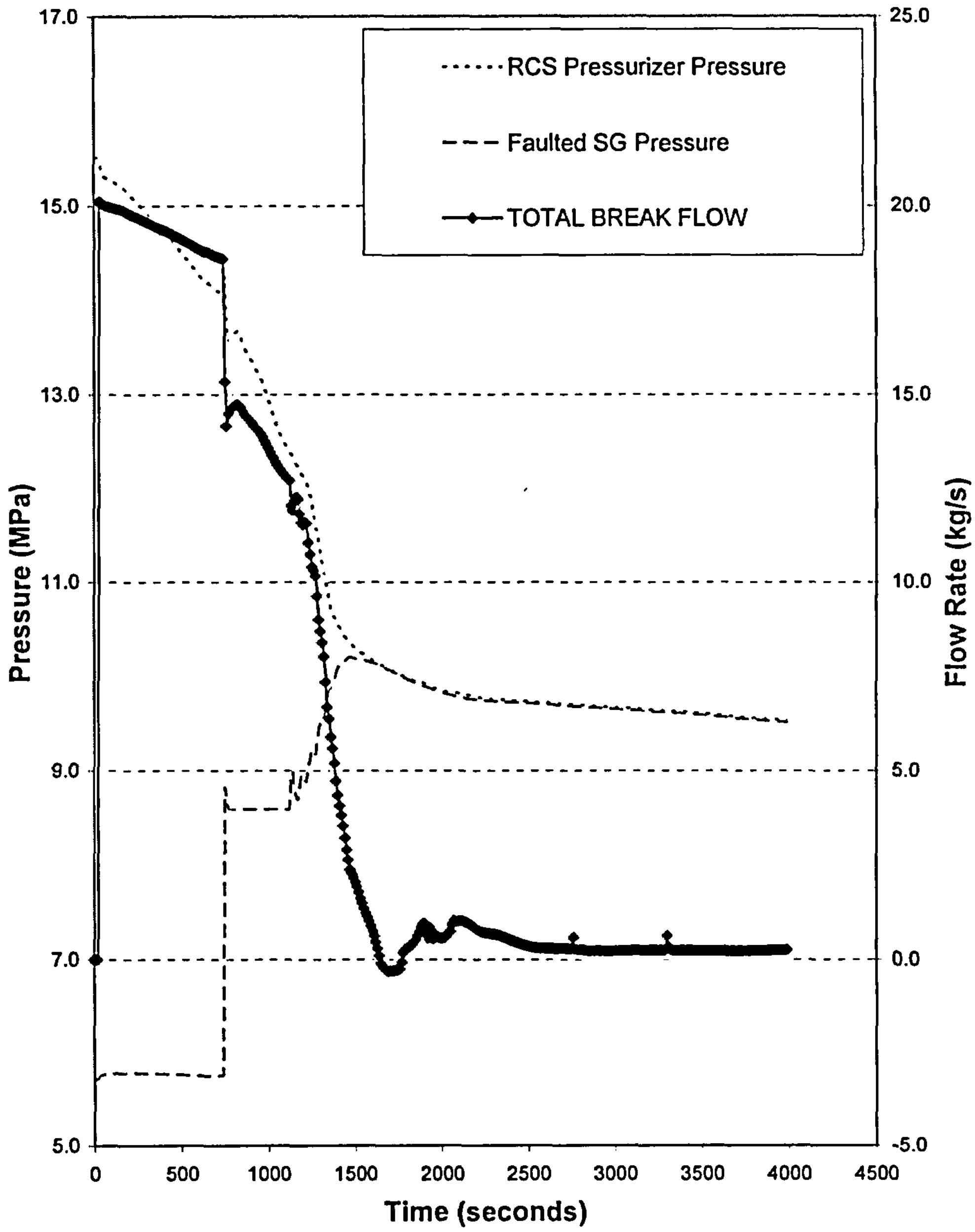

Figure 3
Break Flow and RCS and Faulted Steam Generator Pressure Transient for Double Ended Rupture of a Steam Generator Tube

