CHAPTER 15 - ACCIDENT ANALYSES

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CHAPTER 15 - ACCIDENT ANALYSES

15.0 ORGANIZATION AND METHODOLOGY

This chapter presents analytical evaluations of the nuclear steam supply system (NSSS) response to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. Such incidents (or events) are postulated and their consequences analyzed despite the many precautions which are taken in the design, construction, quality assurance, and plant operation to prevent their occurrence. The effects of these incidents are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations.

15.0.1 Classification of Transients and Accidents

15.0.1.1 Format and Content

This chapter is structured according to the format and content suggested by Reference 1 and Reference 25.

15.0.1.2 Event Frequencies

Reference 1 subjectively classifies initiating events in the following qualitative frequency groups:

a. Moderate frequency events (incidents of moderate frequency)
b. Infrequent events (infrequent incidents)
c. Accidents (limiting faults)

15.0.1.3 Event Categories

Each postulated initiating event has been assigned to one of the following categories:

a. Increase in heat removal by the secondary system
b. Decrease in heat removal by the secondary system
c. Decrease in reactor coolant flow rate
d. Reactivity and power distribution anomalies
e. Increase in RCS inventory
f. Decrease in RCS inventory
g. radioactive release from a subsystem or component
h. anticipated transients without scram (ATWS)

The assignment of an initiating event to one of these eight categories is made according to Reference 1. As discussed in Section 15.8, 10 CFR 50.62, "Requirement for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light Water Cooled Nuclear Power Plants," provides the USNRC's requirements for the reduction of risk from ATWS events. The YGN 3&4 design is sufficiently reliable such that ATWS events need not be considered in the design basis.

Table 15.0.1-2 provides a listing of the initiating events and the corresponding frequencies.

15.0.1.4 Events and Event Combinations

The events and event combinations in this chapter are those identified by Reference 1, and are presented with respect to the event specific acceptance criteria specified therein. For each applicable acceptance criterion in an event category, only the limiting event or event combination is presented in analytical detail. Qualitative discussions are provided for all other events or event combinations explaining why they are not limiting.

For event combinations which require consideration of a single failure, the limiting failure is selected from those listed in Table 15.0.1-1. Only "Delete" independent preexisting failures are considered credible and included in the table. Preexisting failures are equipment failures existing prior to the event initiation which are not revealed until called upon during the event (e.g., a failure of an auxiliary feedwater pump). High-probability dependent occurrences are always included in the event analysis in addition to the selected single failure, if they have an adverse impact (e.g., loss of main feedwater pumps following a loss of electric power).

15.0.1.5 Section Numbering

The incidents analyzed in this chapter are presented in sections in accordance with Reference 1 and are numbered as described in Table 15.0.1-3.

15.0.2 System Operation

During the course of any event various systems may be called upon to function. Some of these systems are described in Chapter 7 and include those electrical, instrumentation, and control systems designed to perform a safety function (i.e., those systems which must operate during an event to mitigate the consequences) and those systems not required to perform a safety function (see Sections 7.2 through 7.6 and 7.7, respectively).

The reactor protection system (RPS) is described in Section 7.2. Table 15.0.2-1 lists the RPS trips for which credit is taken in the analyses discussed in this section, including the setpoints and the trip delay times associated with each trip. The analyses take into consideration the response times of actuated devices after the value of the monitored parameter at the sensor reaches the trip setpoint.

The reactor protection system response time is the sum of the sensor response time and the reactor trip delay time. The sensor response time is defined as the time from when the value of the monitored parameter at the sensor equals the reactor protection system trip setpoint until the sensor output equals the trip setpoint. The sensor response is modeled by using a transfer function for the particular sensor used. The reactor trip delay time (Table 15.0.2-1) is defined as the elapsed time from the time sensor output equals the trip setpoint to the time the reactor trip breakers are fully open.

The interval between trip breaker opening and the time at which the magnetic flux of the control element assembly (CEA) holding coils has decayed enough to allow CEA motion is conservatively assumed to be 0.5 seconds. Finally, a conservative value of 3.5 seconds is assumed for CEA insertion, defined as the elapsed time from the beginning of CEA motion to the time of 90% insertion of the CEAs into the reactor core.

The engineered safety features actuation system (ESFAS) and electrical, instrumentation, and control systems required for safe shutdown are described in Sections 7.3 and 7.4, respectively. The manner in which these systems function during events is discussed in each event description. The instrumentation which is required to be available to the operator in order to assist him in evaluating the nature of the event and determining required action is described in Section 7.5. The use of this instrumentation by the operator is discussed in each event description.

The RPS and ESFAS setpoints assumed in the safety analyses account for uncertainties in a conservative manner. The analytical setpoint assumptions are used in conjunction with normal and postaccident induced uncertainties to develop the Technical Specification setpoints in ~~Chapter 16~~.
ITS

Systems which may but are not required to perform safety functions are described in Section 7.7. These include various control systems and the core operating limit supervisory system (COLSS). In general, normal automatic operation of these control systems is assumed unless lack of operation would make the consequences of the event more adverse. In such cases, the particular control system is assumed to be inoperative, in the manual mode, until the time of operator action.

15.0.3 Core and System Performance

15.0.3.1 Mathematical Model

The NSSS response to various events was simulated using digital computer programs and analytical methods approved by the USNRC. For the limiting events in each section, Table 15.0.3-2 identifies these analytical methods and USNRC approvals.

15.0.3.1.1 Loss of Flow Analysis Method

The method used to analyze events which are initiated by failures which cause a decrease in reactor coolant flowrate is discussed in Reference 18. 812

15.0.3.1.2 CEA Ejection Analysis Method

The method used for analysis of the reactivity and power distribution anomalies initiated by a CEA ejection (Subsection 15.4.8) is documented in Reference 16, Topical Report CENPD-190-A, which was approved by the USNRC in Reference 5.

15.0.3.1.3 CESEC Computer Program

The CESEC III computer program is used to simulate the NSSS (unless specified otherwise for an event). The CESEC computer code is documented in Reference 26, and approved by the USNRC in Reference 28.

CESEC III computes key system parameters during a transient including core heat flux, pressures, temperatures, and valve actuations. A partial list of the dynamic functions included in this NSSS simulation includes point kinetics neutron behavior, Doppler and moderator reactivity feedback, boron and CEA reactivity effects, multi-node average reactor core thermal hydraulics, reactor coolant pressurization and mass transport, reactor coolant system safety valve behavior, steam generation, steam generator water level, turbine bypass, main steam safety and turbine admission valve behavior, as well as alarm, control, protection, and engineered safety features systems. The steam turbines, condensers and their associated controls are not included in the simulation. Steam generator feedwater enthalpy and flowrate are provided as input to CESEC III.

During the course of execution, CESEC III obtains steady-state and transient solutions to the set of equations that mathematically describe the physical models of the subsystems mentioned above. Simultaneous numerical integration of a set of nonlinear, first-order differential equations with time-varying coefficients is carried out by means of a simultaneous solution. As the time variable evolves, edits of the principal systems parameters are printed at prespecified intervals. An extensive library of the thermodynamic properties of uranium dioxide, water, and Zircaloy is incorporated into this program. Through the use of CESEC III symmetric and asymmetric plant response over a wide range of operating conditions can be determined.

The CESEC III code also explicitly models the steam void formation and collapse in the upper head region of the reactor vessel. Other improvements to this version of CESEC include a more detailed thermal-hydraulic model which explicitly simulates the mixing in the reactor vessel from asymmetric transients, an RCS flow model which calculates the time dependent reactor coolant mass flow rate in each loop, a wall heat model, 3D reactivity feedback model, a safety injection tank model, and a primary-to-secondary heat transfer model which calculates the heat transfer for each steam-generator node rather than for a steam generator as a whole.

15.0.3.1.4 COAST Computer Program

The COAST computer program is used to calculate the reactor coolant flow coastdown transient for any combination of active and inactive pumps and forward or reverse flow in hot or cold legs. The program is described in Reference 13. USNRC approval of the COAST computer code is provided in Reference 6.

The equations of conservation of momentum are written for each of the flow paths of the COAST model assuming unsteady one-dimensional flow of an incompressible fluid. The equation of conservation of mass is written for the appropriate nodal points. Pressure losses due to friction and geometric losses are assumed proportional to the flow velocity squared. Pump dynamics are modeled using a head-flow curve for a pump at full speed and using four-quadrant curves, which are parametric diagrams of pump head and torque on coordinates of speed versus flow, for a pump at other than full speed.

15.0.3.1.5 STRIKIN-II Computer Program

The STRIKIN-II computer program is used to simulate the heat conduction within reactor fuel rods and its associated surface heat transfer. The STRIKIN-II program is described in Reference 14. The STRIKIN-II computer code is approved by the USNRC in References 7 through 9.

The STRIKIN-II computer program provides a single or dual, closed-channel model of a core flow channel to calculate the clad and fuel temperatures for an average or hot fuel rod, and the extent of the zirconium water reaction for a cylindrical geometry fuel rod. STRIKIN-II includes

a. incorporation of all major reactivity feedback mechanisms,

b. a maximum of six delayed neutron groups,

c. both axial (maximum of 20) and radial (maximum of 20) segmentation of the fuel element, and

d. control rod scram initiation on high neutron power.

15.0.3.1.6 DNBR and Fuel Failure Methodology

The CETOP computer program is used to simulate the fluid conditions within the reactor core and to calculate fuel pin departure from nucleate boiling ratio (DNBR) for all events except the single RCP rotor seizure and RCP shaft break events. The CETOP program is described in Reference 27, and approved by the USNRC in Reference 29.

For the single RCP rotor seizure and RCP shaft break events, the TORC computer code (Reference 10) was used to calculate the fuel pin DNBR. USNRC approval of the TORC computer code is provided in Reference 11.

To determine whether fuel failure is predicted to occur for an event, the transient results are compared to the Specified Acceptable Fuel Design Limits (SAFDL). The two SAFDL of interest are the peak linear heat generation rate and the minimum DNBR. As described in Section 4.4, the DNBR SAFDL for the YGN 3&4 PLUS7 fuel is 1.21. If the minimum DNBR for an event is larger than 1.21, then fuel failure will not occur. If the minimum DNBR is less than

1.21, then fuel failure is conservatively assumed to occur.

A deterministic method is generally used to calculate the number of fuel pins which might experience DNB. For selected events, a statistical convolution method is used to calculate the number of fuel pins which might experience DNB, as approved by the USNRC in Reference 12. For both methods, all fuel pins which are predicted to experience DNB are conservatively assumed to fail. The deterministic method assumes that fuel rod failure would occur for all rods whose DNBR falls below the SAFDL. The statistical convolution method uses the probability of being in DNB at a given DNBR to more accurately predict the number of fuel rods that may fail. The statistical convolution method of predicting fuel failure was used for the single RCP rotor seizure, the RCP shaft break, and the CEA ejection events.

15.0.3.1.7 HRISE computer Program

The HRISE computer program is used to calculate the transient DNBR when the thermal-hydraulic conditions during a transient is out of the applicable range of KCE-1, which is critical heat flux correlation (Reference 19), which is used in TORC AND CETOP computer program. The HRISE program is described in Reference 40, and approved by the USNRC via Reference 41.

The HRISE program perfoms the thermal-hydraulic calculation using the closed channel model. The minimum transient DNBR of the post-trip steam line break can be calculated by the several critical heat flux correlation including Macbeth correlation, which is approved by the USNRC.

15.0.3.1.8 Reactor Physics Computer Programs

Numerous computer programs are used to produce the input reactor physics parameters required by the NSSS simulation and reactor core programs previously described. These reactor physics computer programs are described in Chapter 4.

15.0.3.2 Initial Conditions

The events discussed in this chapter have been analyzed over a range of initial values for the principal process variables. The ranges were chosen to encompass all steady-state operational configurations (with the exception of part loop operation).

Analysis over a range of initial conditions is compatible with the monitoring function performed by the COLSS which is described in Section 7.7 and the flexibility of plant operation which the COLSS allows. This flexibility is produced by allowing parameter trade-offs by monitoring the principal process variables, synthesizing the margin to fuel thermal design limits, and displaying to the reactor operator the core power operating limit. The required margin to DNB incorporated in COLSS is currently established by the non-LOCA events. The peak linear heat generation rate incorporated in COLSS is established by the loss-of-coolant accident (LOCA). The range of values of each of the principal process variables that was considered in analyses of events discussed in this chapter is listed in Table 15.0.3-1.

15.0.3.3 Input Parameters

The parameters used in the analyses are consistent with those listed in the preceding section and are primarily based on first-core values.

15.0.3.3.1 Doppler Coefficient

The effective fuel temperature coefficient of reactivity (Doppler Coefficient) used in the safety analyses bound the fuel reactivity feedback by accounting for uncertainties in determining the actual fuel temperature reactivity effects. An uncertainty of greater than 15% is applied in the conservative direction for each applicable event.

The effective fuel temperature correlation is discussed in Section 4.3. This correlation relates the effective fuel temperature, which is used to correlate Doppler reactivity, to the core power.

15.0.3.3.2 Moderator Temperature Coefficient

The events analyzed in this chapter model moderator reactivity as a function of moderator temperature instead of a moderator temperature coefficient. The moderator temperature coefficients corresponding to these moderator reactivity functions at nominal full power conditions (see Table 4.3-3) range from

0.0 to $-3.8 \times 10^{-4}\,\triangle\rho/°F$ (0 to $-6.84 \times 10^{-4}\,\triangle\rho/°C$). These values include all uncertainties, and bound the expected moderator temperature coefficients for all first cycle burnups, power levels, CEA configurations, and boron concentrations.

The most conservative, allowable value for the moderator temperature coefficient is assumed for each individual analysis.

15.0.3.3.3 Shutdown CEA Reactivity

The shutdown reactivity is dependent on the CEA worth available on reactor trip, the axial power distribution, the position of the regulating CEAs, and the time in core life. For transient analyses other than CEA ejection and steam line break, CEA worths of 8.0% $\triangle\rho$ and 5.5% $\triangle\rho$ were used for hot full power (HFP) and hot zero power (HZP), respectively. For CEA ejection events CEA worths of 6.0% $\triangle\rho$ and 6.0% $\triangle\rho$ were used for HFP and HZP, respectively. The foregoing values include uncertainties, the most reactive CEA stuck in the fully withdrawn position, and the effect of cooldown to HZP temperature conditions (Subsection 4.3.2.4.3).

For steam line break analyses at full power, a conservative CEA worth of 9.4% 812
$\triangle\rho$ was used. This value is appropriate for an end-of-cycle core, and includes uncertainties and the penalties appropriate to HFP. For steam line break events initiated from HZP, a conservative CEA worth of 6.0% $\triangle\rho$ was sufficient 812
to preclude significant post-trip return-to-power. This value covers uncertainties, the most reactive CEA stuck in the fully withdrawn position, and the penalties appropriate to HZP. The power dependent insertion limit (PDIL) included in the Technical Specifications assures that these worths are available upon reactor trip.

The shutdown reactivity worth versus position curve which is employed in the Chapter 15 analyses, except where noted in individual discussions of events.

is shown in Figure 15.0.3-1. This shutdown worth versus position curve was calculated assuming a more conservative rate of negative reactivity insertion than is expected to occur during the majority of operations, including power maneuvering. Accordingly, it is a conservative representation of shutdown reactivity insertion rates for reactor trips which occur as a result of the events analyzed.

15.0.3.3.4 Effective Delayed Neutron Fraction

The effective neutron lifetime and delayed neutron fraction are functions of fuel burnup. For each analysis, the values of the neutron lifetime and the delayed neutron fraction are selected consistent with the time in life analyzed.

15.0.3.3.5 Decay Heat Generation Rate

Analyses assume decay heat generation based upon an infinite reactor operation at the initial core power level identified for each event.

15.0.4 Radiological Consequences

Several of the events discussed are accompanied by the release of steam or liquid from the reactor coolant system or main steam system. The methodology and important input parameters used to assess the radiological consequences of these releases are discussed below.

The CESEC III computer code (described in Subsection 15.0.3.1.3), in combination with hand calculations, was used to determine the mass and energy releases as a function of time. The CESEC III computer code is used up to the time of operator action. The mass and energy releases subsequent to this time are hand-calculated assuming the Technical Specifications maximum cooldown rate that is consistent with achieving shutdown cooling entry conditions within the next six hours. These data are then used as input to the calculation of radiological release to the atmosphere for determining thyroid and whole body doses.

The assumptions used for calculating radiological releases due to events other than LOCA to the atmosphere are as follows:

a. The initial primary system activity level is based on the maximum activity in the reactor coolant allowed in the Technical Specification. This activity level corresponds to a concentration of 3.63×10^{-4} Ci/lbm (0.8 μCi/gm) dose equivalent I-131 and 0.14Ci/lbm (300μCi/gm) dose equivalent Xe-133. 812

b. The initial secondary system activity level is equal to 3.63×10^{-5} Ci/lbm (0.08 μCi/gm) dose equivalent I-131. 812

c. Primary-to-secondary steam generator tube leakage is included in the calculation of activity releases to atmosphere from the steam generators. The "technical specification leakage" discussed in the analyses of Chapter 15 is a 0.5 gpm (1.89 L/min) primary-to-secondary tube leak. 741

d. For the analysis of events for which Reference 25 requires consideration of "iodine spiking", the following are used:

1. For iodine spiking generated by the event, the iodine appearance rate is increased by a factor of 335 for SGTR and 500 for other accidents. 741

2. For an abnormally high iodine concentration due to a previous iodine spike, a reactor coolant activity of 2.17×10^{-2} Ci/lbm (48 μCi/gm) dose equivalent I-131 is assumed. 812

The dose at the site exclusion area boundary (EAB) is calculated as follows:

(a) Multiply the total primary system mass release by the primary system activity level and divide by the appropriate decontamination factor (DF). This gives the total number of dose equivalent I-131 (or Xe-133) curies released from the primary system. 424

(b) For the applicable secondary system releases, multiply the total secondary system mass release by the secondary system activity level and divide by the appropriate DF to obtain the equivalent I-131 (or Xe-133) curies released to the environment.

(c) The curies of dose equivalent I-131 released to the environment can be converted to a thyroid dose by multiplying by the following factors:

(1) Breathing rate = 3.50×10^{-4} m^3/sec (Reference 2) 741

(2) Atmospheric dispersion factor (x/Q) = 7.050×10^{-4} sec/m^3

(3) I-131 dose conversion factor = 1.08×10^{6} rem/Ci

Combining these parameters gives an effective EAB dose conversion factor equal to 0.267 rem/Ci. Thus the total thyroid dose is calculated by multiplying the total activity release (dose equivalent I-131 curies) by the effective dose conversion factor (0.267 rem/Ci). 741 741

(d) The curies of dose equivalent Xe-133 released to the environment can be converted to a whole-body dose by multiplying by the following factors:

(1) Atmospheric dispersion factor (x/Q) = 7.050×10^{-4} sec/m^3 741

(2) Xe-133 dose conversion factor = 5.77×10^{-3} rem-m^3/Ci-sec

The whole-body dose is calculated by multiplying the total activity release (dose equivalent Xe-133 curies) by the effective dose conversion factor (4.068×10^{-6} rem/Ci). 424 741

(e) Additional assumptions used in the determination of radiological releases to the atmosphere for certain events are as follows:

(1) For pipe breaks outside containment in piping connected to the reactor coolant system, the release to atmosphere accounts for the formation of steam resulting from depressurization of the reactor coolant.

(2) For pipe breaks or valve malfunctions outside containment in the main steam system which result in eventual dry-out of a steam generator, radioactive nuclides within the steam generator are assumed to be released to atmosphere with a DF equal to 1.

(3) For the portion of primary to secondary leakage that flashes to steam in steam generator(SG), the DF is assumed to be I for Iodine and for the unflashed portion of primary to secondary leakage, the DF is assumed to be 100 for Iodines. 424

Refer to Appendix 15E for dose model assumptions related to LOCA and other events.