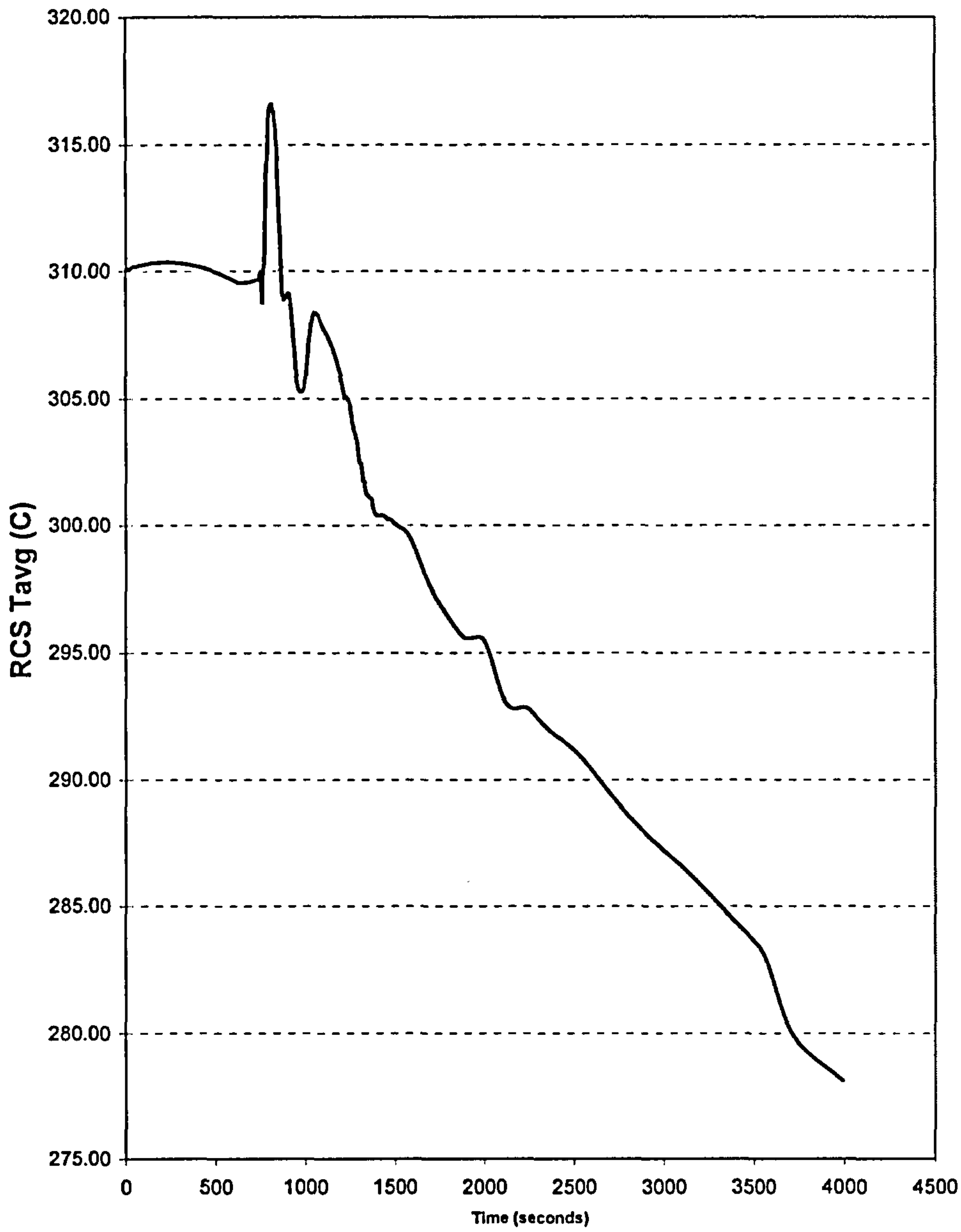

Figure 4
Reactor Coolant System Average Temperature Transient
for Double Ended Rupture of a Steam Generator Tube