Using onsite meteorological data for the period from January 2010 to December 2012, the atmospheric dispersion factors (x/Q) were conservatively determined and applied to offsite dose evaluation of LOCA and other events. The resultant offsite doses of LOCA and other events meet the acceptance criteria. 741 265 741

"Delete"

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TABLE 15.0.1-1 (Sh. 1 of 4)

SINGLE FAILURES

Safety and Electrical System

1. One Main Feedwater Isolation Valve fails to close (two valves exist in series)
2. One Main Feedwater Back-Flow Check Valve fails to close (two valves in series)
3. One Main Steam Isolation Valve fails to close
4. One Atmospheric Dump Valve fails to open
5. One Atmospheric Dump Valve fails to close
6. Failure of any one Auxiliary Feedwater Pump to start or Auxiliary Feedwater Valve to function
7. Failure of one High Pressure Safety Injection Pump
8. Failure of one Low Pressure Safety Injection Pump
9. deleted
10. Failure of one Emergency Diesel Generator to start, run, or load
11. deleted 424
12. Complete failure of Auxiliary Pressurizer Spray
13. Failure of one train of the Reactor Coolant Gas Vent System

Control Systems

1. FWCS power supply failure (Bus N1) resulting in the following conditions:
   a. Inability to monitor and control steam-generator level
   b. Inability to change Steam-Generator Economizer and Downcomer Valve positions
   c. Inability to modulate Feedwater Pump speed on demand (All pumps)

TABLE 15.0.1-1 (Sh. 2 of 4)

Control Systems (Cont'd)

2. SBCS power supply failure (Bus N1) resulting in the following conditions:

   a. Inability to accommodate a load rejection

   b. Loss of Steam Generator Bypass Valve modulation

   c. Inability to generate AWP demand signal

   d. Inability to generate AMI demand signal

   e. Loss of Turbine runback demand

3. SBCS power supply failure (Bus N2) resulting in the following conditions:

   a. Loss of Turbine runback demand

   b. Inability to generate AMI demand signal

   c. Inability to accommodate load rejection

   d. Loss of Steam Generator Bypass Valve modulation

   e. Inability to generate AWP demand signal

4. RPCS power supply failure (bus N1 or bus N2) resulting in the following condition: Inability to generate CEA drop demand and Turbine runback signals

5. Failure to modulate Turbine Bypass Valves open when required - one, two, four, six, or all valves

6. Failure to quick open Turbine Bypass Valves when required - one, four, six, or all valves

7. Excessive modulation of one or more Turbine Bypass Valves after a valid demand

8. Failure to generate AMI signal

9. Failure to prevent CEA withdrawal due to failure to AWP signal

10. Failure to modulate open all Turbine Bypass Valves failure to runback Turbine, and failure to drop selected subgroups of CEAs

TABLE 15.0.1-1 (Sh. 3 of 4)

Control Systems (Cont'd)

11. Failure to quick open all TBVs, failure to runback Turbine, and failure to drop selected subgroups of CEAs
12. Failure of CEA insertion demand signal
13. Failure of CEA withdrawal demand signal
14. High CEA insertion rate demand signal
15. High Turbine load index demand to SBCS
16. Low Turbine load index demand to SBCS
17. Failure of CEA insertion demand signal and failure to generate a Turbine runback demand signal
18. Failure to drop selected subgroups of CEAs
19. Failure to setback or runback Turbine power
20. Failure to generate reactor power cutback signal and failure to generate Turbine setback or runback signal
21. Excessive feedwater flow during post trip operation
22. Insufficient feedwater flow during post trip operation
23. Failure of reactor trip override to function
24. Failure of high level override to function during post trip operation.
25. Insufficient Pressurizer Spray flow
26. Insufficient Heater capacity (Backup Heaters only)
27. High T-avg signal to the PLCS, FWCS, SBCS, (At full power) resulting in the following conditions:
    a. Failure to withdraw CEAs when required
    b. Failure to runback the Turbine upon Reactor power cutback
    c. Excessive feedwater flow after reactor trip

TABLE 15.0.1-1 (Sh. 4 of 4)

Control Systems (Cont'd)

28. Excessive Pressurizer Spray flow (due to inadvertent actuation)

29. Insufficient Heater capacity, excessive Pressurizer Spray flow, and excessive modulation of one or more TBVs (Backup Heaters and excessive spray flow after actuation of the Sprays)

TABLE 15.0.1-2 (Sh. 1 of 3)

INITIATING EVENTS AND FREQUENCIES

FSAR SECTION NUMBER	EVENT DESCRIPTION	EVENT FREQUENCY
15.1	INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	
15.1.1	Decrease in Feedwater Temperature	MF(1)
15.1.2	Increase in Feedwater Flow	MF
15.1.3	Increased Main Steam Flow	MF
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve (+ Loss of Offsite Power)	MF (I)(2)
15.1.5	Steam System Piping Failures Inside and Outside Containment	Accident (3)
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	
15.2.1	Loss of External Load	MF
15.2.2	Turbine Trip	MF
15.2.3	Loss of Condenser Vacuum	MF
15.2.4	Main Steam Isolation Valve Closure	MF
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	MF
15.2.7	Loss of Normal Feedwater Flow	MF
15.2.8	Feedwater System Pipe Breaks	Accident
15.3	DECREASE IN REACTOR COOLANT FLOW RATE	
15.3.1	Total Loss of Reactor Coolant Flow	MF
15.3.3	Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power	Accident
15.3.4	Reactor Coolant Pump Shaft Break with Loss of Offsite Power	Accident

TABLE 15.0.1-2 (Sh. 2 of 3)

FSAR SECTION NUMBER	EVENT DESCRIPTION	EVENT FREQUENCY
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES	
15.4.1	Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low Power Condition	MF
15.4.2	Uncontrolled Control Element Assembly Withdrawal at Power	MF
15.4.3	Single Full Strength Control Element Assembly Drop	MF
15.4.4	Startup of an Inactive Reactor Coolant Loop	MF
15.4.6	Inadvertent Deboration	MF
15.4.7	Inadvertent Loading of a Fuel Assembly into the Improper Position	MF
15.4.8	Control Element Assembly (CEA) Ejection	Accident
15.5	INCREASE IN REACTOR COOLANT INVENTORY	
15.5.1	Inadvertent Operation of the ECCS	MF
15.5.2	CVCS Malfunction - Pressurizer Level Control System Malfunction with Loss of Offsite Power	MF
15.6	DECREASE IN REACTOR COOLANT SYSTEM INVENTORY	
15.6.1	Inadvertent Opening of a Pressurizer Safety/Relief Valve	Accident
15.6.2	Double-Ended Break of a Letdown Line Outside Containment	MF
15.6.3	Steam Generator Tube Rupture (with or without Loss of Offsite Power)	Accident
15.6.5	Loss-of-Coolant Accidents (LOCA)	Accident

TABLE 15.0.1-2 (Sh. 3 of 3)

FSAR SECTION NUMBER	EVENT DESCRIPTION	EVENT FREQUENCY
15.7	RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT	
15.7.1	Waste Gas System Failure	Accident
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	Accident
15.7.3	Postulated Radioactive Release due to Liquid-Containing Tank Failures	Accident
15.7.4	Fuel Handling Accident	Accident
15.7.5	Spent Fuel Cask Drop Accidents	Accident
15.8	ANTICIPATED TRANSIENTS WITHOUT SCRAM	N/A
APPENDIX 15 D	Steam Generator Tube Rupture with Loss of Offsite Power Plus Stuck Open ADV	Accident(4)

Note:
(1) MF : Moderate Frequency
(2) I : Infrequent Incident
(3) N/A : Not Applicable
(4) This is a special analysis which goes beyond the requirements of Reference 25 by including the additional failure of an ADV. Since this was done for Palo Verde, it is also included in the YGN 3&4 FSAR.

TABLE 15.0.1-3

CHAPTER 15 SUBSECTION DESIGNATION

Each subsection is identified as 15.W.X.Y. with trailing zeros omitted where:

W = 1 Increase in heat removal by the secondary system
2 Decrease in heat removal by the secondary system
3 Decrease in reactor coolant system flow rate
4 Reactivity and power distribution anomalies
5 Increase in reactor coolant inventory
6 Decrease in reactor coolant inventory
7 Radioactive release from a subsystem or component

X = 1, 2, etc. Event Title from Reference 1

Y = 1 Identification of event and causes
2 Sequence of events and system operations
3 Analysis of effects and consequences
4 Conclusions

TABLE 15.0.2-1
REACTOR PROTECTION SYSTEM TRIPS USED IN THE SAFETY ANALYSIS

Event	RPS	Analysis Setpoint [f]	Reactor Trip Delay Time [c]
Events not Mentioned Below	High Logarithmic Power Level	0.05%	550 ms
	Variable Overpower	14 or 116%[a]	550 ms
	High Pressurizer Pressure	2407 psia (169.2kg/cm^2 A)	550 ms
	Low Pressurizer Pressure	1705 psia (119.8 kg/cm^2 A)	550 ms
	Low Steam Generator Pressure	851 psia (59.8 kg/cm^2 A)	550 ms
	Low Steam Gneerator Water Leve	135% wide range[b]	6000 ms
	High Steam Generator Water Leve	195% narrow range[e]	550 ms
	Low DNBR [i]	1.21	100 ms
	High Local Power Density	21 kW/ft[d] (689 W/cm)	100 ms
	Steam Generator P Low Flow	80%[g]	(h)
Feedwater and Steamline Breaks	Variable Overpower	14 or 116%[a]	550 ms
	High Pressurizer Pressure	2460 psia (173 kg/cm^2 A)	550 ms
	Low Pressurizer Pressure	1555 psia (109 kg/cm^2 A)	550 ms
	Low Steam Generator Pressure	789 psia (55.3 kg/cm^2 A)	550 ms
	Low Steam Generator Water Level	28% wide range	600 ms
	High Steam Generator Water Level	95% narrow range[e]	550 ms
	Low DNBR [i]	1.21	100 ms
	High Local Power Density[j]	21 kW/ft[d](689 W/cm)	100 ms

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NOTE:

a. Band of 14%, ceiling of 116%, and rate of 15%/min. apply.
b. Percent of distance between the wide range instrument taps above the lower tap. See Chapter 5 for details
c. The reactor trip delay times are also discussed in Section 7.2.
d. Setpoint value is set below the value at which fuel centerline melting would occur. See Section 4.4.
e. Percent of distance between the narrow range instrument taps above the lower tap. See Chapter 5 for details.
f. Some Chapter 15 analyses assumed more conservative setpoints for specific events.
g. Percent of hot leg flow
h. 1.0 second from time of occurrence of low flow trip condition until the reactor trip breakers open.
i. This trip may be actuated based on several conditions including low DNBR, low RCP speed, and variable overpower as discussed in Section 7.2.
j. This trip may be actuated based on several conditions including High Local Power density and variable overpower trip as discussed in Section 7.2.

TABLE 15.0.3-1

INITIAL CONDITIONS

Parameter	Units	Range	
Core Power	% of 2815 MWt	0 - 102	
Radial 1-Pin Peaking Factor(with uncertainty)		1.3 - 1.7	
Axial Shape Index		$-0.3 \leq$ ASI* $\leq +0.3$	
Reactor Vessel Inlet Coolant Flow Rate	% of 330,000 gpm (1.24×10^6 L/min)	95 - 116	
Pressurizer Water Level	% distance between upper tap and lower tap above lower tap	26 to 60	
Core Inlet Coolant Temperature	°F(°C)	550 - 572** (287.8 - 300.0)	
Pressurizer Pressure	psia (kg/cm^2 A)	2130 - 2325 (149.8 - 163.4)	812
Steam Generator Water Level	% distance between upper tap and lower tap above lower tap	35 - 98 (wide range)	
Steam Generator Tube Plugging Rate	% of total tube number	0 - 8***	812

$$* \quad ASI = \frac{\text{power in lower half of core - power in upper half of core}}{\text{total core power}}$$ 446

** Additional restrictions were applied to : Minimum core inlet coolant temperature above 90% power equals 560°F (293.3°C) and above 30% power maximum core inlet coolant temperature equals 570°F (298.9°C) 812

*** Assume the limiting steam generator tube plugging rate for each event 741

TABLE 15.0.3-2 (Sh. 1 of 2)

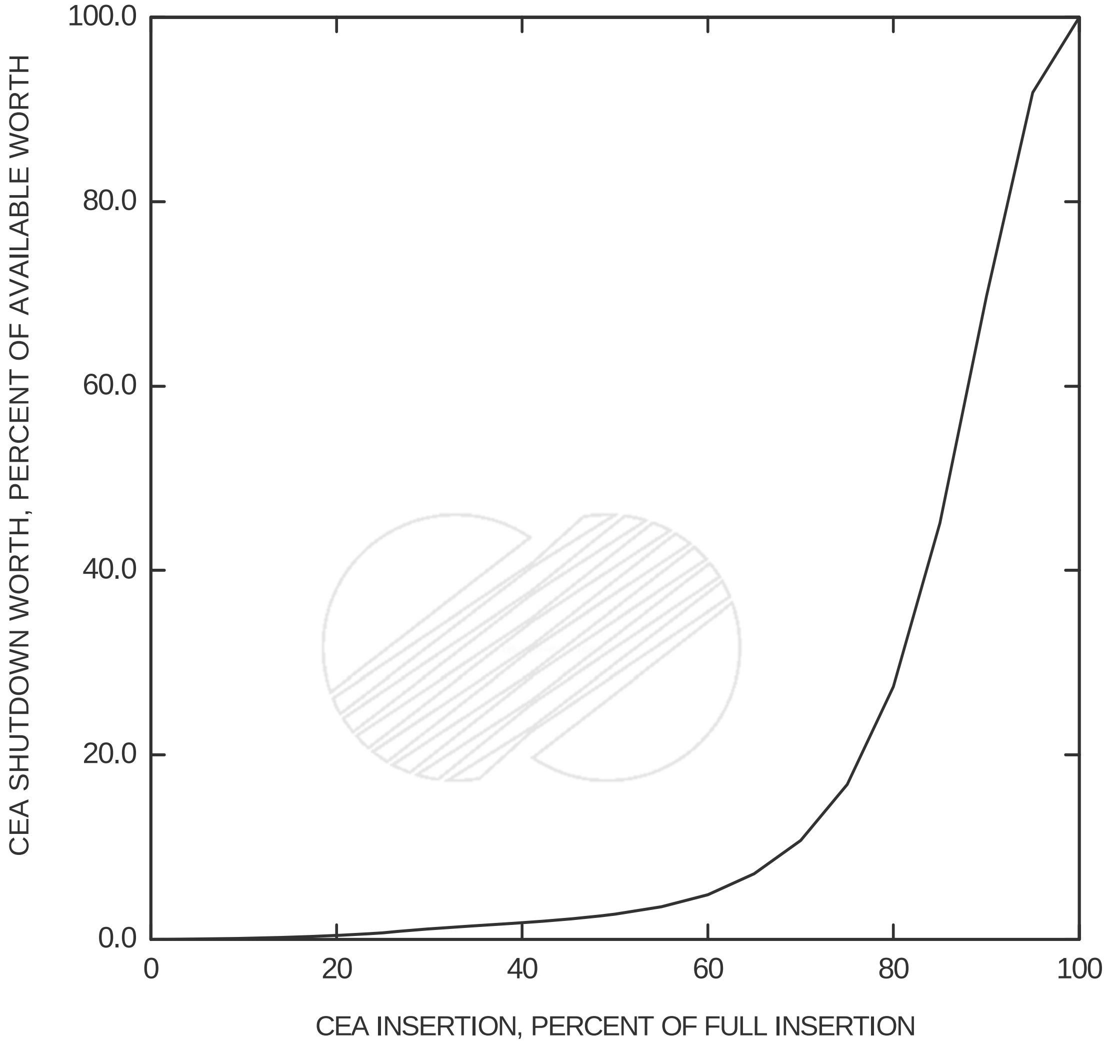
ANALYTICAL METHODS USED IN THE CHAPTER 15

SAFETY ANALYSES

Event	Reference for Event Methods	Reference for USNRC Approval of Methods
Inadvertent Opening of a Steam-Generator Relief or Safety Valve (Section 15.1)	Ref. 30	Ref. 12
Steam System piping Failures Inside and Outside Containment (Section 15.1)	Ref. 31	Ref. 32
Loss of Condenser Vacuum (Section 15.2)	Ref. 2	Ref. 21
Feedwater System Pipe Breaks (Appendix 15B)	Ref. 2	Ref. 21
Total Loss of Reactor Coolant Flow (Section 15.3)	Ref. 33	Ref. 18
Single Reactor Coolant Pump Rotor Seizure with Loss-of-Offsite Power (Section 15.3)	Ref. 2	Ref. 21
Uncontrolled Control Element Assembly Withdrawal and Single Control Element Assembly Drop (Section 15.4)	Ref. 2	Ref. 21
Inadvertent Deboration (Section 15.4)	Ref. 34	Ref. 35
Control Element Assembly Ejection (Section 15.4)	Ref. 2 & 16	Ref. 5 & 21
CVCS malfunction - Pressurizer Level Control System Malfunction with Loss-of-Offsite Power (Section 15.5)	Ref. 2	Ref. 21

TABLE 15.0.3-2 (Sh. 2 of 2)

Event	Reference for Event Methods	Reference for USNRC Approval of Methods
Double Ended Break of a letdown Line Outside Containment (Section 15.6)	Ref. 2	Ref. 21
Steam-Generator Tube Rupture (Section 15.6)	Ref. 2	Ref. 21
Steam-Generator Tube Rupture with a Loss-of-Offsite Power and Single Failure (Appendix 15D)	Ref. 36	Ref. 37



KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

CEA SHUTDOWN WORTH VS. POSITION

Figure 15.0.3-1

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.1.1 DECREASE IN FEEDWATER TEMPERATURE

15.1.1.1 Identification of Event and Causes

A decrease in feedwater temperature may result from a loss of high-pressure feedwater heaters. Loss of one of two high-pressure feedwater heater trains, which may result from tube failures or high water level in any feedwater heater, results in the loss of three of six high-pressure heaters. No other single failure would result in the loss of more than one heater train. The maximum feedwater enthalpy decrease due to a failure in the main feedwater system is 108 Btu/lbm (60 kcal/kg).

15.1.1.2 Sequence of Events and System Operations

A decrease in feedwater temperature causes a decrease in the temperature of the reactor coolant, an increase in reactor power due to the negative moderator temperature coefficient, and a decrease in the reactor coolant system (RCS) and steam generator pressures. Detection of these conditions is accomplished by the pressurizer and steam generator low-pressure alarms and the high linear power alarm. If the transient were to result in an approach to specified acceptable fuel design limits (SAFDL), trip signals generated by the core protection calculators (CPCs) would assure that low departure from nucleate boiling ratio (DNBR) or high local power density limits are not exceeded.

15.1.1.3 Analysis of Effects and Consequences

A comparison of core power shows that the core power increase for the decrease in feedwater temperature event is larger than that for the inadvertent opening of a steam-generator atmospheric dump valve (IOSGADV). However, a core power increase greater than that obtained for the IOSGADV would cause an immediate CPC reactor trip on low DNBR or high local power density which terminates the degradation in fuel performance. Therefore, the systems operation described above and the resulting sequence of events would produce a transient DNBR equal to or less adverse than that associated with the IOSGADV event presented in Subsection 15.1.4. A loss of offsite power (LOOP) is assumed to occur in 3 second after turbine trip as a basic assumption. This event "Delete" results in an event similar to and bounded by the IOSGADV event "Delete" which is also presented in Subsection 15.1.4.

All increased heat removal events analyzed in this section are characterized by decreasing RCS pressure due to the cooldown of the primary system. Thus, this event, or this event plus a single failure, will result in an insignificant increase in RCS pressure.

15.1.1.4 Conclusions

The decreased feedwater temperature event results in a DNBR greater than 1.21 throughout the transient. "Delete" The RCS pressure remains well below 2750 psia (193.33 kg/㎠A), and the steam generator pressure remains well below 1397 psia (98.2 kg/㎠A).

15.1.2 INCREASE IN FEEDWATER FLOW

15.1.2.1 Identification of Event and Causes

An increase in feedwater flow is caused by the further opening of a feedwater control valve or an increase in feedwater pump speed. The maximum increase at full power does not exceed 40% above nominal for the main feedwater system.

15.1.2.2 Sequence of Events and System Operations

An increase in feedwater flow causes a decrease in the temperature of the reactor coolant, an increase in reactor power due to the negative moderator temperature coefficient, a decrease in the RCS and steam-generator pressures and an increase in steam-generator water level. Detection of these conditions is accomplished by the pressurizer low-pressure alarm and steam generator low pressure and high water level alarms. Protection against the violation of specified acceptable fuel design limits (SAFDL), as a consequence of an increase in feedwater flow, is provided by the CPC low DNBR and high local power density trips. Protection against high steam-generator water level is provided by the high steam-generator water level trip.

15.1.2.3 Analysis of Effects and Consequences

A comparison of core power shows that the core power increase for the increase in feedwater flow event is greater than that for the inadvertent opening of a steam-generator atmospheric dump valve (IOSGADV) event.

However, a core power increase greater than that obtained for the IOSGADV would cause an immediate CPC reactor trip on low DNBR or high local power density which terminates the degradation in fuel performance.

If the power increase was identical to that assumed in the IOSGADV, the transient fuel performance thermal hydraulic results would be identical to that assumed for the IOSGADV. However, the transient would not be able to achieve a new stabilized condition due to the fact that a reactor trip would occur rapidly on a high steam-generator level. Therefore, the systems operation described above and the resulting sequence of events would produce a DNBR transient no more adverse than that associated with the IOSGADV event presented in Subsection 15.1.4. A loss of offsite power (LOOP) is assumed to occur in 3 seconds after tubine trip as a basic assumption. This event

"Delete" results in an event similar to and bounded by the IOSGADV event

"Delete" which is also presented in Subsection 15.1.4.

All increased heat removal events analyzed in this section are characterized by decreasing RCS pressure due to the primary system cooldown. Thus, this event, or this event plus a single failure, will result in an insignificant increase in RCS pressure.

15.1.2.4 Conclusions

The increased feedwater flow event results in a DNBR greater than 1.21 throughout the transient. "Delete" The RCS pressure remains below 2750 psia (193.33 kg/㎠A) and the steam-generator pressure remains below 1397 psia (98.2 kg/㎠A).

15.1.3 INCREASED MAIN STEAM FLOW

15.1.3.1 Identification of Event and Causes

An increase in main steam flow is caused by an inadvertent increased opening of the turbine stop valves. This may be caused by operator error or turbine load limit malfunctions and will result in no more than an 11% increase over the nominal full power steam flow rate. An increase in main steam flow can also result from the inadvertent opening of a turbine bypass valve or an atmospheric dump valve; however, these events are discussed separately in Subsection 15.1.4.

15.1.3.2 Sequence of Events and System Operations

An increase in main steam flow causes a decrease in the temperature of the reactor coolant, an increase in core power and heat flux, and a decrease in reactor coolant system and steam-generator pressures. Detection of these conditions is accomplished by the pressurizer and steam generator low pressure alarms and the high reactor power alarm. If the transient were to result in an approach to specified acceptable fuel design limits, trip signals generated by the core protection calculators would assure that low departure from nucleate boiling ratio (DNBR) or high local power density limits are not exceeded.

15.1.3.3 Analysis of Effects and Consequences

A comparison of core power shows that the core power increase for the increased main steam flow event is identical to that for the inadvertent opening of a steam-generator atmospheric dump valve (IOSGADV) event. This is due to the fact that both events cause an increase in main steam flow of 11%. Thus, the subsequent DNBR transient are also identical. Therefore, the systems operation described above and the resulting sequence of events for the increased main steam flow event will be similar to the IOSGADV event presented in Subsection 15.1.4.

A loss of offsite power (LOOP) is assumed to occur in 3 seconds after tubine trip as a basic assumption. This event " Delete" is similar to and bounded by the IOSGADV " Delete" event which is also presented in Subsection 15.1.4.

All increased heat removal events analyzed in this section are characterized by decreasing RCS pressure due to the cooldown of the primary system. Thus, this event, or this event plus a single failure, will show an insignificant increase in RCS pressure.

15.1.3.4 Conclusions

The increased main steam flow event results in a DNBR greater than 1.21 throughout the transient. " Delete" The RCS pressure remains well below 2750 psia (193.33 kg/cm² A), and the steam generator pressure remains well below 1397 psia (98.2 kg/cm² A).

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

15.1.4.1 Identification of Event and Causes

Case 1: Inadvertent Opening of Steam-Generator Atmospheric Dump Valve

An atmospheric dump valve (ADV) or a turbine bypass valve may be inadvertently opened by the operator or may open due to a failure of the control system which operates the valve. A main steam safety valve will remain open only as a result of a valve failure. The opening of any of these valves will result in similar consequences because they relieve steam at the same maximum flow rate (11% of full power turbine flow rate). The inadvertent opening of a steam-generator atmospheric dump valve (IOSGADV) is presented here to illustrate these events. A loss of offsite power (LOOP) is considered as a basic assumption rather than a single failure for IOSGADV.

Case 2: Inadvertent Opening of a Steam-Generator Atmospheric Dump Valve Plus a Single Failure (IOSGADV + SF)

For the events of this section, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a fuel design limit has been violated and thus whether fuel cladding degradation might be anticipated.

Those factors which cause a decrease in local DNBR are as follows:

a. Increasing local core heat flux (including radial and axial power distribution effects)

b. Decreasing reactor coolant flow

c. Decreasing reactor coolant pressure

d. Increasing reactor coolant temperature

The single failure (SF) which yields the minimum transient hot channel DNBR is the SF which combines the greatest decrease in DNBR after initiation of a reactor trip signal with the lowest possible pre-trip DNBR. An evaluation of the SFs listed in Table 15.0.1-1 shows that the limiting SF for the event of this section is an excessive feedwater supply after the turbine trip following a reactor trip. An increase in the feedwater flow after the reactor trip can be caused by a malfunction in the feedwater system. Since this event is assumed to be initiated at a LCO condition, the DNBR starts to decrease during the initial phase of the event due to decreasing RCS pressure and increasing core power, then reaches to a almost constant value remains there after the primary system reaches to a steady state. In normal cases the feedwater flow in rapidly reduced after reactor trip. However, if an excessive feedwater is supplied after reactor trip due to a SF in the feedwater control system, it will adversely affect the MDNBR before the core power is reduced. Any other SFs have no significant impact on the MDNBR after the inition of reactor trip. Therefore, the limiting event with a SF in this section is the inadvertant opening of a steam generator atmospheric dump valve accompanied by a single failure which results in an excessive feedwater flow after reactor trip. In addition to the assumed single failure, it is assumed that the most reactive CEA is held in the fully withdrawn position following reactor trip. A loop is considered to occur in 3 seconds after turbine trip as a basic assumption. 424

15.1.4.2 Sequence of Events and System Operations

Case 1: Inadvertent Opening of a Steam-Generator Atmospheric Dump Valve (IOSGADV)

The opening of a steam generator ADV increases the rate of heat removal by the steam generators, causing cooldown of the RCS. Due to the negative moderator temperature coefficient, core power increases from the initial value of 102% of rated core power, reaching a new stabilized value of 113%. The feedwater control system, which is assumed to be in the automatic mode, supplies feedwater to the steam generators such that steam-generator water levels are maintained. Acting upon the large power mismatch between the reactor and turbine and the audible indication of steam blowdown, the reactor operator recognizes that the plant is in an abnormal state and manually trips the reactor. The analysis presented herein assumes this initial operator action is delayed until after 30 minutes following the first indication of the event.

Following the generation of a turbine trip on reactor trip, feedwater flow to the steam generators is assumed to be terminated. Since the steam bypass control system is assumed to be in the manual mode with all bypass valves closed, the main steam safety valves (MSSVs) open to limit main steam system pressure and remove heat stored in the core and RCS. The main steam system pressure then decreases due to the cooldown caused by flow through the MSSVs and the ADV, and the MSSVs close. The main steam system pressure continues to decrease to the point where a main steam isolation signal (MSIS) is generated. This causes one steam generator to be isolated from the flow path through the open ADV. The affected steam generator continues to blow down and the level falls below the auxiliary feedwater actuation signal (AFAS) setpoint. However, the AFAS logic, acting upon the fact that the pressure in the affected steam generator is much lower than in the intact steam generator, prevents actuation of auxiliary feedwater flow to the affected steam generator. As a result, the affected steam generator eventually boils dry. During the period of blowdown following reactor trip, reactor coolant temperatures and pressure decrease slowly. After dryout of the affected steam generator, decay heat and heat addition from the walls and structure of the primary coolant system cause a gradual increase in reactor coolant temperatures and pressure. Relief of steam by the safety valves on the unaffected steam generator provides cooling which limits reactor coolant temperatures. Reactor coolant pressure is limited by the pressurizer safety valves.

Subsequent to tripping the reactor, the operator manually closes the ADV which had been inadvertently opened, terminating steam release to the atmosphere from the affected steam generator. In the analysis presented herein, it is conservatively assumed that this action to close the ADV is delayed 20 minutes beyond the operator's initial action to trip the reactor, or a total of 50 minutes after event initiation. RCS heat removal for plant stabilization and cooldown is accomplished by using the turbine bypass valves. The operator is assumed to initiate plant cooldown 30 minutes after he manually trips the reactor.

Case 2: Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Single Failure (IOSGADV + SF)

Up until the time of the assumed reactor trip, the transient due to the IOSGADV is identical with or without the single failure. For the IOSGADV + SF event the reactor is assumed to trip manually at 1800 seconds into the transient. After reactor and turbine trip, the feedwater flow is not reduced because of the assumed single failure. This is a very conservative assumption, since a LOOP will terminate the normal feedwater flow to steam generators. Since the steam bypass control system is assumed to be in the manual mode with all bypass valves closed, the MSSVs open to limit main steam system pressure and remove heat stored in the core and RCS. The main steam system pressure then decreases, due to the cooldown caused by the flow through the MSSVs and the ADV, and the MSSVs close. The main steam system pressure continues to decrease to the point where an MSIS is generated. This causes one steam generator to be isolated from the flow path through the open ADV. The affected steam generator continues to blow down, and the level falls below the AFAS setpoint. However, the AFAS logic, acting upon the fact that the pressure in the affected steam generator is much lower than that in the intact steam generator prevents actuation of auxiliary feedwater flow to the affected steam generator. As a result, the affected steam generator eventually boils dry. During the period of blowdown following reactor trip, reactor coolant temperatures and pressure decrease slowly. After dryout of the affected steam

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generator, decay heat and heat addition from the walls and structure of the primary coolant system cause a gradual increase in reactor coolant temperatures and pressure. Relief of steam by the safety valves on the unaffected steam generator provides cooling which in turn maintains natural circulation flow through the core and limits reactor coolant temperatures.

Acting upon a variety of indications--including the initial large power mismatch between the reactor and turbine, the steady decrease in steam generator pressures and water levels after reactor trip, the continued decrease in pressure and level in the affected steam generator after MSIS, the low steam generator pressure and water level alarms, and the audible indication of steam blowdown--the reactor operator diagnoses the incident and manually closes the ADV which had been inadvertently opened, terminating steam release to the atmosphere from the affected steam generator. The analysis presented herein assumes that this initial operator action to close the opened ADV is delayed until 50 minutes. RCS heat removal for plant stabilization and cooldown is accomplished by manual control of the ADVs on the unaffected steam generator. The operator is assumed to initiate plant cooldown 10 minutes after he manually closes the ADV which had been inadvertently opened.

15.1.4.3 Analysis of Effects and Consequences

a. Mathematical Model

The nuclear steam supply steam (NSSS) response to the IOSGADV and the IOSGADV + SF was simulated using the CESEC-III computer program described in Subsection 15.0.3. The time-dependent thermal margin on DNBR in the reactor core was calculated using the CETOP computer program which uses the KCE-1 critical heat flux correlation described in Chapter 4.

b. Input Parameters and Initial Conditions

Table 15.1.4-1 lists the assumptions and initial conditions used for these analyses in addition to those discussed in Section 15.0. Conditions were chosen such that the overpower condition caused by the increase in steam flow results in the closest approach to the specified acceptable fuel design limits (SAFDL) without causing a reactor trip. If core power increases more than the 11% due to the increasing steam flow, the core protection calculators (CPC) will initiate a reactor trip and there will be no further degradation in thermal margin. For transients initiated at other sets of initial conditions, a trip may or may not be required depending on whether the initial thermal margin is as low as for the combination of conditions used in these analyses.

c. Results

Case 1: Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOSGADV)

The dynamic behavior of the salient NSSS parameters following the IOSGADV is presented in Figures 15.1.4-1 to 15.1.4-16. Table 15.1.4-2 summarizes the major events, times, and results for this transient.

The opening of an ADV increases the rate of heat removal by the steam generators causing cooldown of the RCS. Due to the negative moderator temperature coefficient, core power increases from 102% of rated core power, reaching a new, stabilized value of 113% after approximately 100 seconds. The feedwater control system, which is assumed to be in the automatic mode, supplies feedwater to the steam generators such that the steam-generator water levels are maintained.

During the IOSGADV transient, the minimum transient DNBR rapidly decrease at the initial stage of the event, and remains above 1.21 until the operator manually trips the reactor at 1800 sec. At 1800.1 seconds, the trip breakers open. At this point, both the local and core average power decrease rapidly and DNBR increases. The MSSVs don't open during the event. 741

At 2187.05 seconds, the steam-generator pressure reaches the MSIS setpoint of 851 psia (59.83 kg/cm^{2}A). The MSIS initiates closure of the MSIVs and MFIVs at 2188.20 seconds. The MSIVs and MFIVs close by 2193.20 seconds and 2198.20 seconds, respectively. The affected steam generator dries out at about 3000 seconds. At 3000 seconds, the operator manually closes the open ADV. The operator initiates plant cooldown at 3600 seconds. 741 741 424

Case 2: Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Single Failure (IOSGADV + SF)

The dynamic behavior of the salient NSSS parameters following IOSGADV with a single failure is presented in Figures 15.1.4-17 to 15.1.4-32. Table 15.1.4-3 summarizes the major events, times, and results for this transient.

The opening of an ADV increases the rate of heat removal by the steam generators causing cooldown of the RCS. Due to the negative moderator temperature coefficient, core power increases from 102% of rated core power, reaching a new, stabilized value of 113% after approximately 100 seconds. The reactor is tripped manually at 1800 seconds. The reactor trip breakers open at 1800.1 seconds. A LOOP is assumed 3 seconds after the turbine trip which is caused by the reactor trip.

During the IOSGADV + SF transient, the minimum transient DNBR rapidly decrease at the initial stage of event, and remains above 1.21 until the operator manually trips the reactor at 1800 sec.
The reactor coolant pumps begin to coast down following the loss of offsite power. As a result of the conservative assumption of the single failure, the main feedwater flow does not cutback after the reactor trip. The MSSVs don't open during the event.

At 2122.45 seconds, the steam generator pressure drops below the MSIS setpoint of 851 psia(59.83 kg/cm^2A). The MSIS initiates closure of the MSIV and MFIVs at 2123.60. The MSIVs and the MFIVs were closed by 2128.60 seconds and 2133.60 812
seconds, respectively.

Voids begin to form in the upper head of the reactor vessel at 2242.15 seconds.

At 3000 seconds, the operator manually closes the open ADV. The operator initiates plant cooldown at 3600 seconds.

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15.1.4.4 Conclusions

For the IOSGADV and IOSGADV with single failure events, the minimum DNBR remains above 1.21. For both cases, the pressurizer safety valves are not challenged as the RCS pressure "Delete" remains below 2,400 psia (168.72 kg/cm^{2}A) throughout the event. Thus, the RCS pressure remains well below 2750 psia (193.33 kg/cm^{2}A), ensuring that the integrity of the RCS is maintained, and the steam-generator pressure remains well below 1397 psia (98.2 kg/cm^{2}A). 424

"Delete"

TABLE 15.1.4-1

ASSUMPTIONS AND INITIAL CONDITIONS FOR FULL POWER
INADVERTENT OPENING OF ATMOSPHERIC DUMP VALVE
(IOSGADV AND IOSGADV + SF)

Parameter	Value	
Initial Core Power Level, MWt	2871.3	
Initial Core Inlet Coolant Temperature, °F (°C)	570 (298.89)	
Initial Core Mass Flow rate, 10^6 lbm/hr (10^6 kg/hr)	138 (62.59)	
Initial Pressurizer Pressure, psia (kg/cm^{2}A)	2325 (163.45)	
Initial Pressurizer Water Volume, ft^3 (m^3)	1033 (29.25)	
Initial Steam Generator Pressure, psia (kg/cm^{2}A)	1141.8 (80.28)	812
Initial Steam Generator Inventory, lbm (kg) per SG	198,000 (89.811)	812
CEA Worth on Trip, 10^{-2} △ρ	-8.0	
Core Burnup	End of cycle	
ASI	0.3	
Max. Radial Peaking Factor	2.3157	812

TABLE 15.1.4-2

SEQUENCE OF EVENTS FOR FULL POWER
INADVERTENT OPENING OF A STEAM-GENERATOR
ATMOSPHERIC DUMP VALVE (IOSGADV)

Time (sec)	Event	Setpoint or Valve	
0.0	One atmospheric dump valve opens fully	--	424
1800.0	Manual trip	--	
1800.1	MDNBR achieved	> 1.21	
1803.1	Loss of Offsite power/RCPs begin to coastdown	--	
2191.75	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, psia(kg/cm^2A)	851 (59.83)	812
2197.90	MSIVs close completely	--	
2202.90	MFIVs close completely	--	
2306.35	Voids begin to form in RV upper head	--	
3000	Operator manually closes ADV	--	424
3600	Operator initiates plant cooldown	--	

TABLE 15.1.4-3

SEQUENCE OF EVENTS FOR FULL POWER INADVERTENT OPENING OF A STEAM-GENERATOR ATMOSPHERIC DUMP VALVE WITH SINGLE FAILURE AFTER TURBINE TRIP (IOSGADV+SF)

Time (sec)	Event	Setpoint or Valve	
0.0	One atmospheric dump valve opens fully	--	424
1800.0	Manual trip	--	
1800.1	MDNBR achieved	> 1.21	
1803.1	Loss of Offsite power/RCPs begin to coastdown	--	
2122.45	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, psia(kg/cm^{2}A)	851 (59.83)	812
2128.60	MSIVs close completely	--	
2133.60	MFIVs close completely	--	
2242.15	Voids begin to form in RV upper head	--	
3000	Operator manually closes ADV	--	424
3600	Operator initiates plant cooldown	--	

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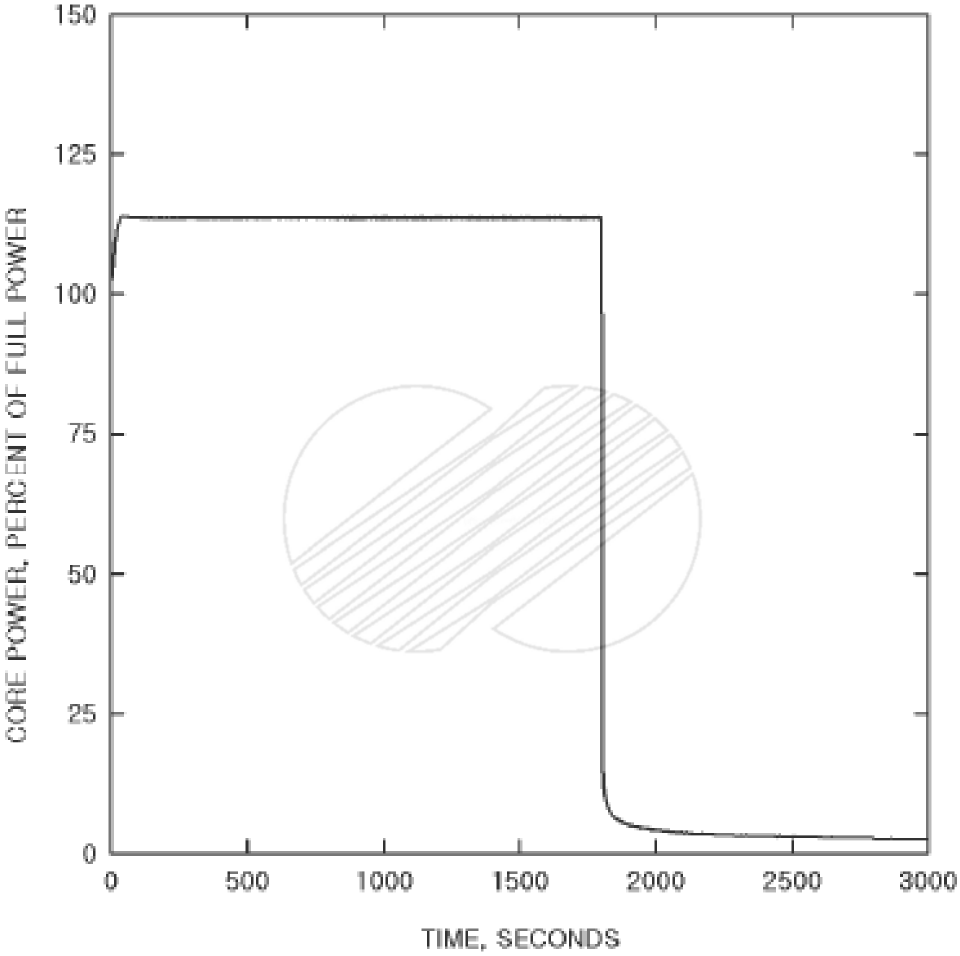

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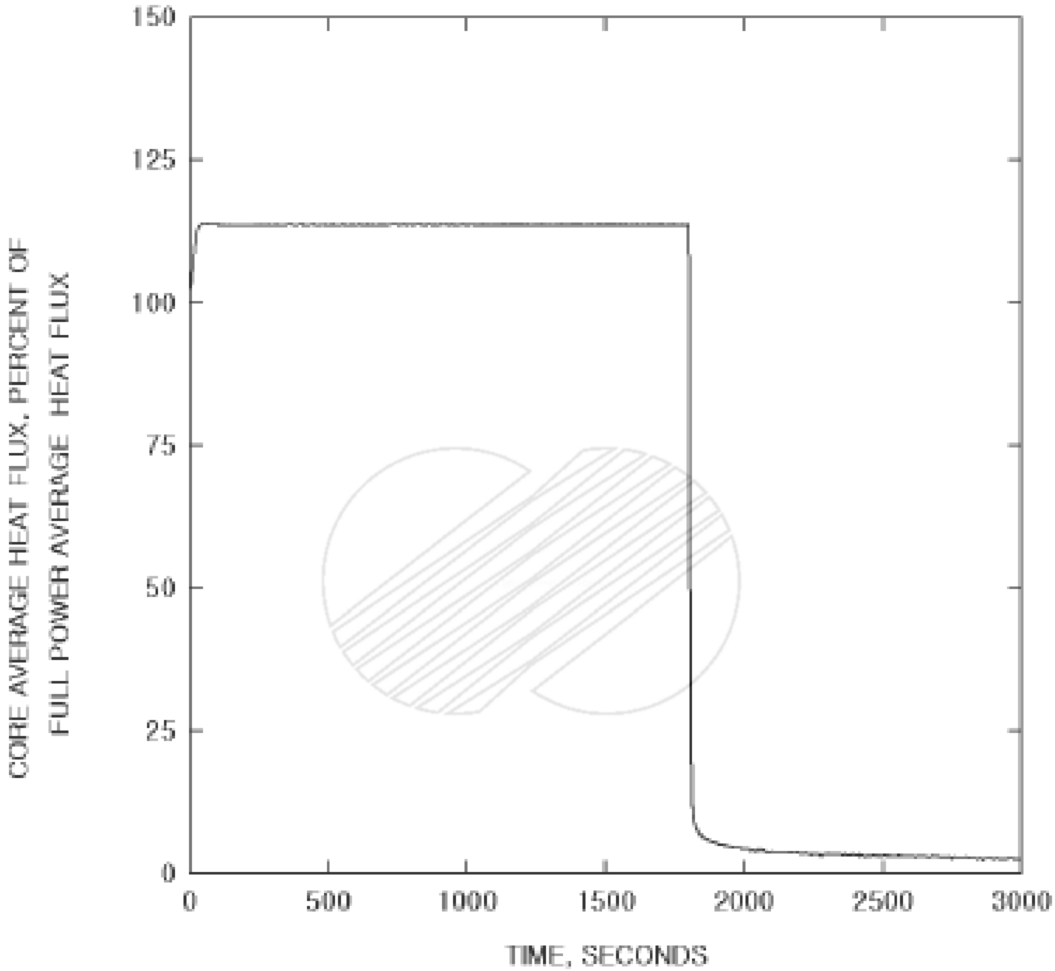
INADVERTENT OPENING OF AN ATMOSPHERIC DUMP VALVE : CORE POWER VS. TIME

Figure 15.1.4-1



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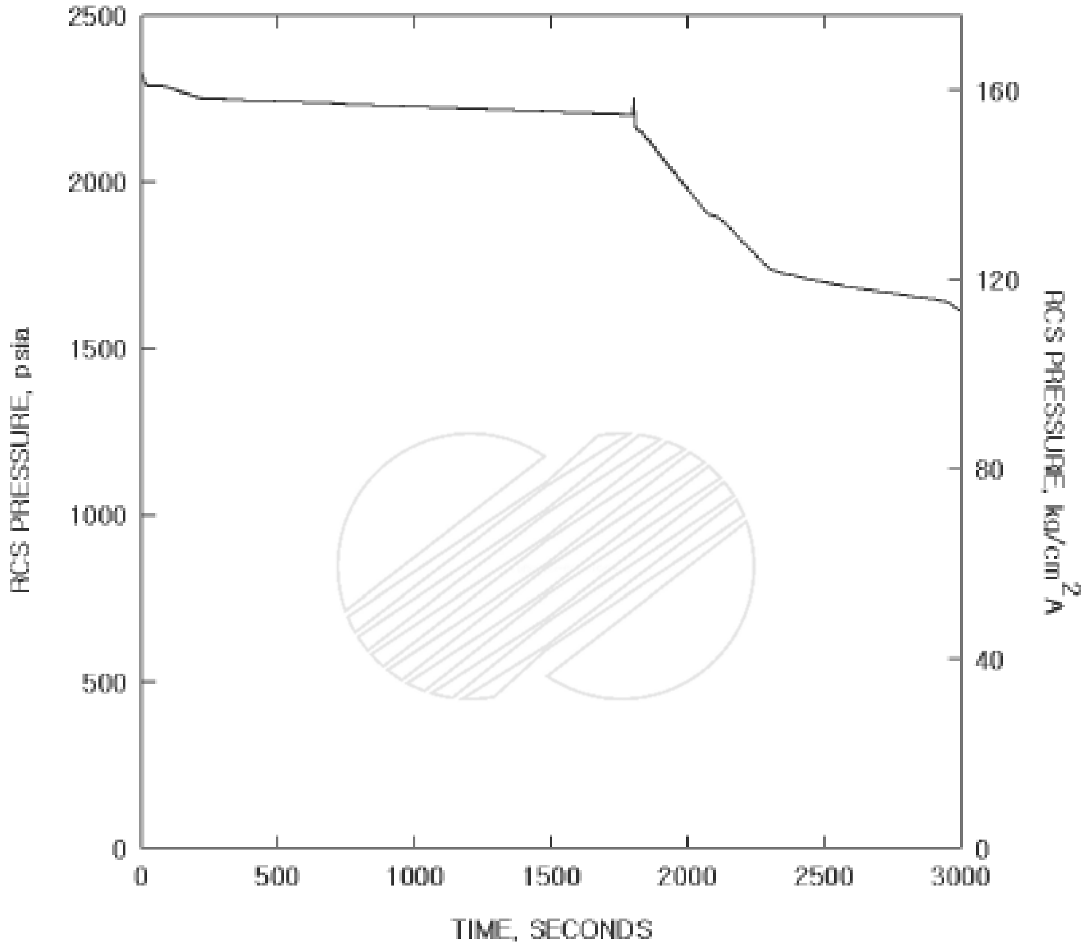
INADVERTENT OPENING OF AN ATMOSPHERIC DUMP VALVE : CORE AVERAGE HEAT FLUX VS. TIME