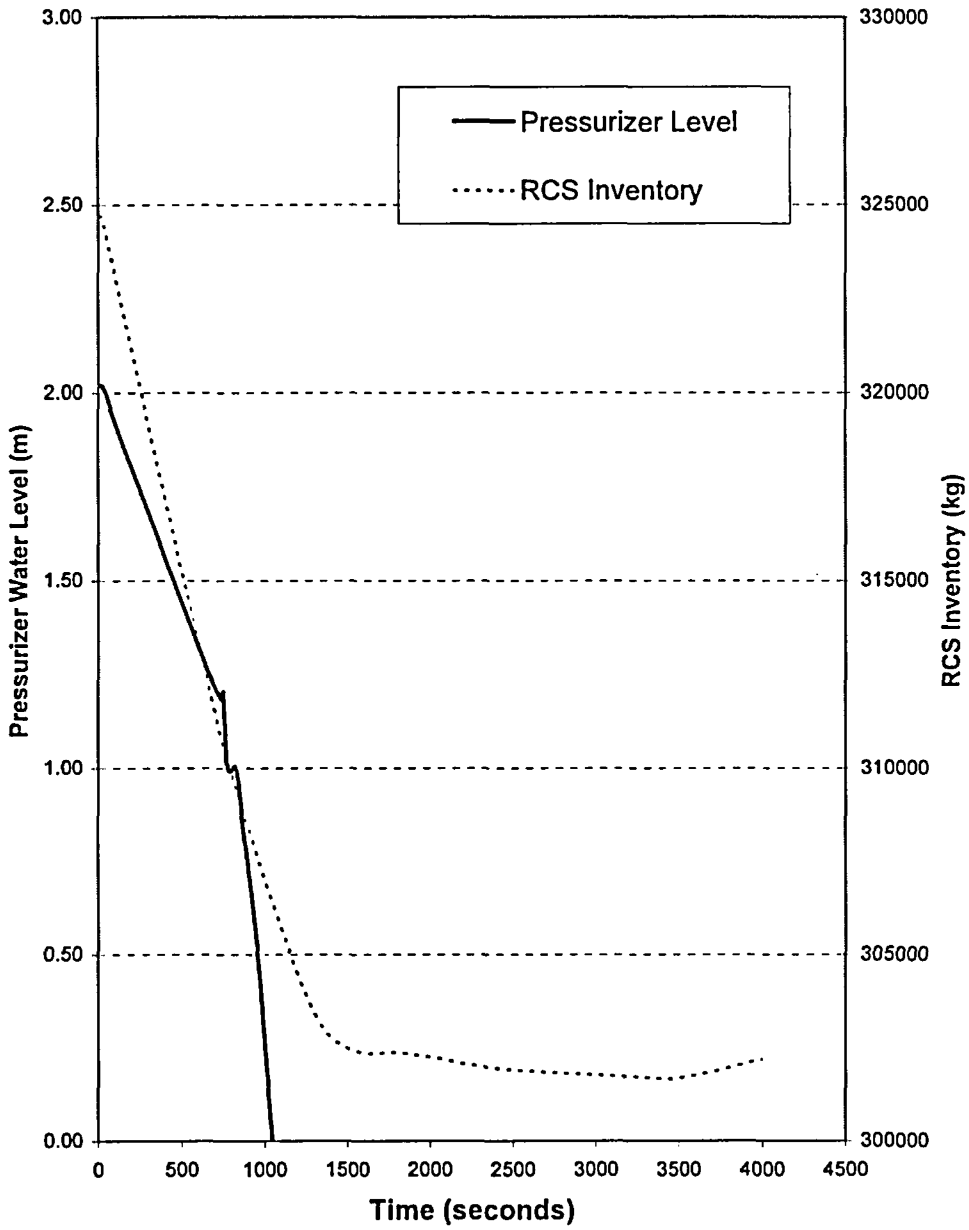

Figure 5
Pressurizer Water Level and RCS Liquid Inventory Transient for Double Ended Rupture of a Steam Generator Tube