Figure 15.1.4-2



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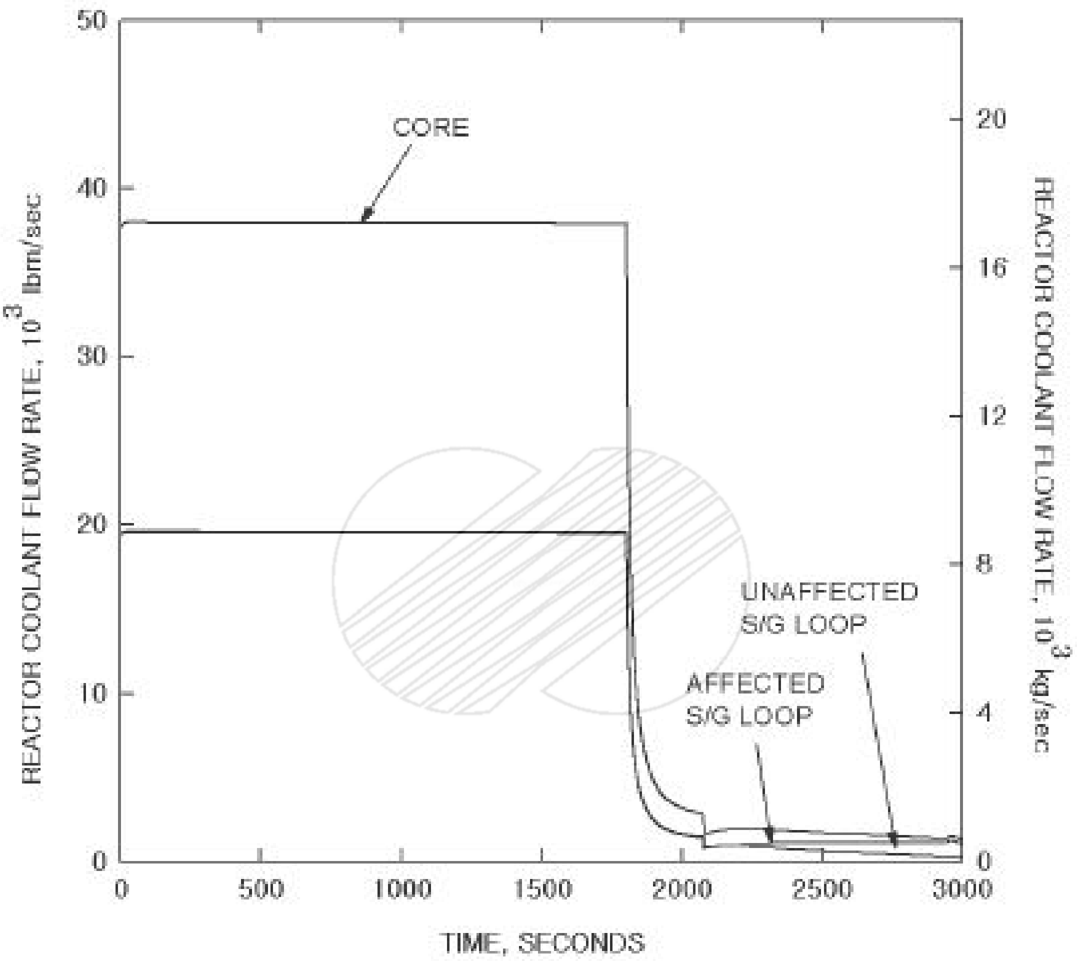
INADVERTENT OPENING OF AN ATMOSPHERIC DUMP VALVE : RCS PRESSURE VS. TIME

Figure 15.1.4-3



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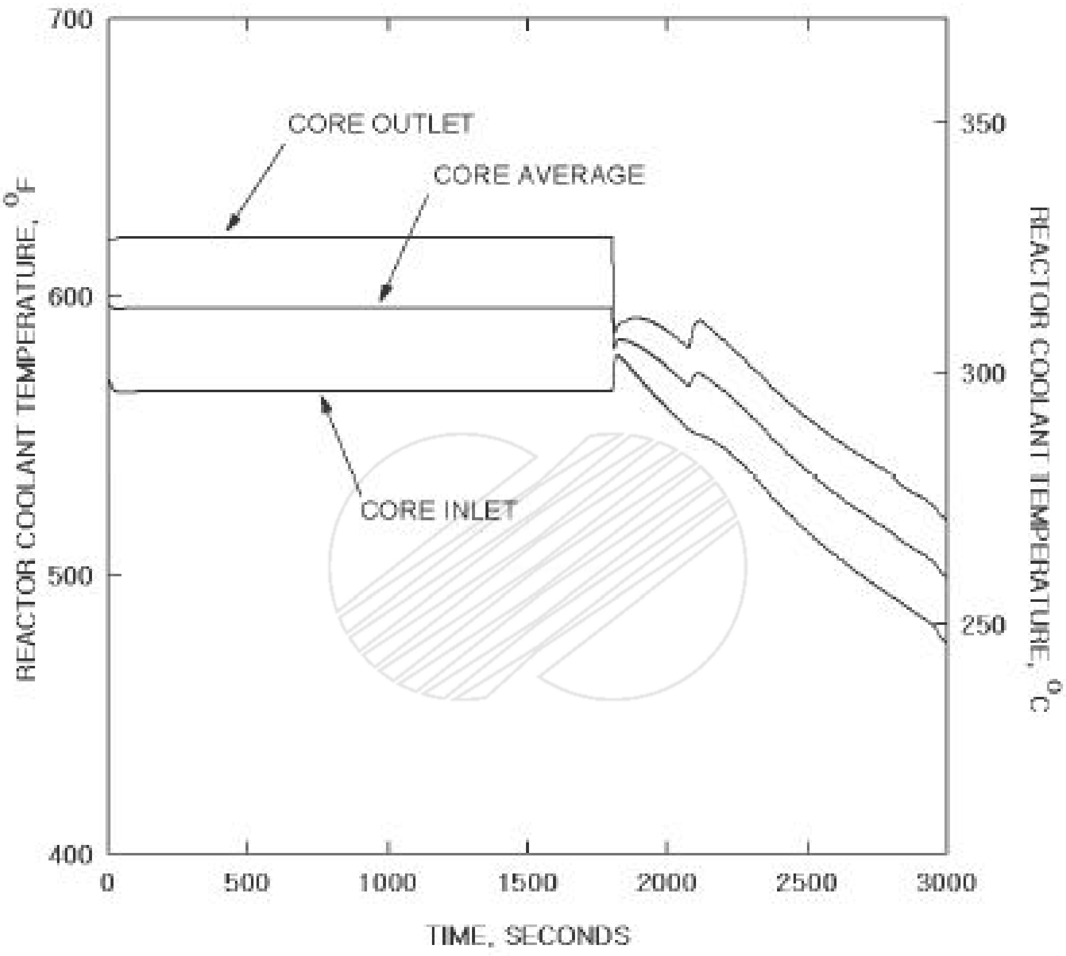
INADVERTENT OPENING OF AN ATMOSPHERIC DUMP VALVE : REACTOR COOLANT FLOW RATE VS. TIME

Figure 15.1.4-4



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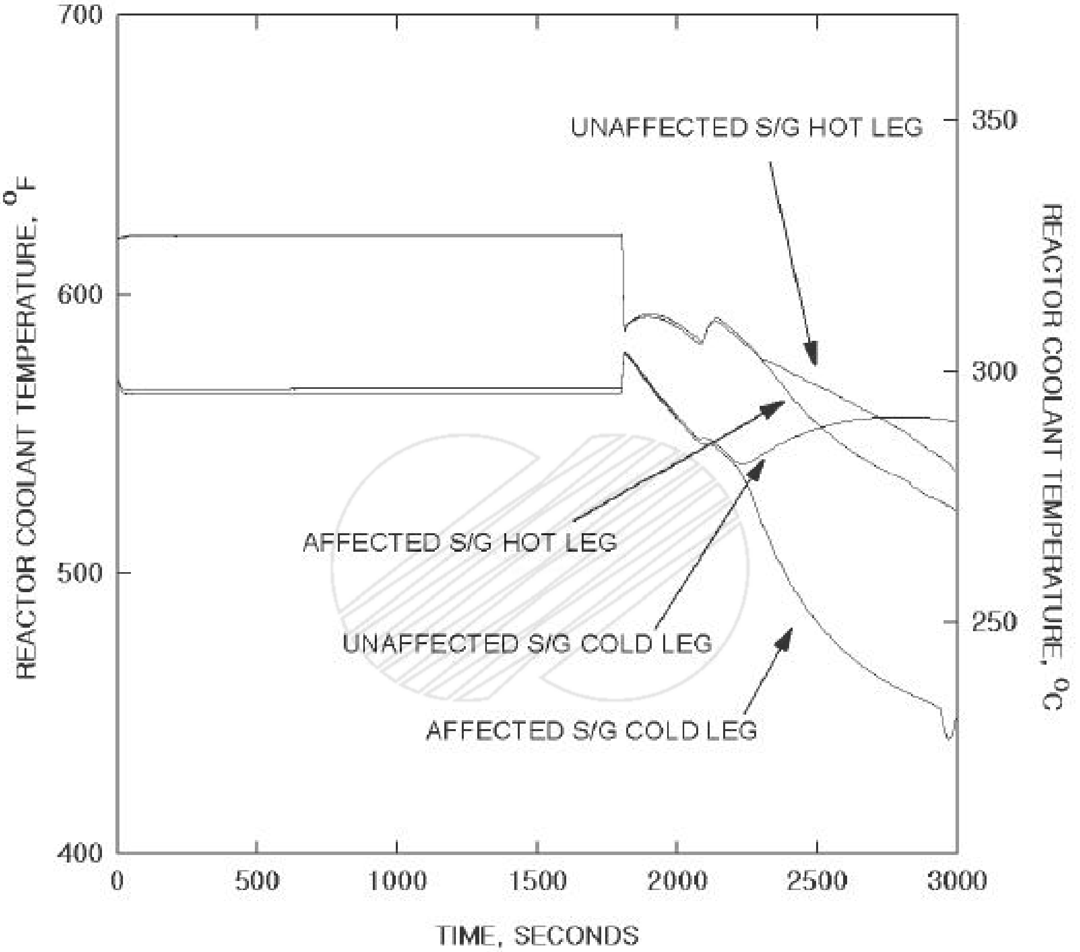
INADVERTENT OPENING OF AN
ATMOSPHERIC DUMP VALVE :
REACTOR COOLANT TEMPERATURE (A) VS. TIME

Figure 15.1.4-5



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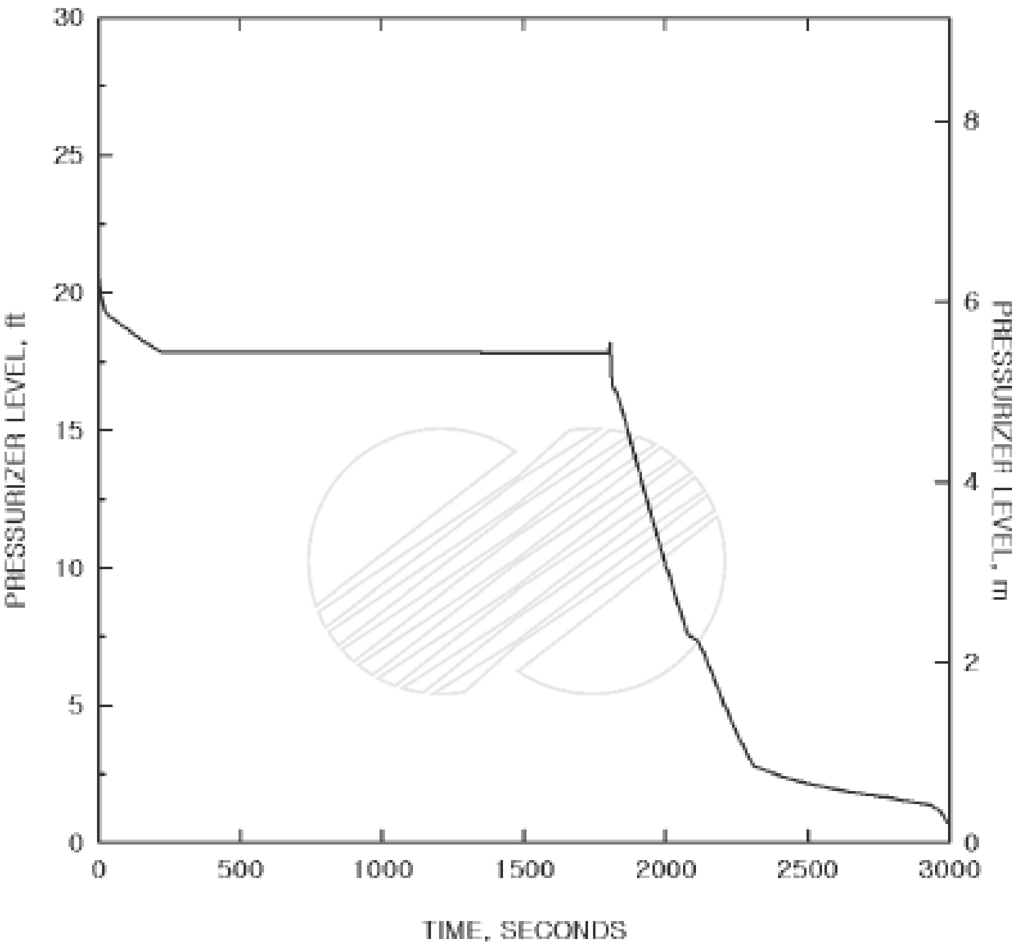
INADVERTENT OPENING OF AN
ATMOSPHERIC DUMP VALVE :
REACTOR COOLANT TEMPERATURE (B) VS. TIME

Figure 15.1.4-6



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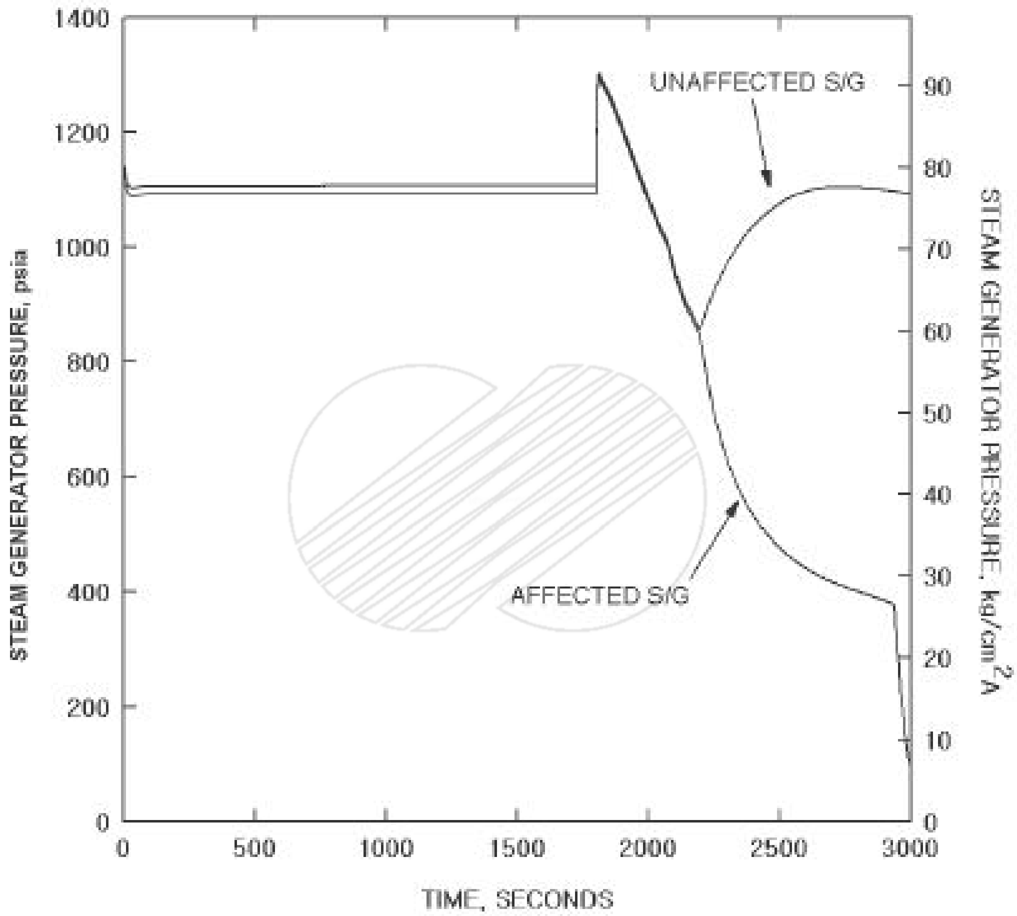
INADVERTENT OPENING OF AN ATMOSPHERIC DUMP VALVE : PRESSURIZER WATER LEVEL VS. TIME

Figure 15.1.4-7



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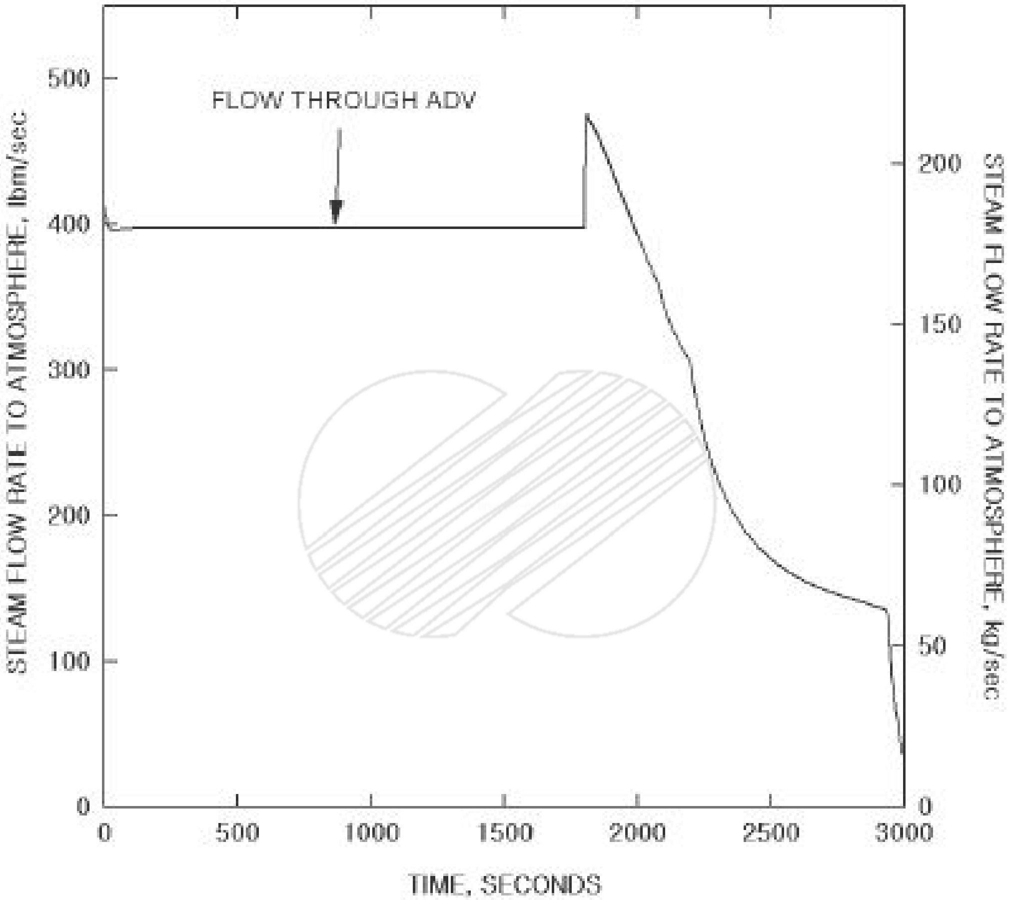
INADVERTENT OPENING OF AN ATMOSPHERIC DUMP VALVE : STEAM-GENERATOR PRESSURES VS. TIME

Figure 15.1.4-8



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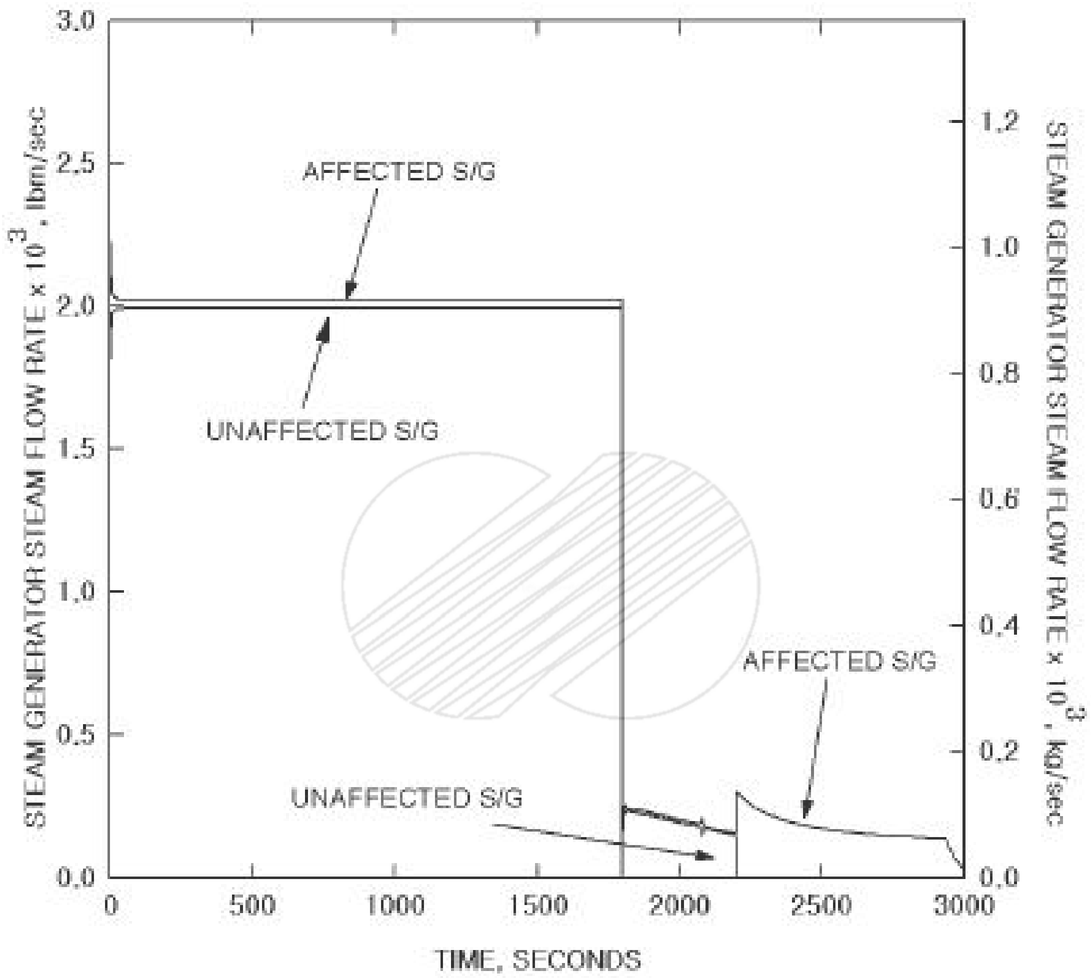
INADVERTENT OPENING OF AN
ATMOSPHERIC DUMP VALVE :
STEAM FLOW RATE TO ATMOSPHERE VS. TIME

Figure 15.1.4-9



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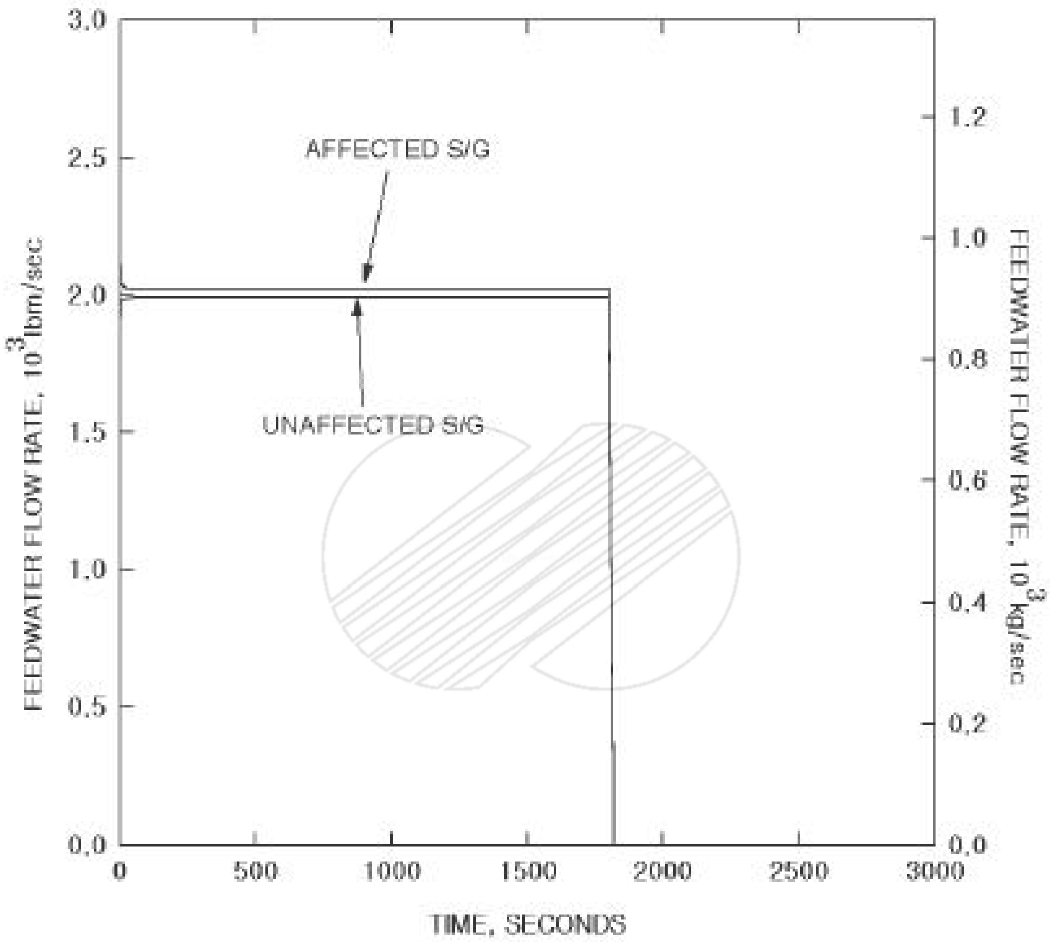
INADVERTENT OPENING OF AN
ATMOSPHERIC DUMP VALVE :
STEAM-GENERATOR STEAM FLOW RATE VS. TIME

Figure 15.1.4-10



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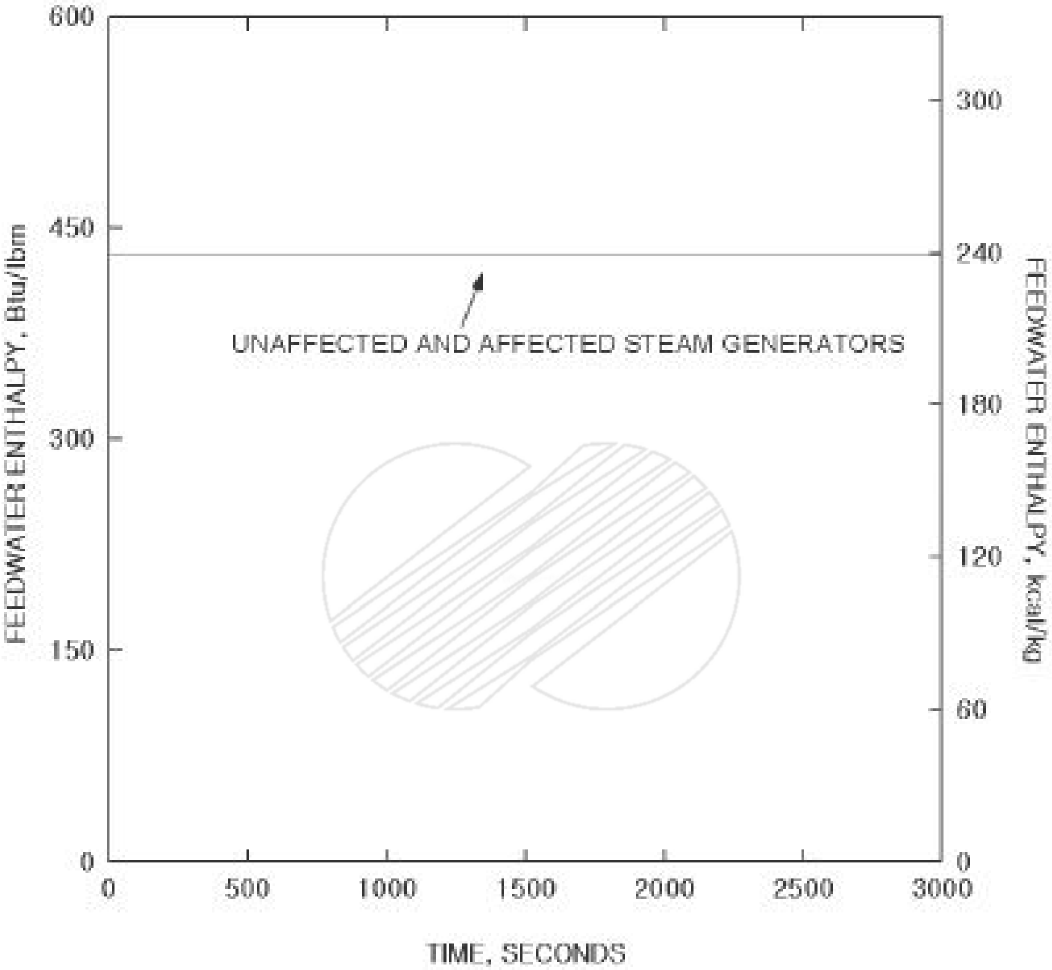
INADVERTENT OPENING OF AN
ATMOSPHERIC DUMP VALVE :
FEEDWATER FLOW RATE VS. TIME

Figure 15.1.4-11



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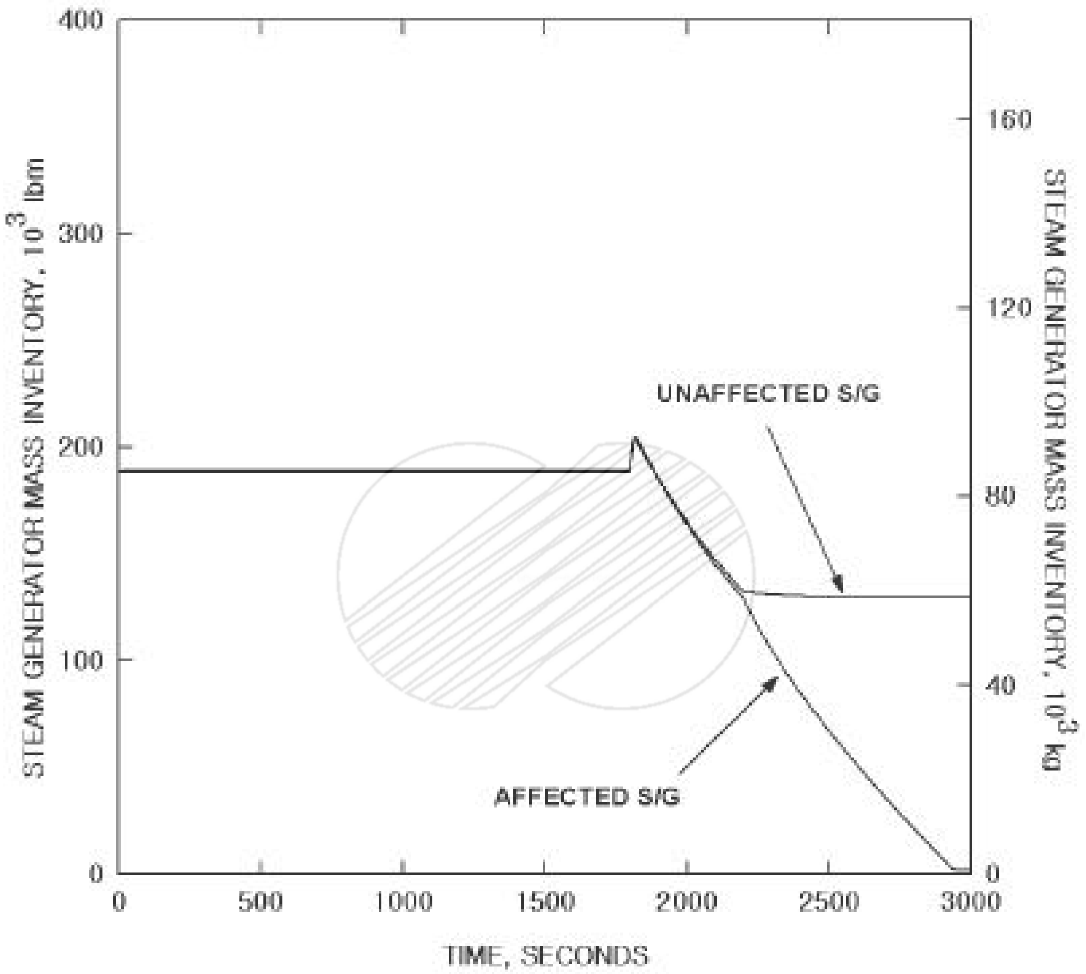
INADVERTENT OPENING OF AN
ATMOSPHERIC DUMP VALVE :
FEEDWATER ENTHALPY VS. TIME

Figure 15.1.4-12



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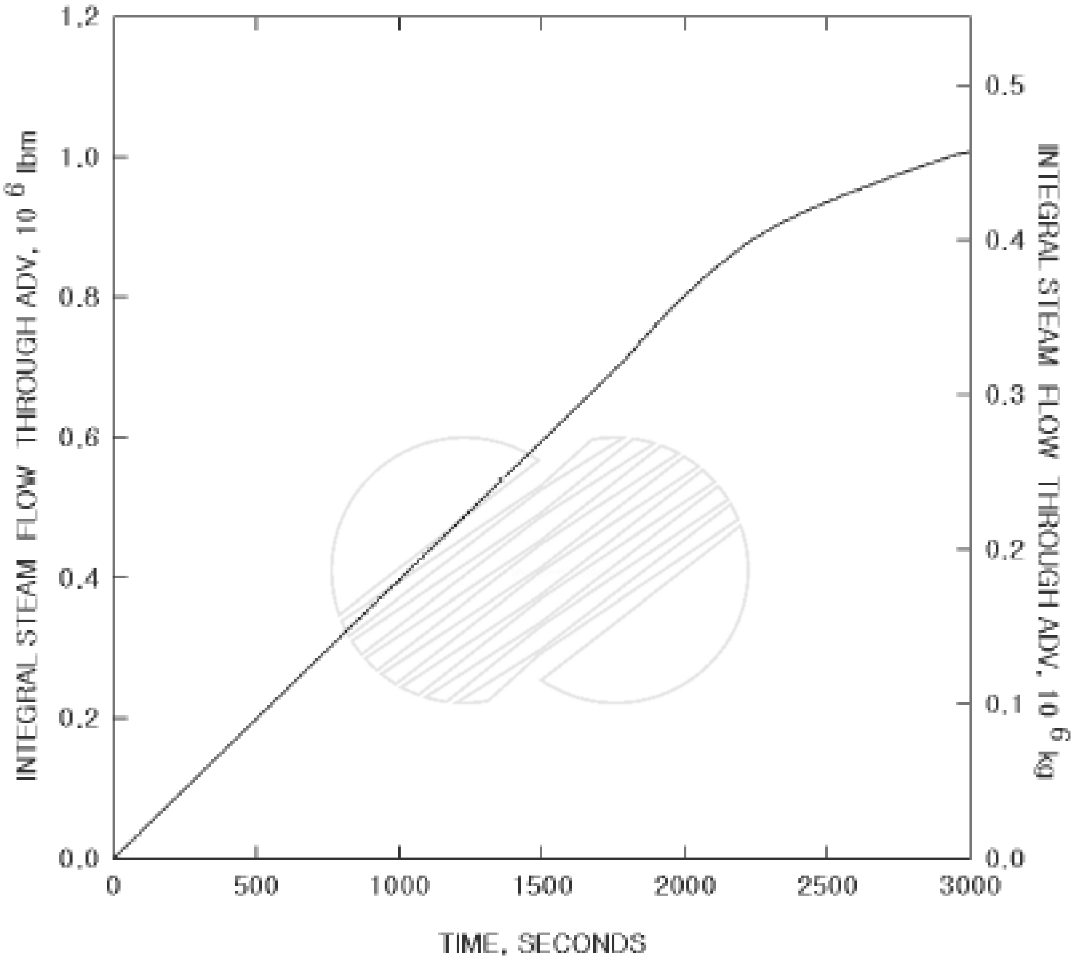
INADVERTENT OPENING OF AN
ATMOSPHERIC DUMP VALVE :
STEAM-GENERATOR MASS INVENTORY VS. TIME

Figure 15.1.4-13



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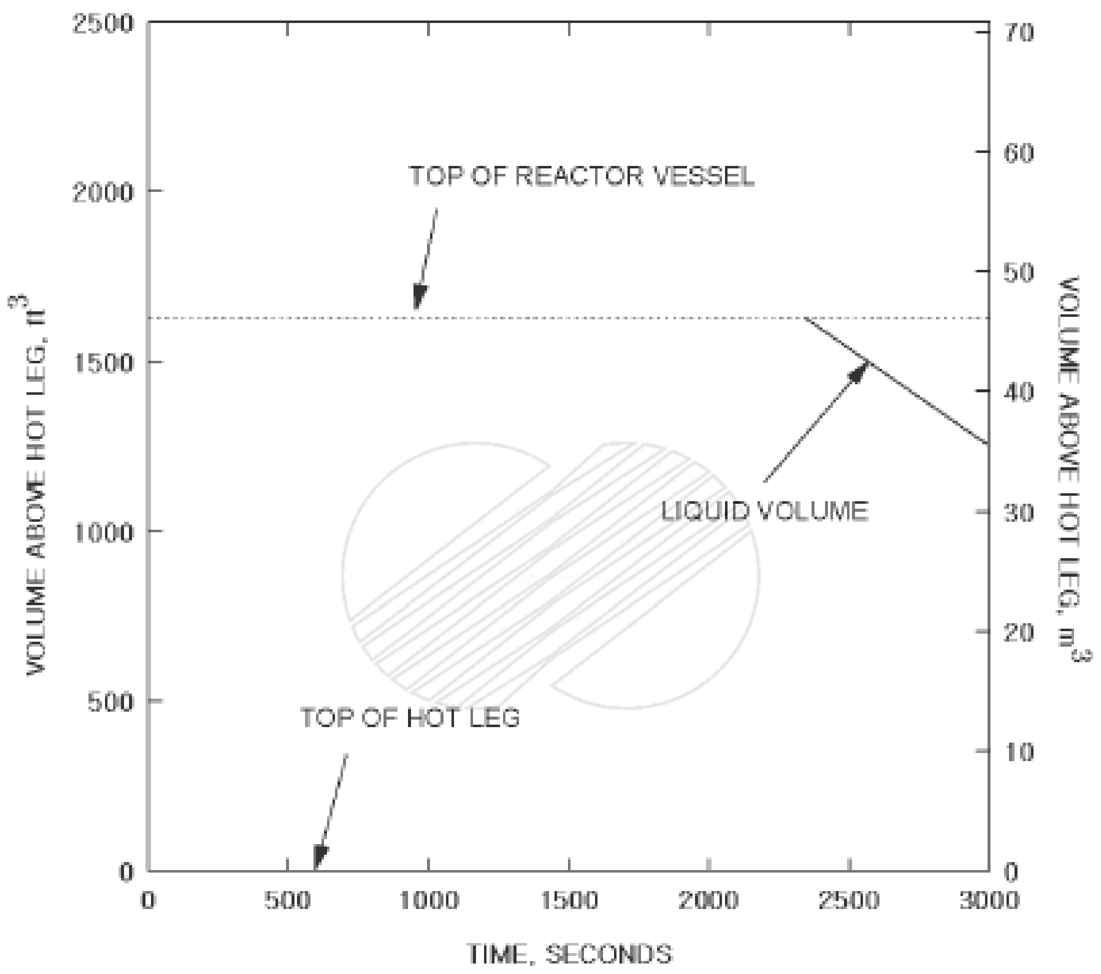
INADVERTENT OPENING OF AN ATMOSPHERIC DUMP VALVE : INTEGRAL STEAM FLOW TO ATMOSPHERE THROUGH ADV VS. TIME

Figure 15.1.4-14



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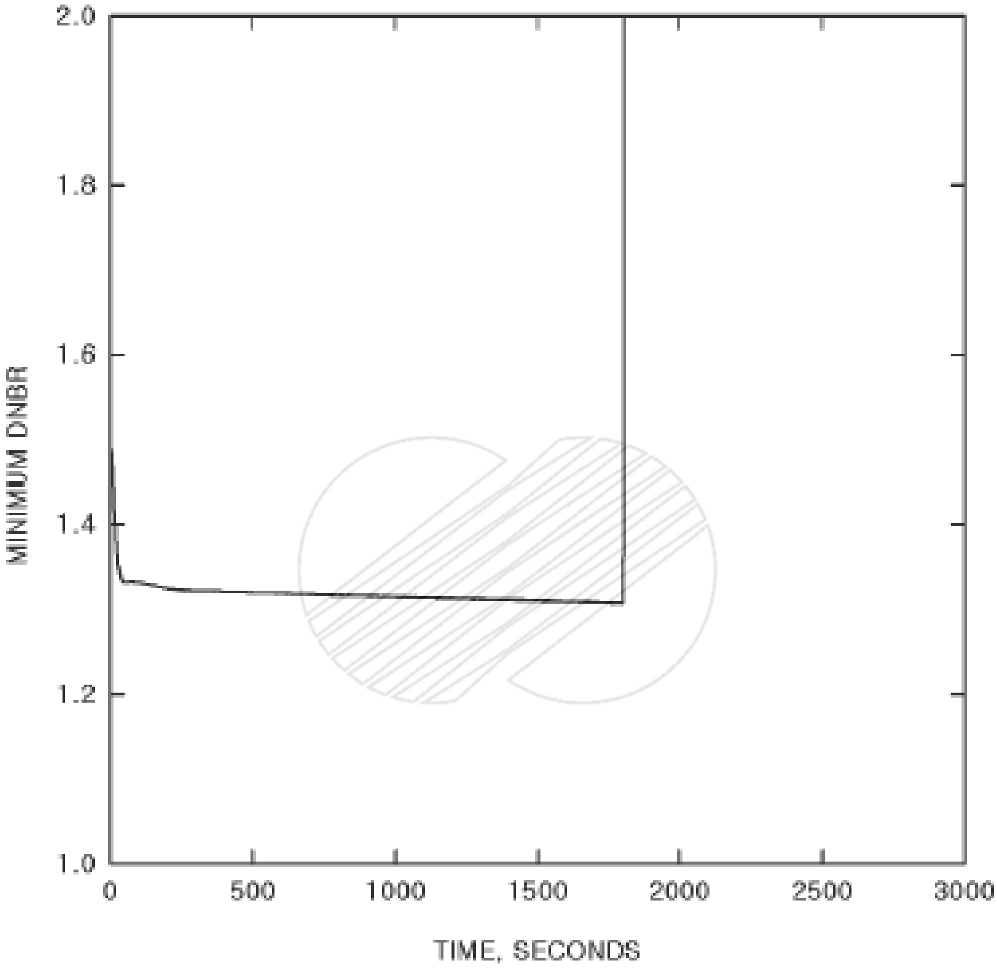
INADVERTENT OPENING OF AN
ATMOSPHERIC DUMP VALVE :
VOLUME ABOVE HOT LEG VS. TIME