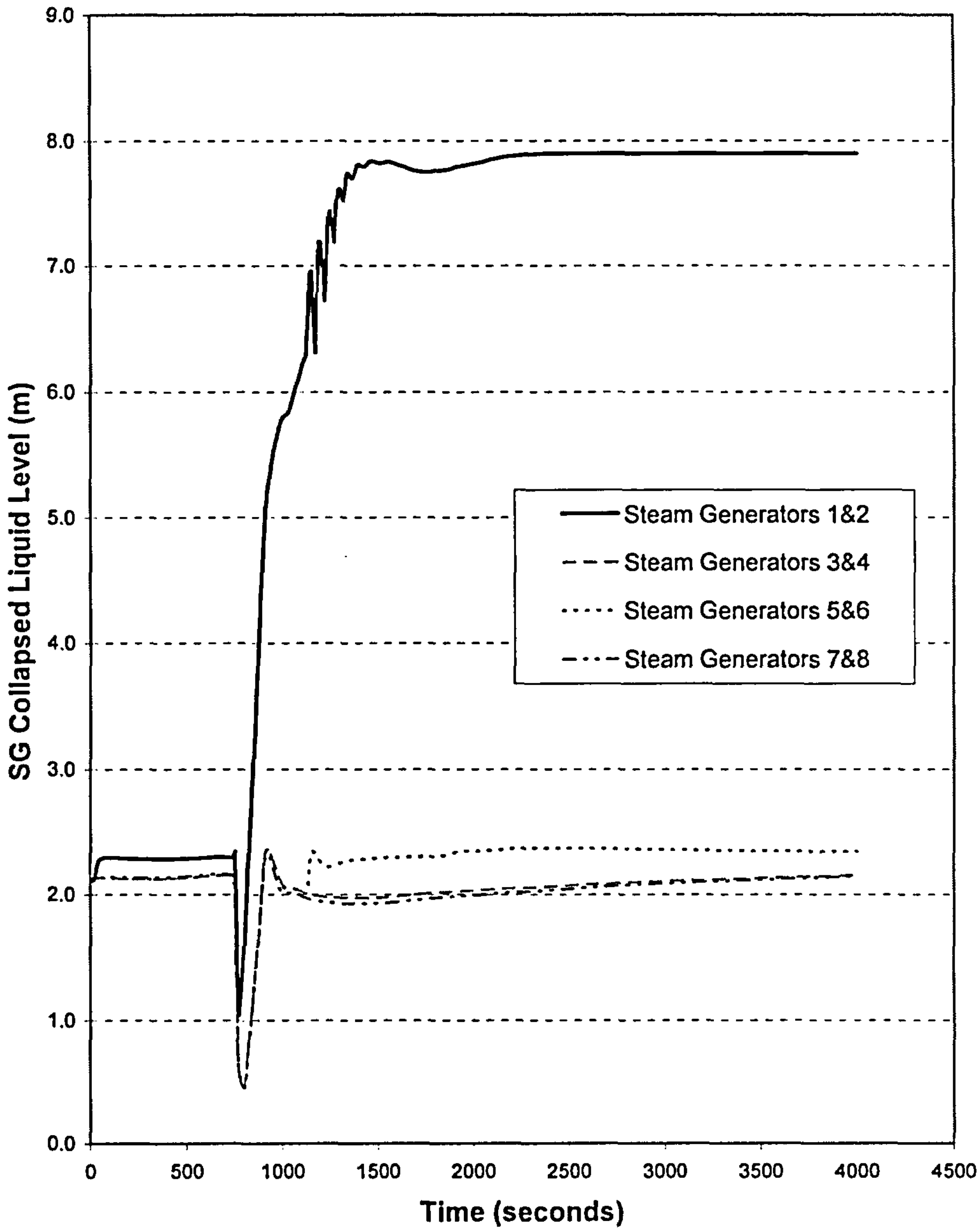

Figure 6
Collapsed Liquid Level in each of the Steam Generator Pairs for Double Ended Rupture of a Steam Generator Tube