Figure 15.1.4-15



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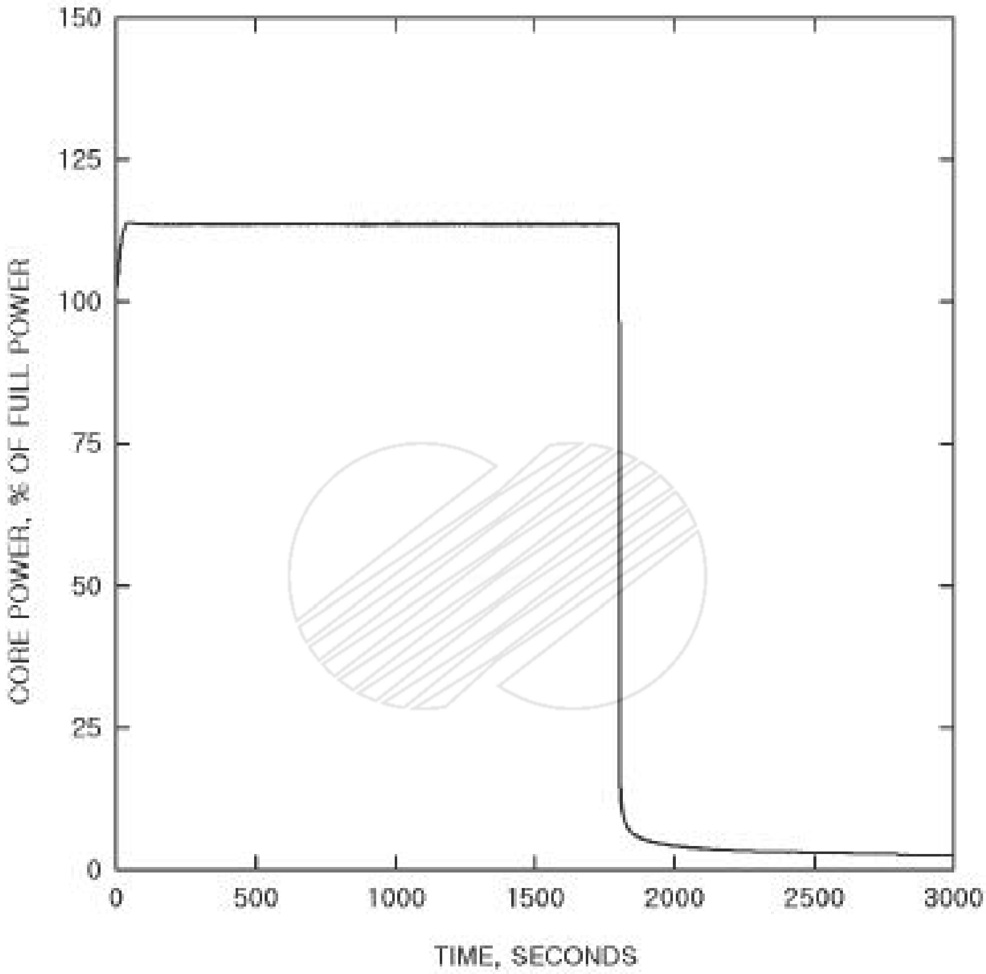
IOSGADV WITH SINGLE FAILURE:
MINIMUM DNBR VS. TIME

Figure 15.1.4-16



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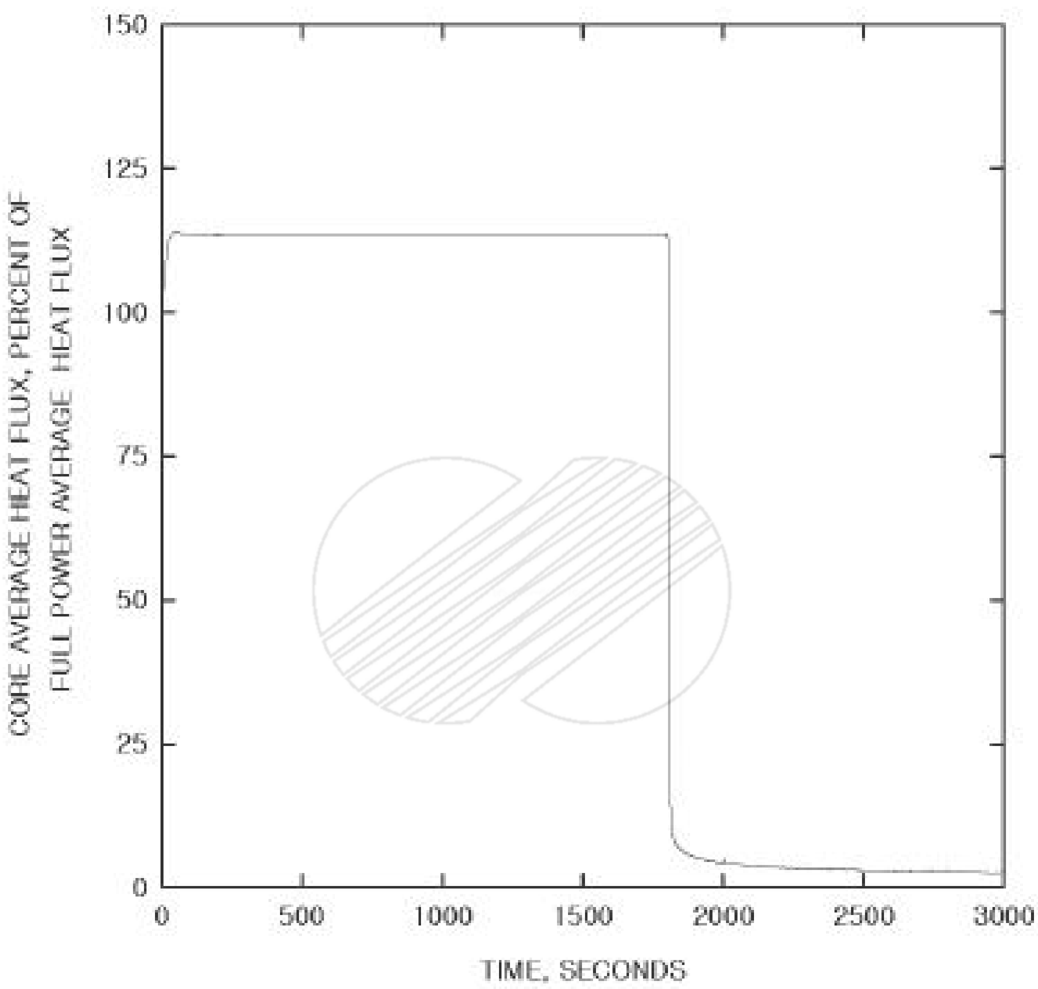
IOSGADV WITH SINGLE FAILURE:
CORE POWER VS. TIME

Figure 15.1.4-17



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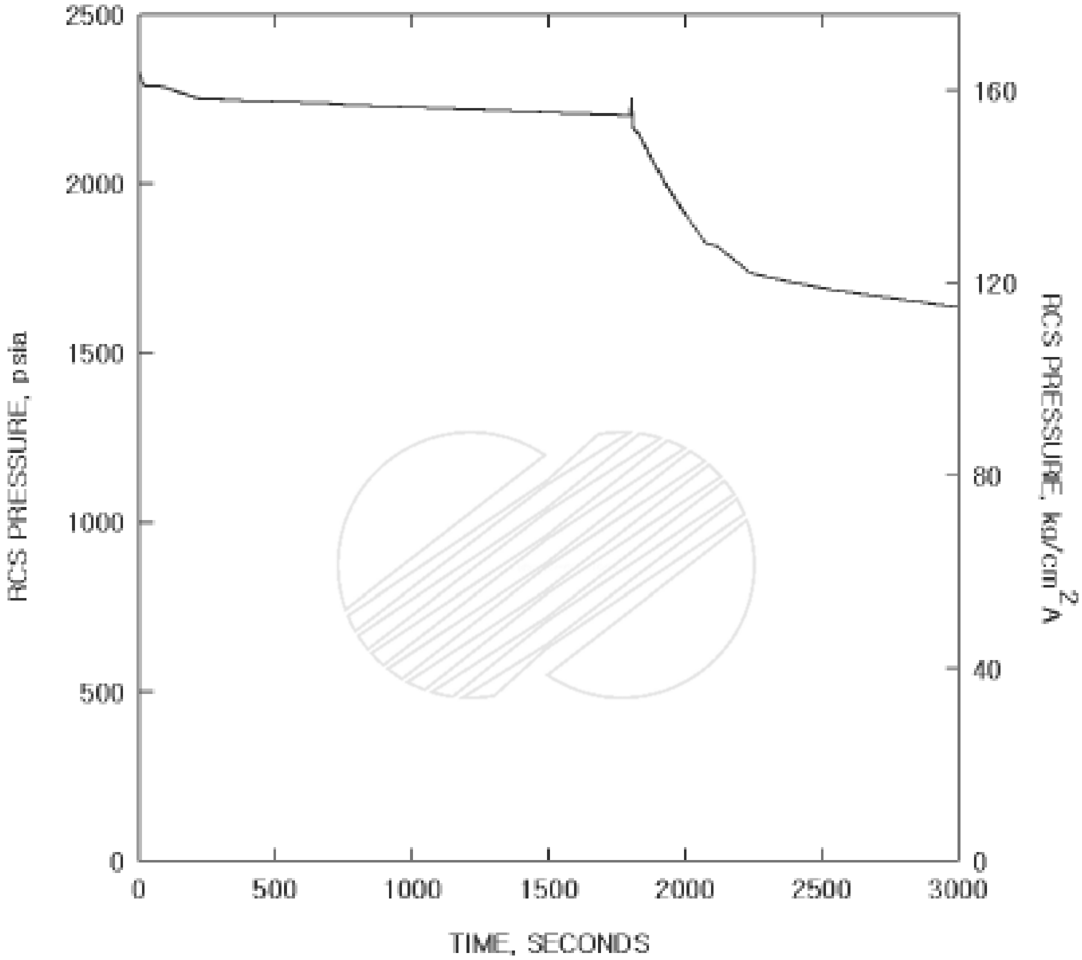
IOSGADV WITH SINGLE FAILURE:
CORE AVERAGE HEAT FLUX VS. TIME

Figure 15.1.4-18



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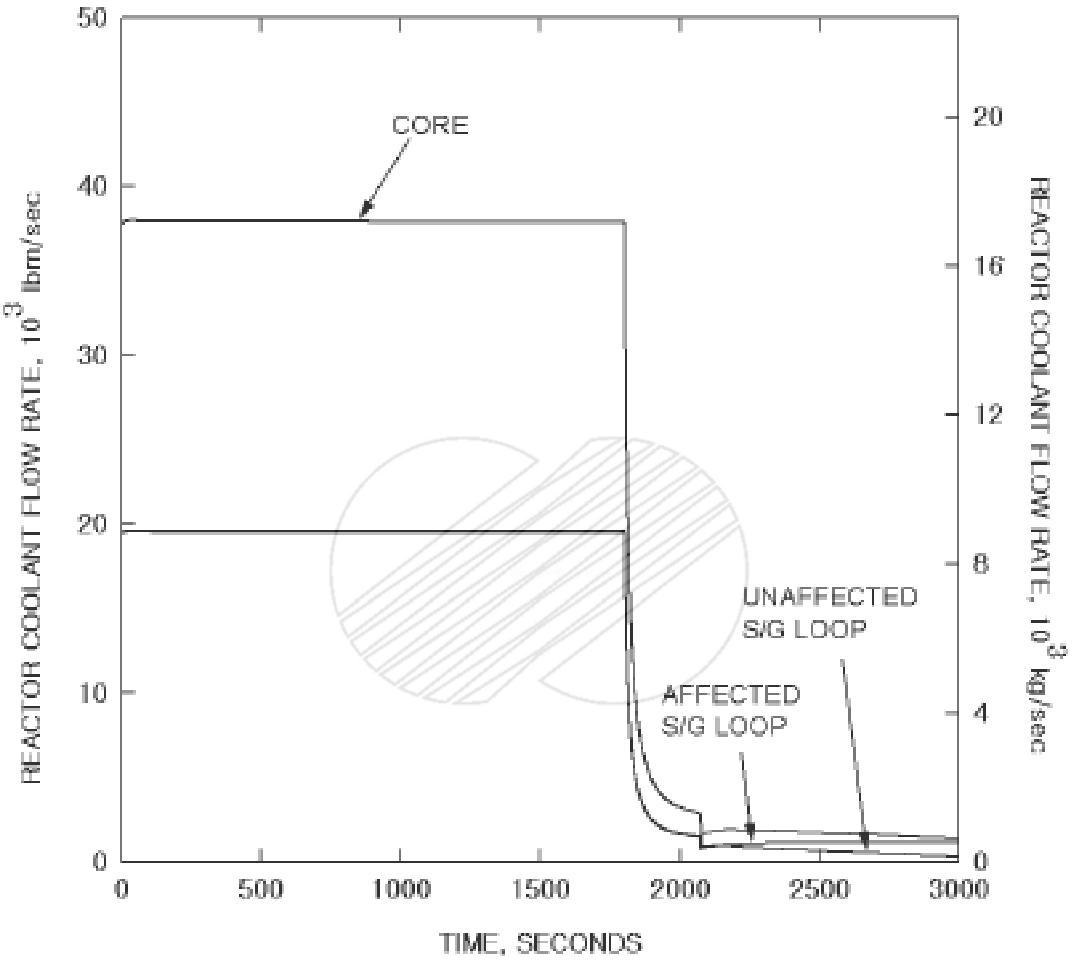
IOSGADV WITH SINGLE FAILURE:
RCS PRESSURE VS. TIME

Figure 15.1.4-19



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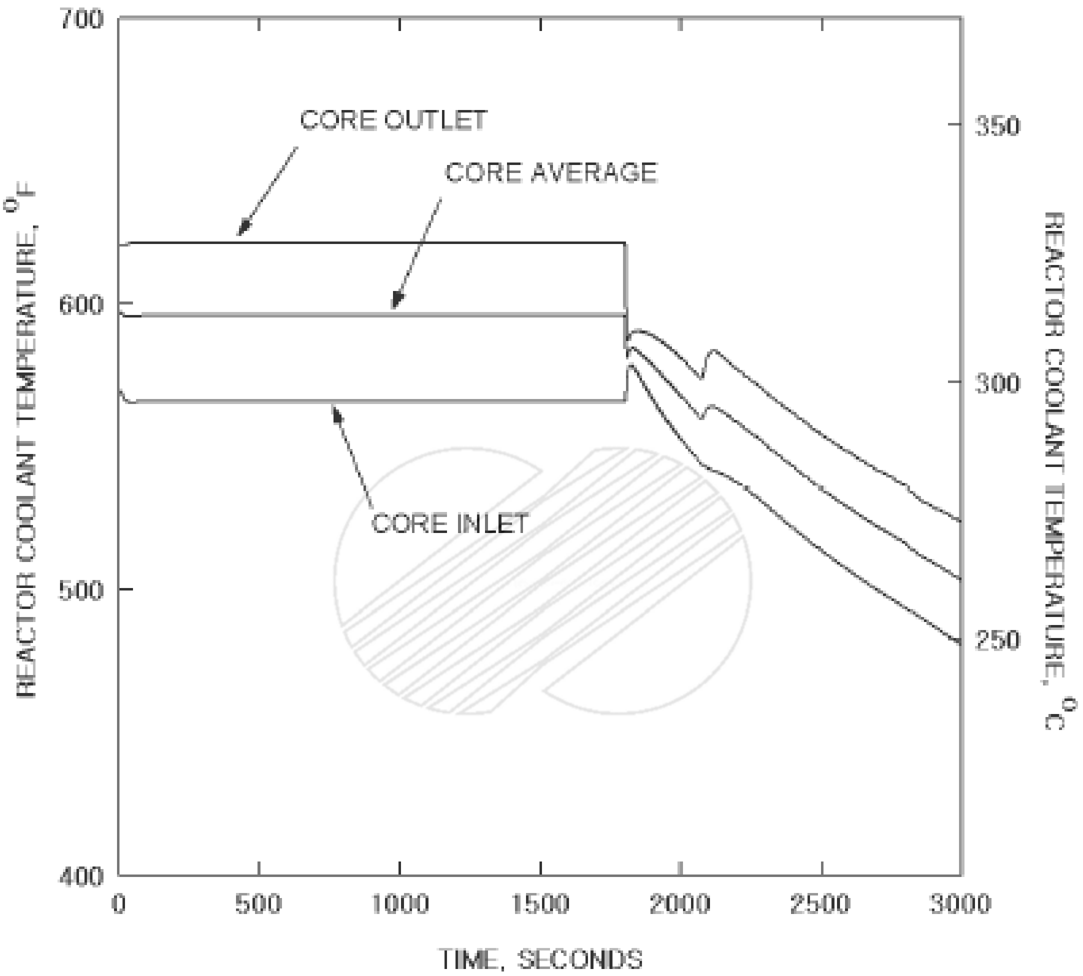
IOSGADV WITH SINGLE FAILURE:
REACTOR COOLANT FLOW RATE VS. TIME

Figure 15.1.4-20



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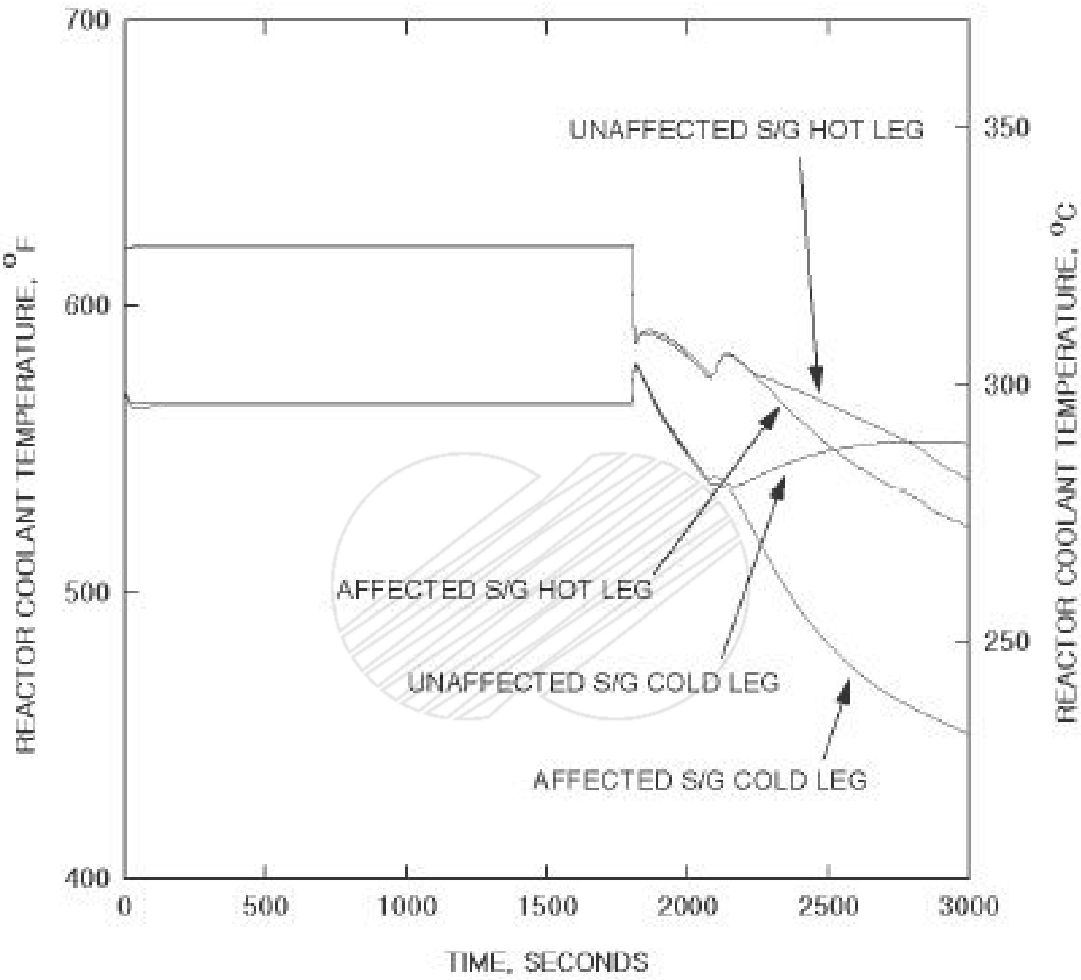
IOSGADV WITH SINGLE FAILURE:
REACTOR COOLANT TEMPERATURE (A) VS. TIME