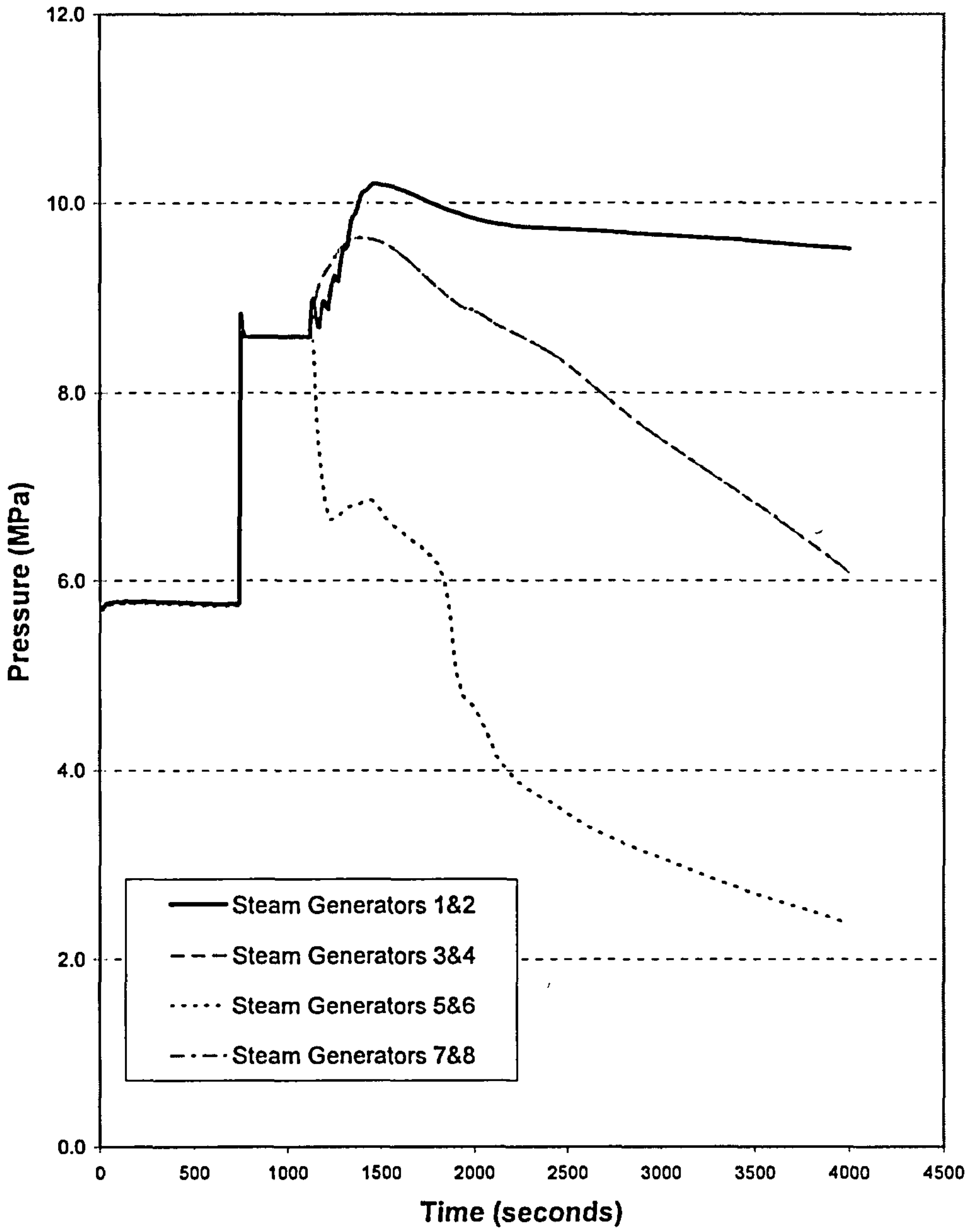

Figure 7
Steam Generator Pressure Transient for Double Ended Rupture of a Steam Generator Tube