Figure 15.1.4-21



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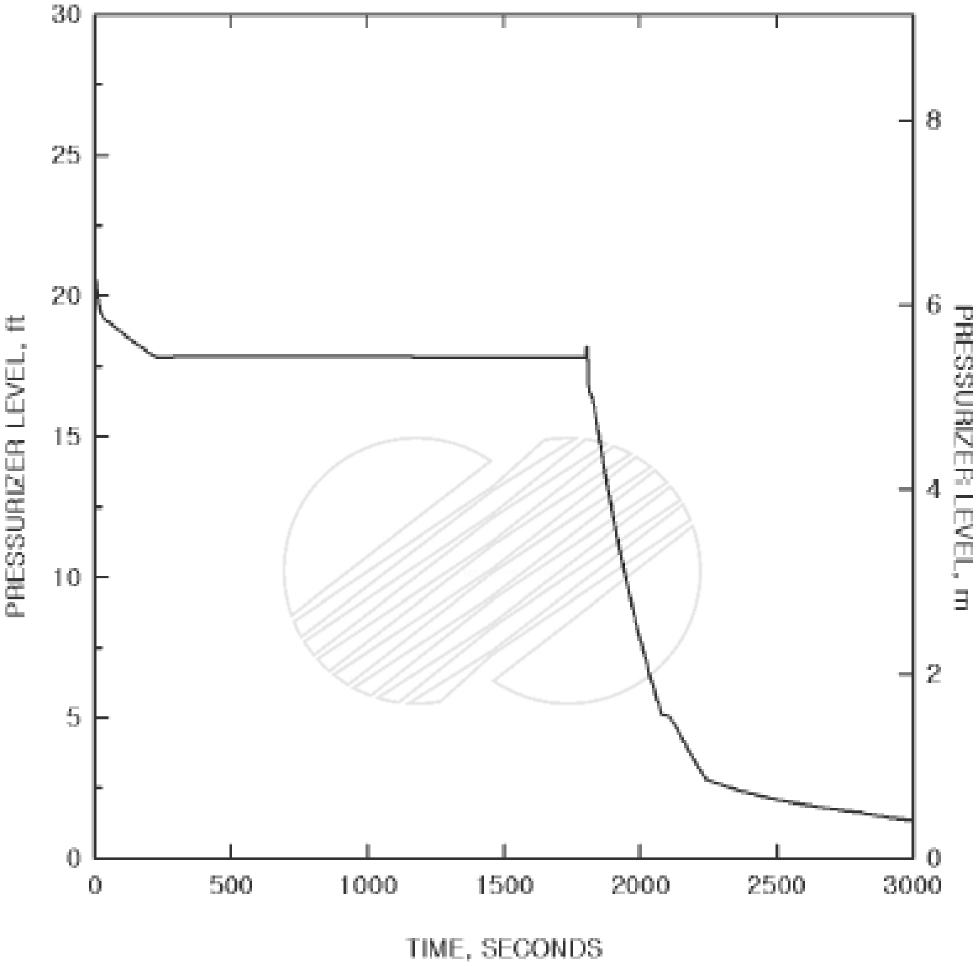
IOSGADV WITH SINGLE FAILURE:
REACTOR COOLANT TEMPERATURE (B) VS. TIME

Figure 15.1.4-22



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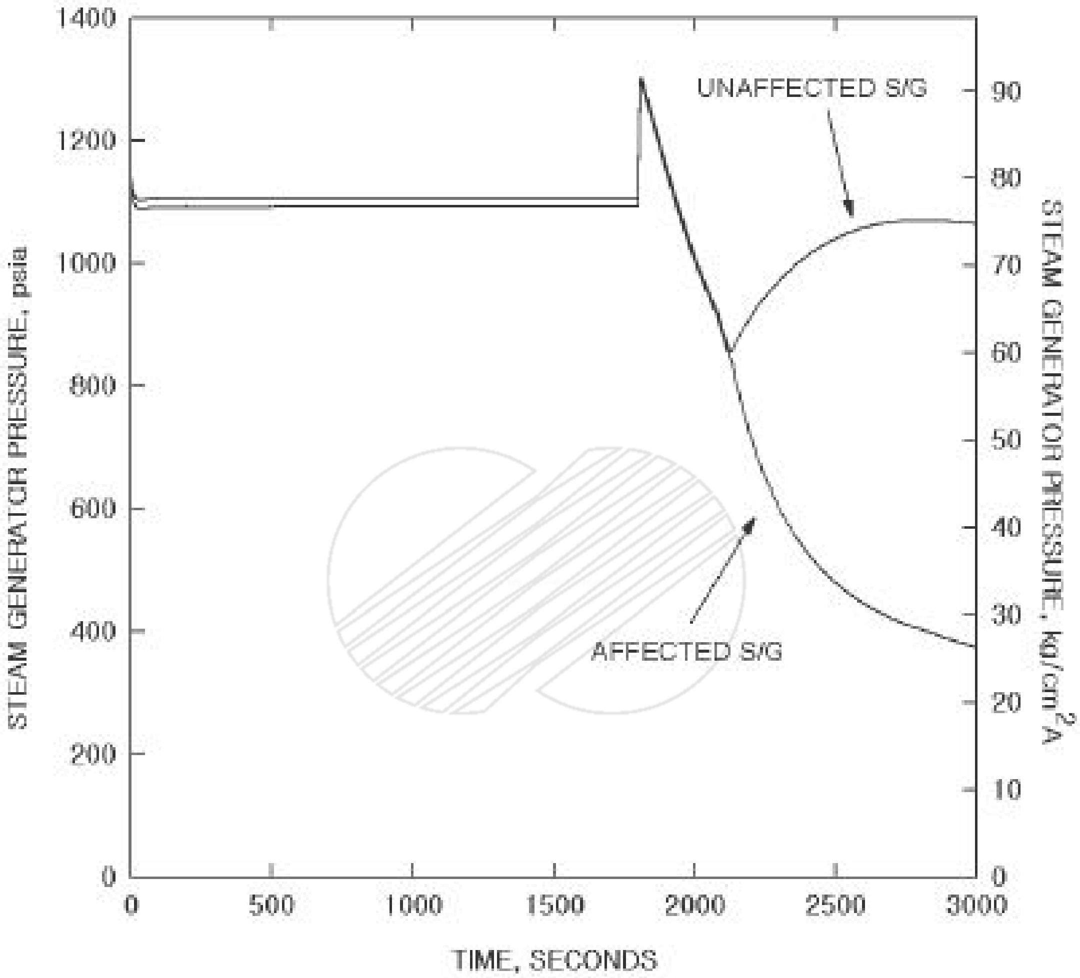
IOSGADV WITH SINGLE FAILURE:
PRESSURIZER WATER LEVEL VS. TIME

Figure 15.1.4-23



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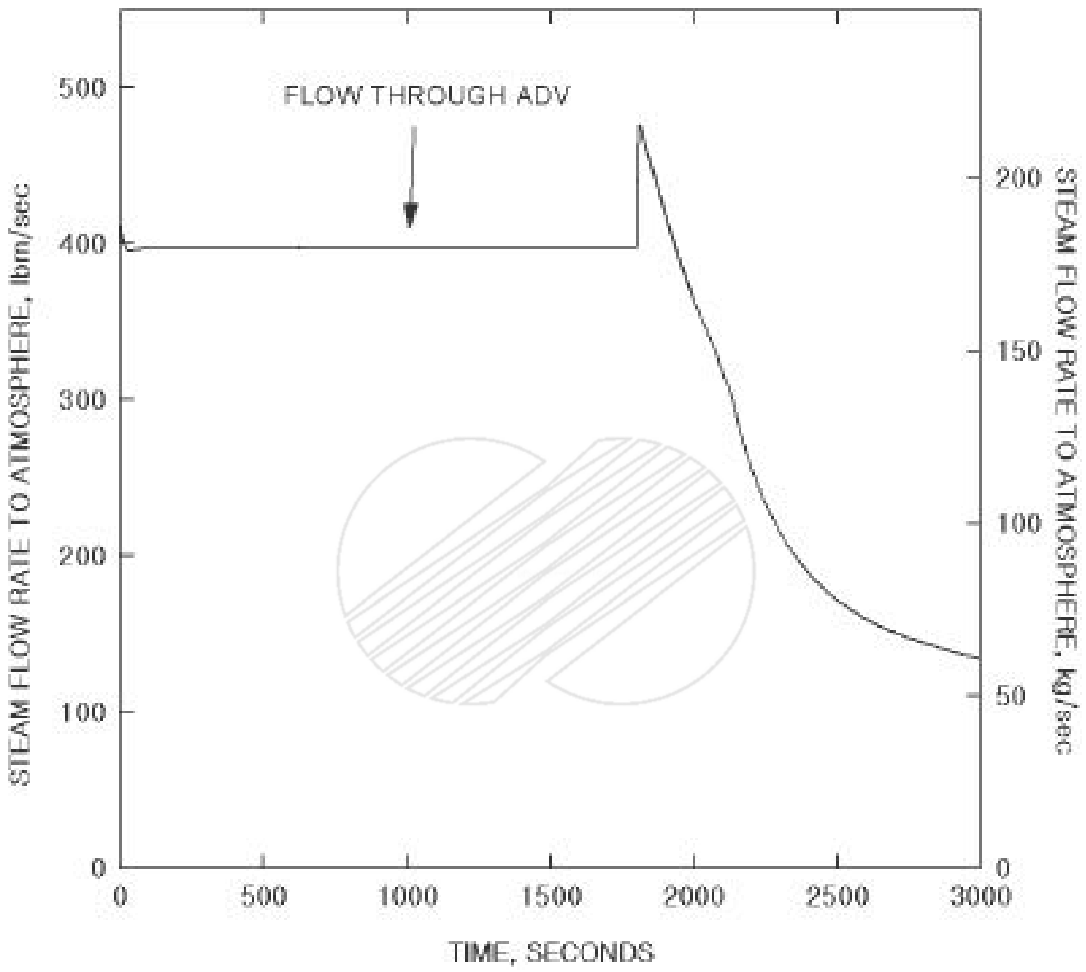
IOSGADV WITH SINGLE FAILURE:
STEAM-GENERATOR PRESSURES VS. TIME

Figure 15.1.4-24



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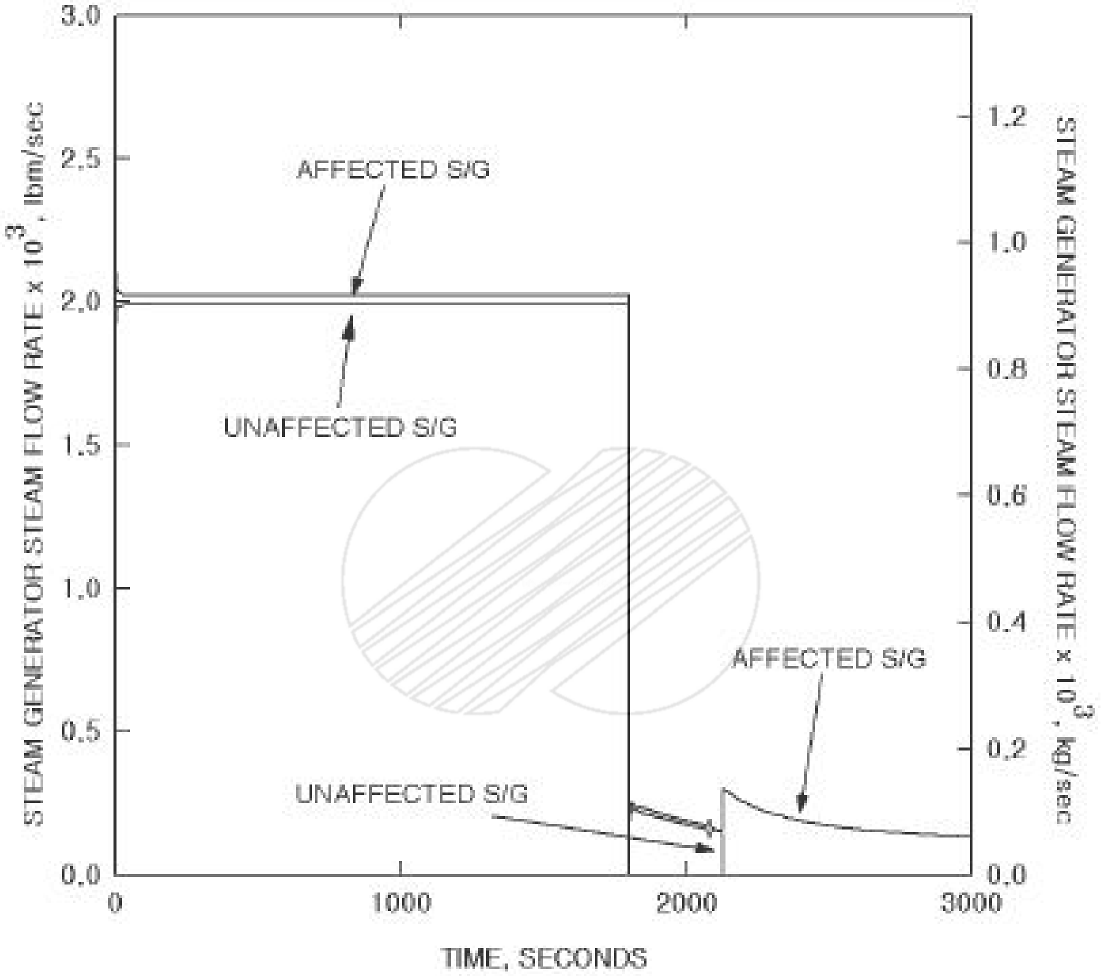
IOSGADV WITH SINGLE FAILURE:
STEAM FLOW RATE TO ATMOSPHERE VS. TIME

Figure 15.1.4-25



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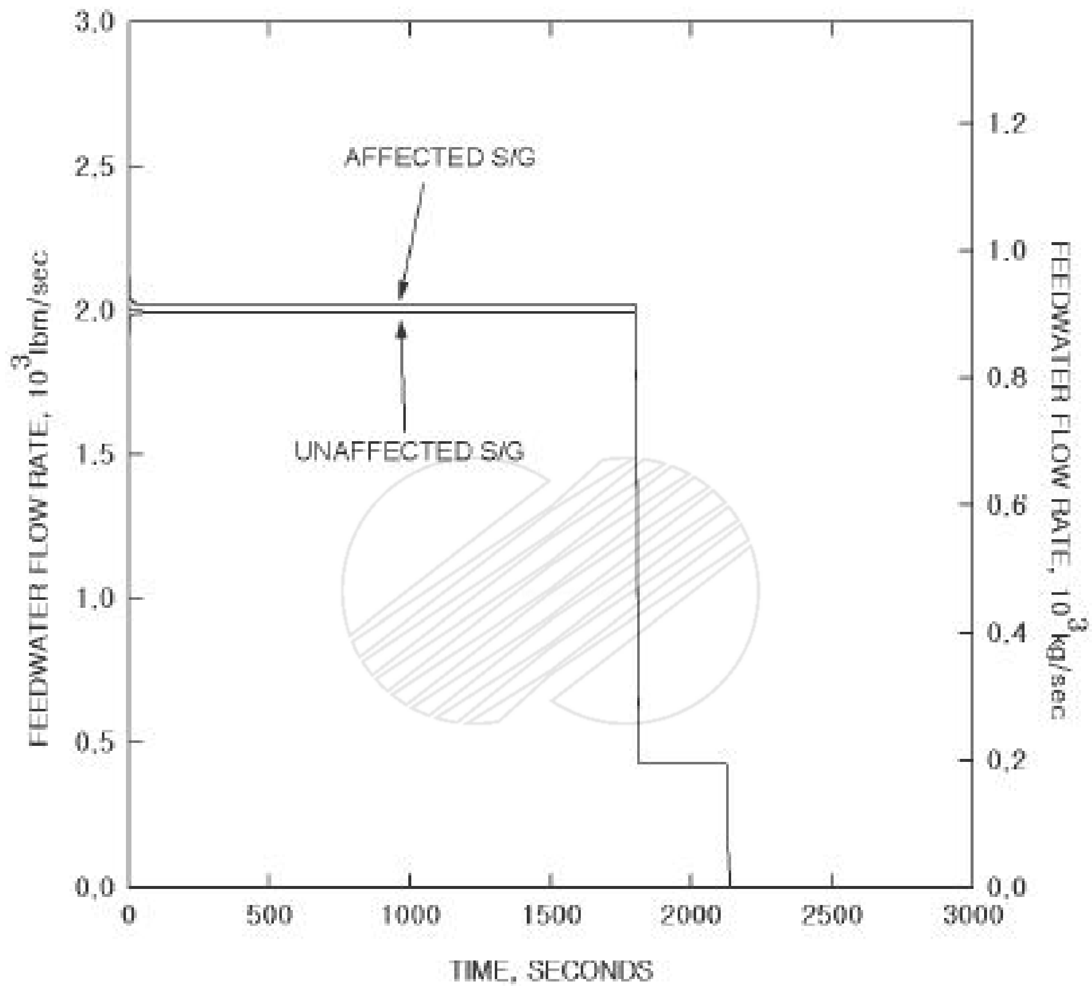
IOSGADV WITH SINGLE FAILURE:
STEAM-GENERATOR STEAM FLOW RATE VS. TIME

Figure 15.1.4-26



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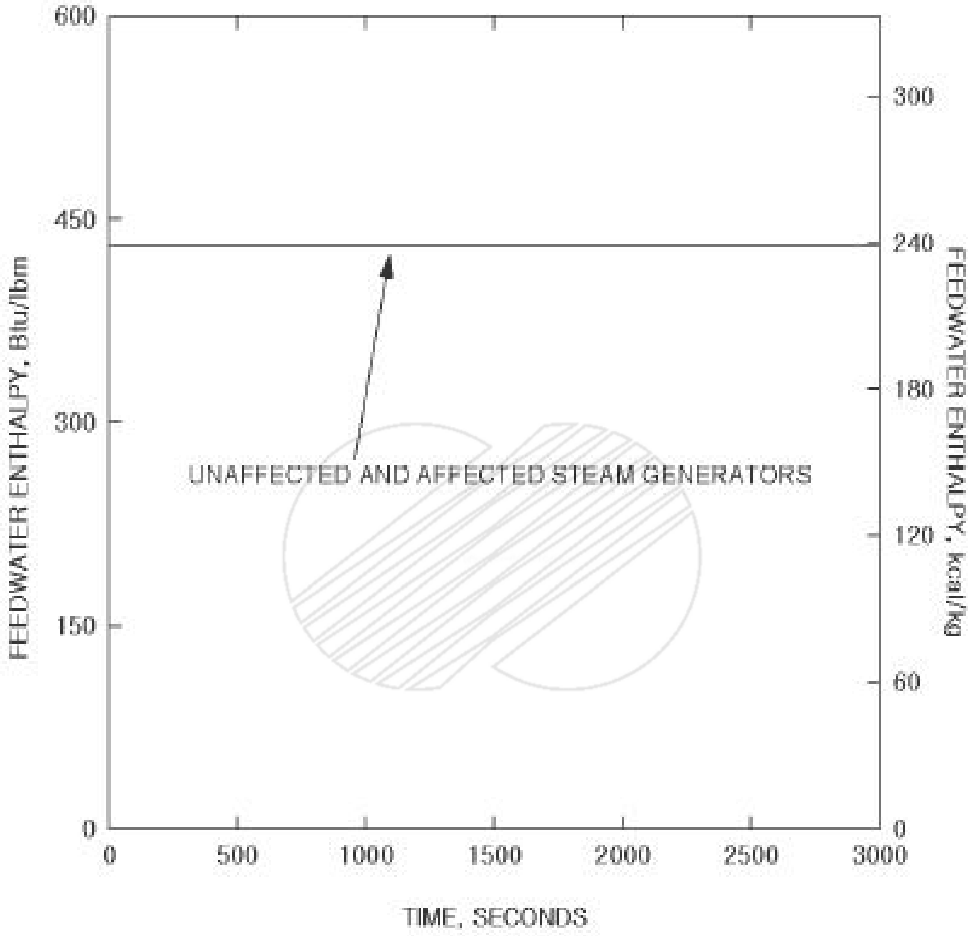
IOSGADV WITH SINGLE FAILURE:
FEEDWATER FLOW RATE VS. TIME

Figure 15.1.4-27



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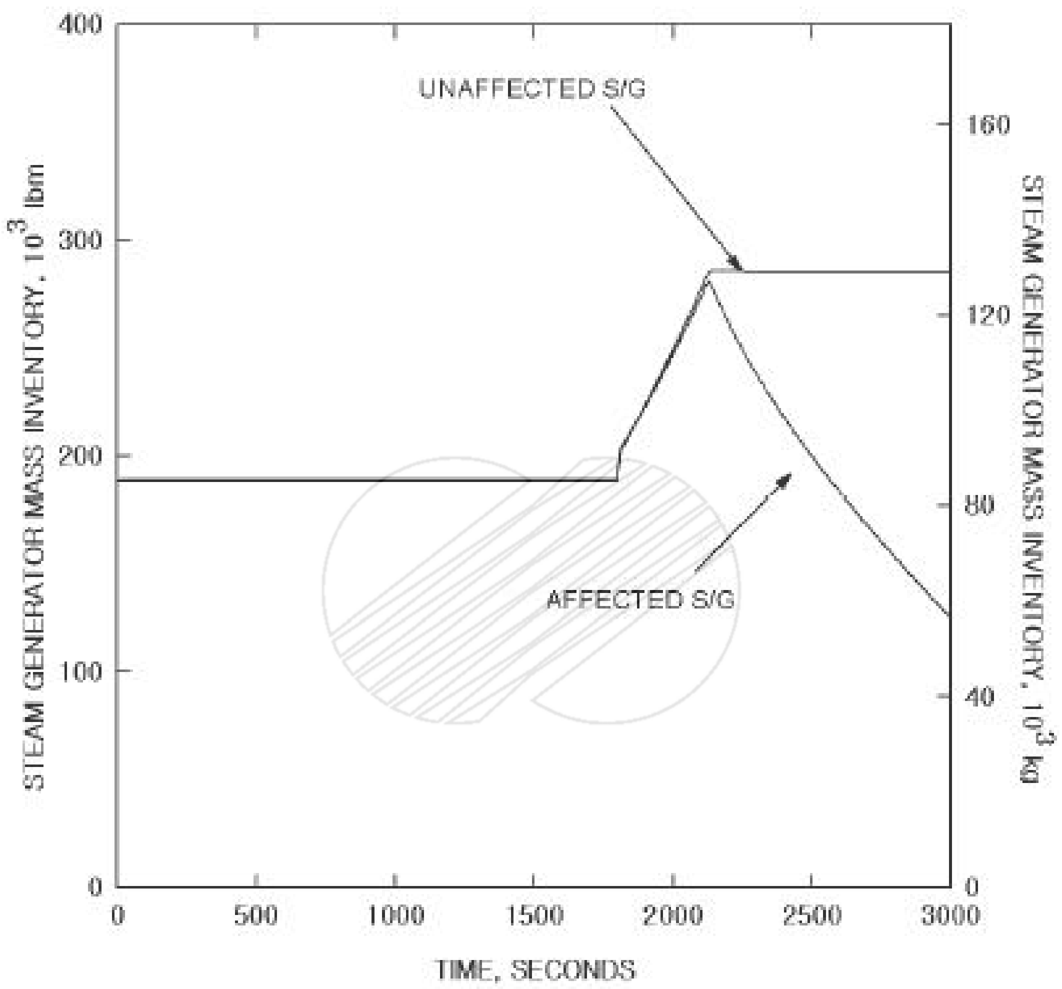
IOSGADV WITH SINGLE FAILURE:
FEEDWATER ENTHALPY VS. TIME