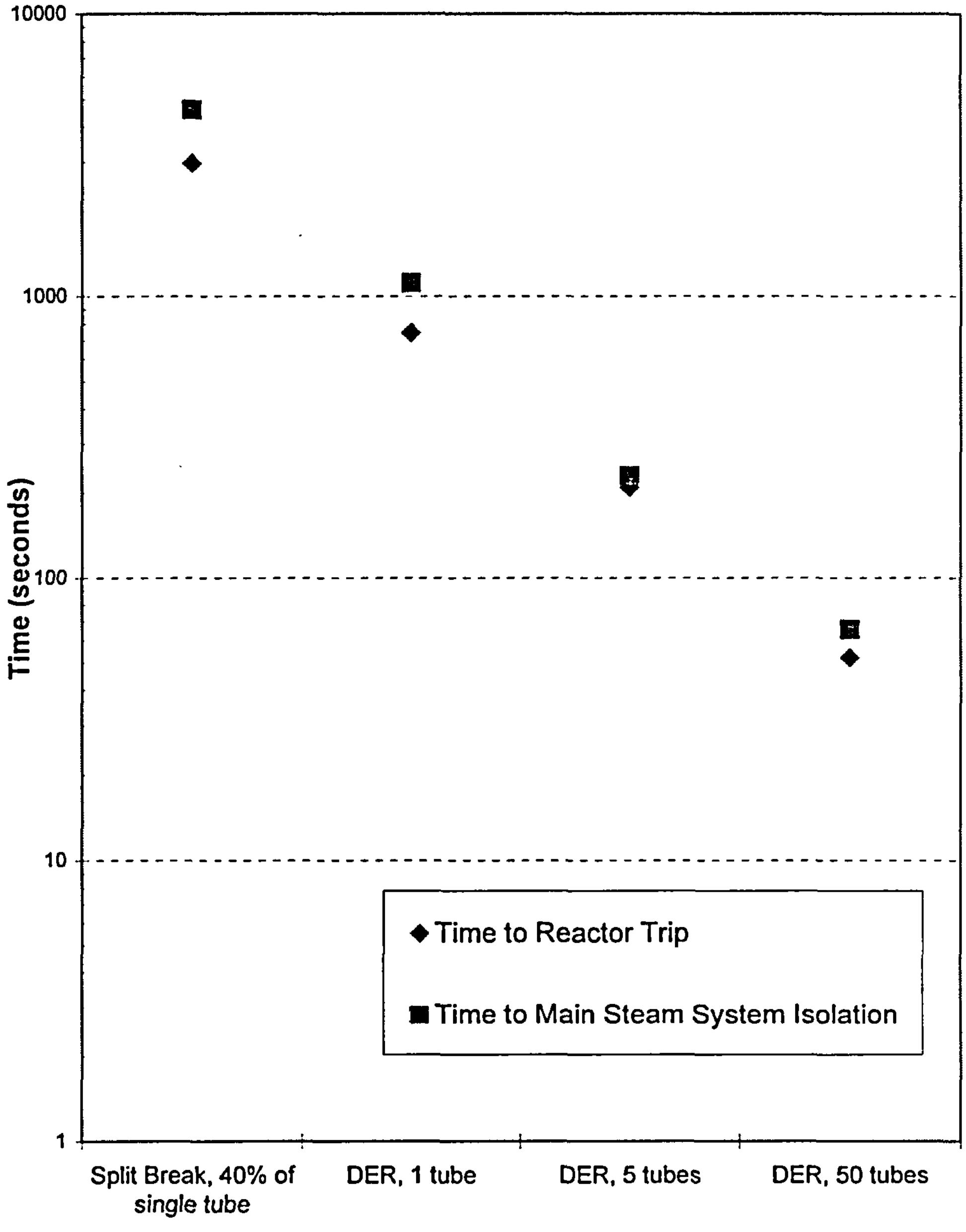

Figure 8
Time to reactor trip and steamline isolation following postulated breaks of different sizes: double ended ruptures of 1, 5 and 50 tubes, and a split break with area equivalent to 40% of a single tube.

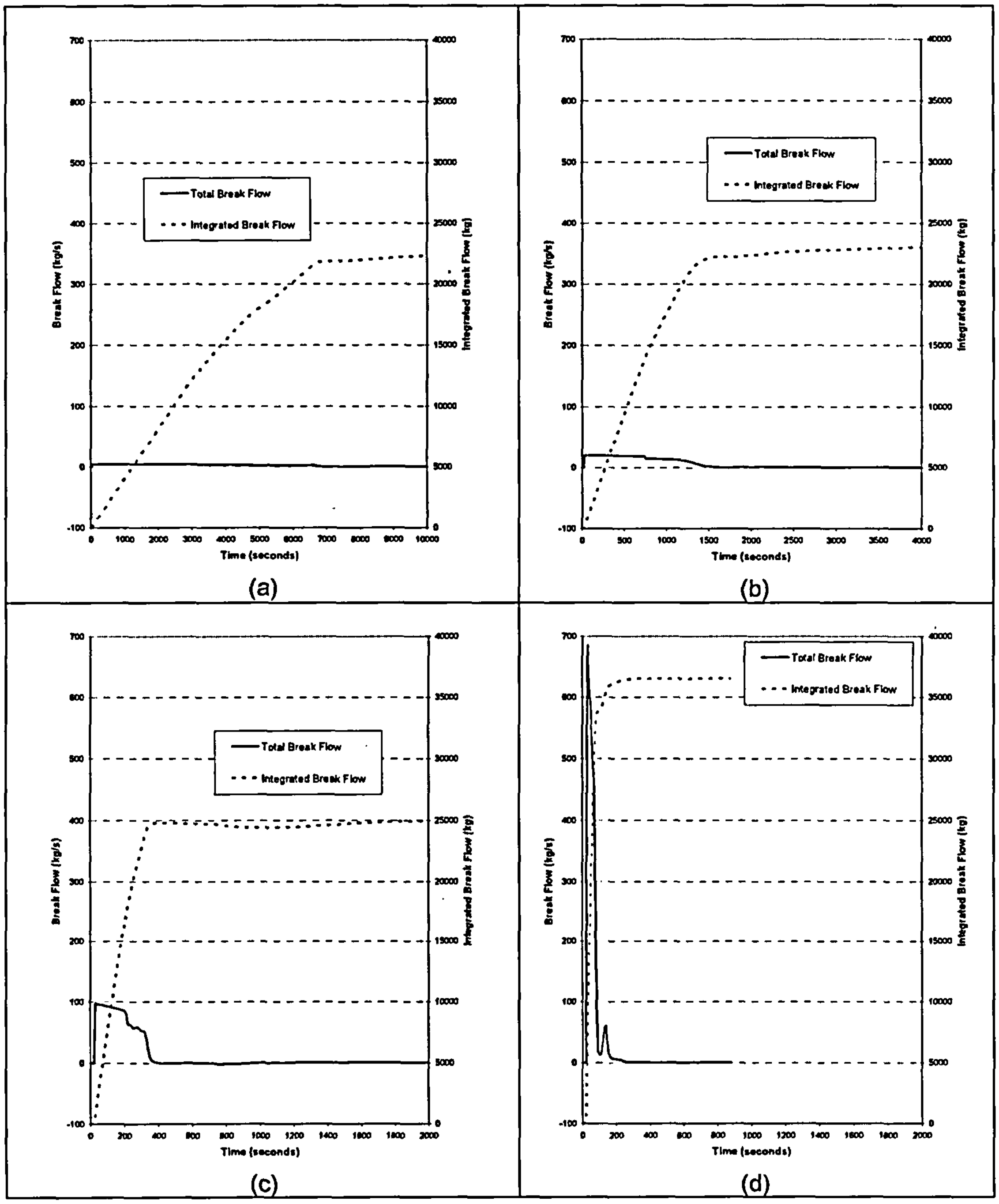

Figure 9
Break Flow Rate and integrated mass loss from the reactor coolant system for different break sizes: (a) split break, 40% of a single tube area, (b) double ended single tube rupture (base case), (c) 5 tubes guillotine break and (d) 50 tubes guillotine break. Note that different time scales are used for the different events.