Figure 15.1.4-28



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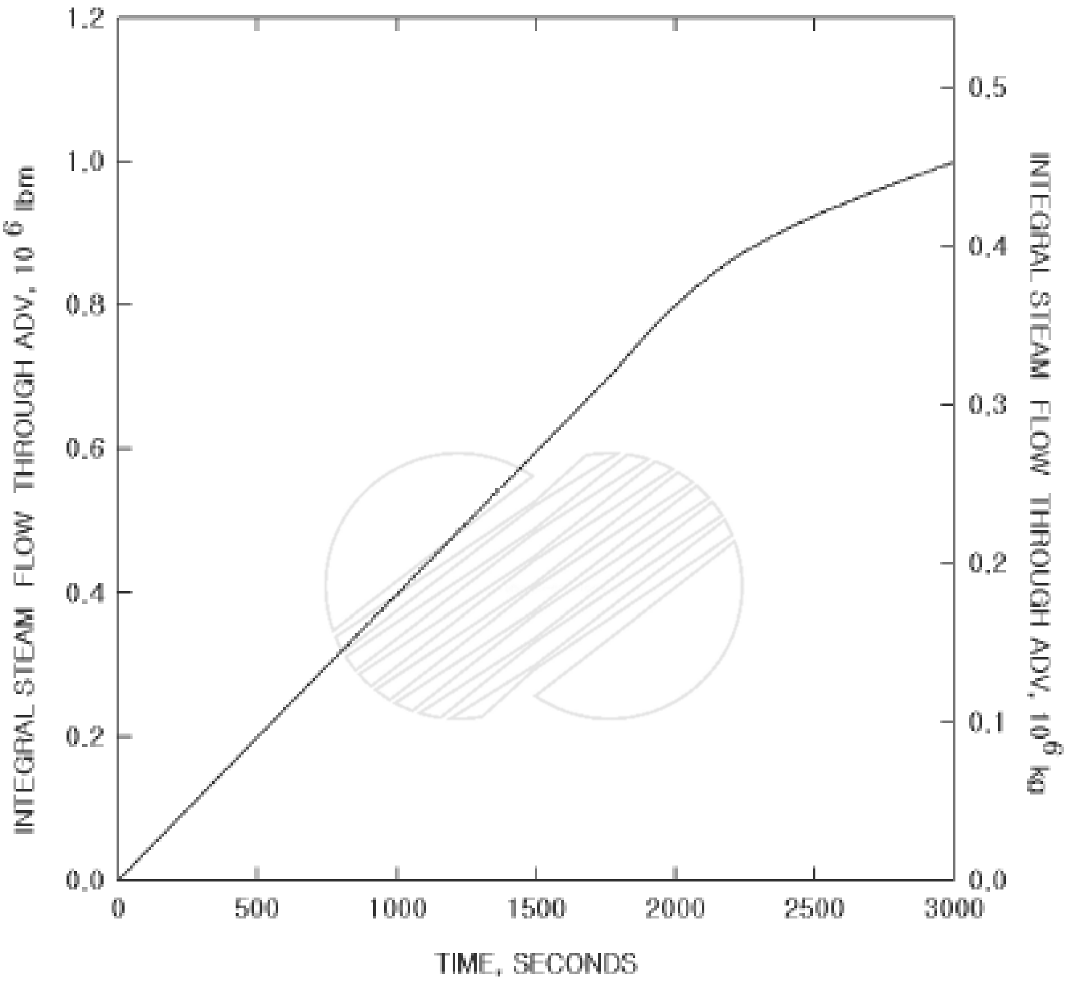
IOSGADV WITH SINGLE FAILURE:
STEAM-GENERATOR MASS INVENTORY VS. TIME

Figure 15.1.4-29



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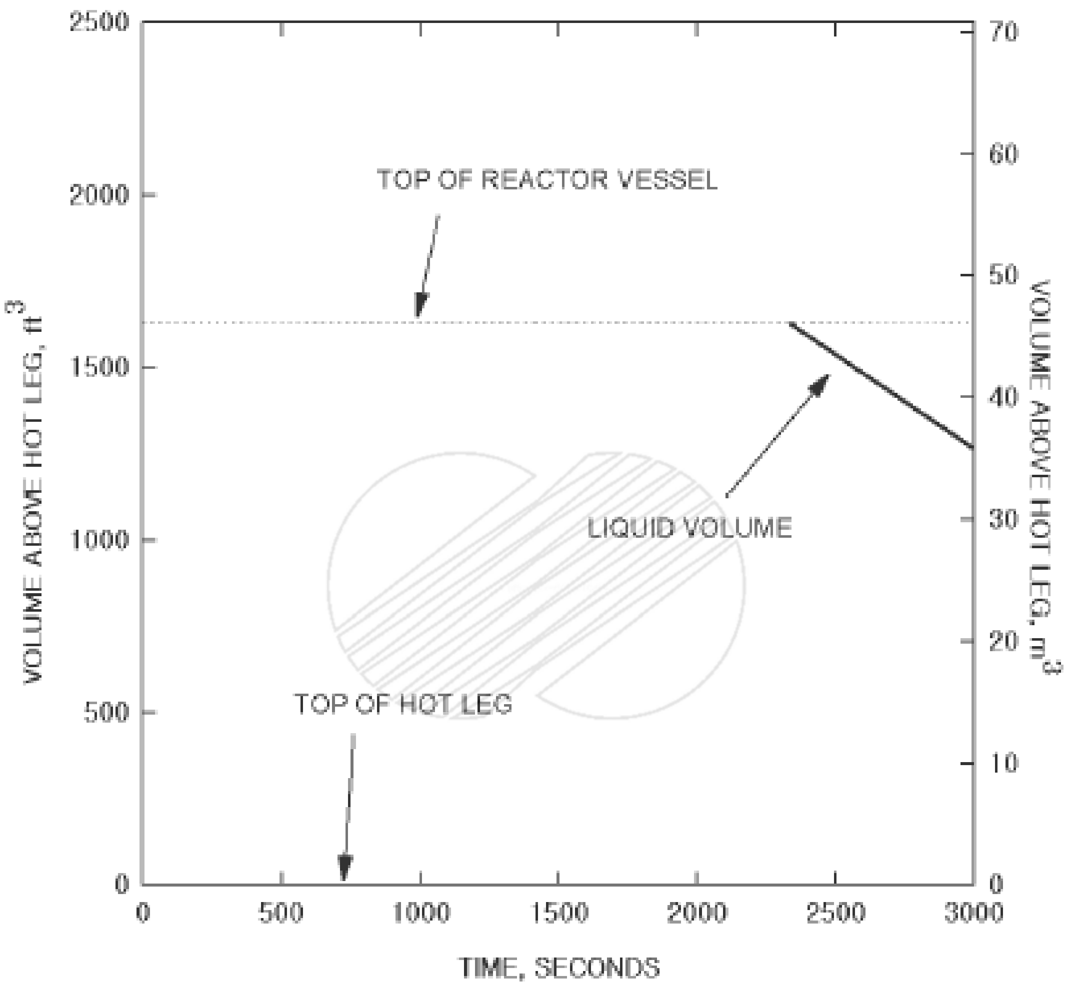
IOSGADV WITH SINGLE FAILURE:
INTEGRAL STEAM FLOW TO
ATMOSPHERE THROUGH ADV VS. TIME

Figure 15.1.4-30



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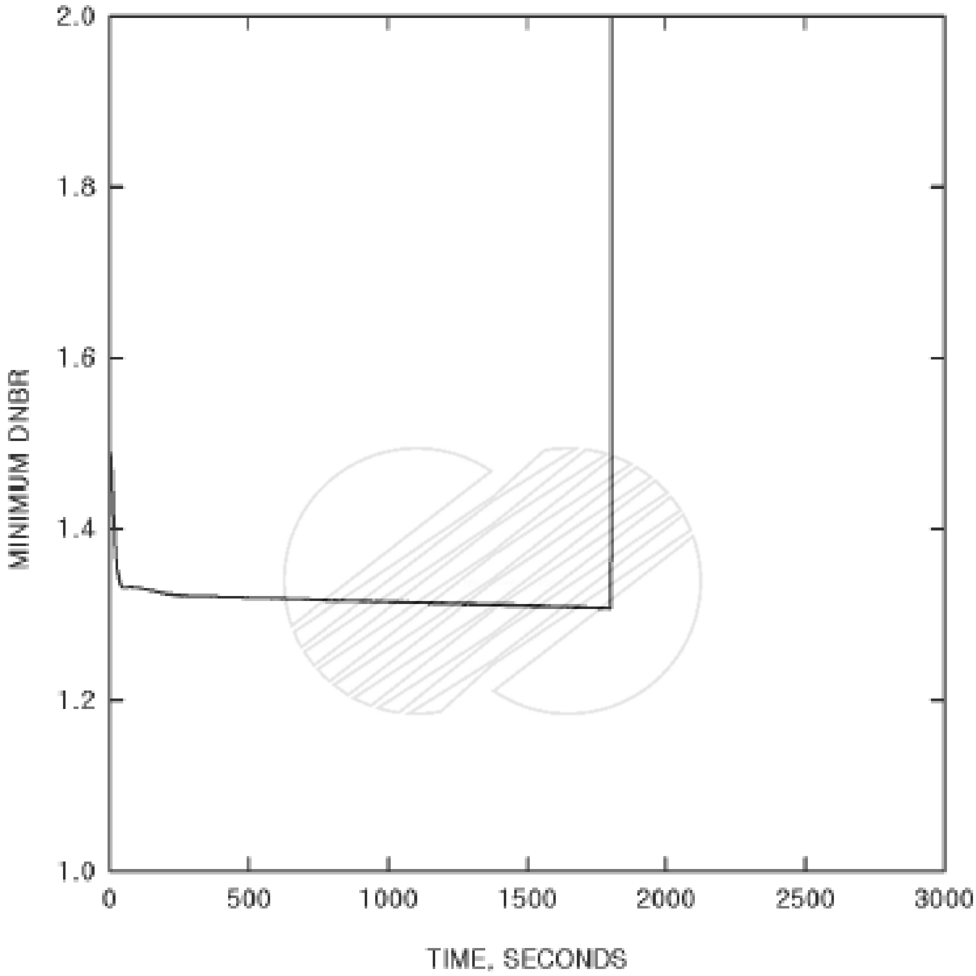
IOSGADV WITH SINGLE FAILURE:
VOLUME ABOVE HOT LEG VS. TIME

Figure 15.1.4-31



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IOSGADV WITH SINGLE FAILURE:
MINIMUM DNBR VS. TIME

Figure 15.1.4-32

15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT

15.1.5.1 Identification of Event and Causes

A steamline break (SLB) is defined as a pipe break in the main steam system. SLB cases are chosen to maximize potential for a post-trip return to power, to maximize potential for degradation in fuel cladding performance, and to maximize dose at the site exclusion area boundary. The results show that fission power levels remain sufficiently low following reactor trip to preclude degradation in fuel performance as a result of post-trip return to power, that degradation in fuel performance prior to trip is of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability, and that doses are within 10 CFR 100 guidelines. The steamline breaks presented are as follows:

a. Cases chosen to maximize potential for a post-trip return to power:

1. A large steamline break inside containment during full power operation with concurrent loss-of-offsite power in combination with a single failure, and a stuck CEA (SLBFPLOOP).

2. A large steamline break inside containment during full power operation with offsite power available in combination with a single failure and a stuck CEA (SLBFP).

3. A large steamline break inside containment during zero power operation with concurrent loss-of-offsite power in combination with a single failure and a stuck CEA (SLBZPLOOP).

4. A large steamline break inside containment during zero power operation with offsite power available in combination with a single failure and a stuck CEA (SLBZP).

b. Cases chosen to maximize potential for degradation in fuel performance and dose at the site exclusion area boundary:

1. A large steamline break outside of containment upstream of the main steam isolation valve (MSIV) during full power operation with offsite power available in combination with a single failure, technical specification steam generator tube leakage, and a stuck CEA (SLBFPD).

2. A large steamline break outside of containment upstream of the MSIV during zero power operation with concurrent loss-of-offsite power in combination with a single failure, technical specification steam-generator tube leakage, iodine spike, and a stuck CEA (SLBZPLOOPD).

The largest possible steamline break size is the double-ended rupture of a steamline upstream of the MSIV. In the YGN 3&4 design, an integral flow restrictor exists in each steam generator outlet nozzle. The largest effective steam blowdown area for each steamline, which is limited by the flow restrictor throat area, is 30% of the steamline cross-sectional area, or 0.942 ft^2 (0.0875 m^2).

Results are presented in Appendix 15C which demonstrate that the cases listed above bound the results obtained for a spectrum of break sizes, loss-of-offsite power times, and single failures.

15.1.5.2 Sequence of Events and System Operations

Steamline breaks are characterized as cooldown events due to the increased steam flow rate, which causes excessive energy removal from the steam generators and the reactor coolant system (RCS). This results in a decrease in reactor coolant temperatures and in RCS and steam generator pressure. The cooldown causes an increase in core reactivity due to the negative moderator and Doppler reactivity coefficients.

Detection of the cooldown is accomplished by the pressurizer and steam-generator low-pressure alarms, by the high reactor power alarm, and by the low steam-generator water level alarm. Reactor trip as a consequence of a steamline break is provided by one of several available reactor trip signals including low steam-generator pressure, low RCS pressure, low steam-generator water level, high reactor power, low DNBR trip initiated by the core protection calculators, and for inside containment breaks, high containment pressure. For an SLB that occurs with a concurrent loss-of-offsite power, the events of turbine stop valve closure, termination of feedwater to both steam generators, and coastdown of the reactor coolant pumps are assumed to be initiated simultaneously. Full auxiliary feedwater (AFW) actuation to the ruptured steam generator is conservatively assumed to be initiated at any time between the event initiation and the time point when the SG level
reaches the minimum AFW actuation setpoint, which is determined 812
considering measurement uncertainty. Following reactor trip the most reactive control rod is conservatively assumed to be held in the fully withdrawn position. The depressurization of the affected steam generator results in the actuation of a main steam isolation signal (MSIS). This closes the MSIVs, isolating the unaffected steam generator from blowdown and closes the main feedwater isolation valves (MFIVs), terminating main feedwater flow to both steam generators. When the differential pressure between the two steam generators exceeds the setpoint, the AFW logic isolates AFW flow to the affected steam generator.

"Delete" 812

The pressurizer pressure decreases to the point where a safety injection actuation signal (SIAS) is initiated. The isolation of the unaffected steam generator and subsequent emptying of the affected steam generator terminate the cooldown. The introduction of safety injection boron upon SIAS causes core reactivity to decrease. The operator, via the appropriate emergency procedures, may initiate plant cooldown by manual control of the atmospheric steam dump valves, or, in the event that offsite power is available, by using the MSIV bypass valves associated with the unaffected steam generator and the turbine bypass valves, any time after the affected steam generator empties. The analysis presented herein conservatively assumes operator action is delayed until 30 minutes after first indication of the event. The plant is then cooled to 350°F (177°C) and 410 psia (28.82 kg/cm^{2}A), at which point shutdown cooling system operation is initiated.

A parametric study of single failures (see Appendix 15C) that would have an adverse impact on the SLB has determined that the failure of one of the high pressure safety injection (HPSI) pumps to start following SIAS has the most adverse effect for all cases except “Delete” Case 2 and Case 5. Consequently, one HPSI pump is conservatively assumed to fail for all cases except Case 2 and Case 5. The failure of an MSIV in the unaffected steam generator to close yields the most adverse effect for Case 2. 812

“Delete” For Case 5 (SLBFPD) there is no single failure which increases the potential for degradation in fuel cladding performance or which increases the offsite dose. However, the failure of an MSIV in the unaffected steam generator to close was used in the analysis to be consistent with Case 2 (SLBFP). 812

“Delete”

The sequence of events for Cases 1 through 5 above are presented in Tables 15.1.5-1 through 15.1.5-5, respectively. The sequence of events for Case 6 is the same as for Case 3.

15.1.5.3 <u>Analysis of Effects and Consequences</u>

a. <u>Mathematical Models</u>

The mathematical models and data transfer between codes used in the SLB analysis are presented in Appendix 15C.

b. Input Parameters and Initial Conditions

The initial conditions assumed in the analysis of the NSSS response to Cases 1 through 5 are presented in Tables 15.1.5-6 through 15.1.5-10, respectively. The initial conditions for Case 6 are the same as those for Case 3.Justification of the selection of initial conditions and input parameters is presented in Appendix 15C.

c. Results

Case 1: Large Steamline Break During Full Power Operation with Concurrent Loss-of-Offsite Power (SLBFPLOOP)

The dynamic behavior of the salient NSSS parameters following the SLBFPLOOP is presented in Figures 15.1.5-1 through 15.1.5-16. Table 15.1.5-1 summarizes the major events, times, and results for this transient.

Concurrent with the steamline break, a loss of offsite power occurs. At this time an actuation signal for the emergency diesel generators is initiated. Due to decreasing core flow following loss of power to the reactor coolant pumps, conditions exist for a low DNBR trip. At 1.0 second, a low-reactor-coolant-pump-shaft-speed-trip signal is initiated by the core protection calculators. Simultaneously, auxiliary feedwater flow is assumed to be initiated to the affected steam generator. At 1.1 second the reactor trip breakers open. At 8.82 seconds, voids begin to form in the upper head of the reactor vessel. At 9.97 seconds, the steam-generator pressure drops below the MSIS setpoint of 789 psia (55.47 kg/cm^{2}A). The MSIS initiates 812 closure of the MSIVs and MFIVs at 11.12 seconds. The MSIVs and MFIVs close by 16.12 seconds and 21.12 seconds, respectively. At 35.88 seconds, the pressure difference between the steam generators reaches the analysis setpoint of 352 psid (24.750 kg/cm^{2}D) for isolation of

AFW to the ruptured steam generator. As a result, AFW to the ruptured
steam generator is isolated. At 195.31 seconds, the pressurizer 812
empties. At 312.62 seconds, the pressurizer pressure has dropped below
safety injection setpoint of 1555 psia (109.32 kg/cm^{2}A). Within 30
seconds of SIAS the operable HPSI pump is loaded on an emergency
diesel generator and reaches full speed and the HPSI valves are fully
open.

Safety injection boron begins to reach the core at 519.78 seconds. At 812
662.42 seconds, the maximum core reactivity (-0.131% $\triangle\rho$) occurs. The
values of DNBR remains above 10 during the post-trip 645
approach-to-criticality portion of this transient. At a maximum of 30
minutes the operator, via the appropriate emergency procedure,
initiates plant cooldown by manual control of the atmospheric dump
valves, assuming that offsite power has not been restored. Shutdown
cooling system operation is initiated when the RCS reaches 350 °F
(176.7 °C) and 410 psia (28.82 kg/cm^{2}A).

CASE 2: Large Steamline Break During Full Power Operation with Offsite Power Available (SLBFP)

The dynamic behavior of the salient NSSS parameters following the SLBFP is presented in Figures 15.1.5-17 through 15.1.5-32. Table 15.1.5-2 summarizes the major events, times, and results for this transient.

At 4.68 seconds after the initiation of the steamline break, a trip
signal is initiated by the core protection calculators on variable
over power of 103%.

“Delete”

At 4.78 812
seconds, the reactor trip breakers open. At 10.98 seconds, voids begin
to form in the upper head of the reactor vessel. At 13.96 seconds,
the steam-generator pressure drops below the MSIS setpoint of 789 psia

(55.47 kg/cm^{2}A). The MSIS initiates closure of the MSIVs and MFIVs at 15.11 seconds. The “Delete” operable MSIVs and MFIVs close by 20.11 seconds and 25.11 seconds, respectively. At 120.62 seconds, the pressurizer 812 empties. At 136.32 seconds, auxiliary feedwater flow is assumed to be initiated to the affected steam generator. At 158.12 seconds, the pressurizer pressure drops below safety injection setpoint of 1555 psia (109.32 kg/cm^{2}A). Within 30 seconds of SIAS, the “Delete” HPSI pumps reach full speed and the HPSI valves are fully open. Safety injection boron begins to reach the core at 280.3 seconds. At 382.42 seconds, 812 the maximum core reactivity (-0.154% △p) occurs. The values of DNBR remains above 10 during the post-trip approach-to-criticality portion of this transient. At a maximum of 30 minutes the operator, via the appropriate emergency procedure, initiates plant cooldown by manual control of the turbine bypass valves. Shutdown cooling system operation is initiated when the RCS reaches 350 °F (176.7 °C) and 410 psia (28.82 kg/cm^{2}A).

<u>Case 3: Large Steamline Break During Zero Operation with Concurrent Loss-of-Offsite Power (SLBZPLOOP)</u>

The dynamic behavior of the salient NSSS parameters following the SLBZPLOOP is presented in Figures 15.1.5-33 through 15.1.5-48. Table 15.1.5-3 summarizes the major events, times, and results for this transient.

Concurrent with the steamline break, a loss-of-offsite power occurs. At this time an actuation signal for the emergency diesel generators is initiated. Due to decreasing core flow following loss of power to the reactor coolant pumps, conditions exist for a CPC trip. At 1.0 second, a low-reactor-coolant-pump-shaft-speed-trip signal is initiated by the core protection calculators.

“Delete”

At 1.1 seconds, the reactor trip breakers open. At 812 10.49 seconds, the steam-generator pressure drops below the MSIS setpoint of 789 psia (55.47 kg/cm^{2}A). The MSIS initiates closure of the MSIVs and MFIVs at 11.64 seconds. The MSIVs and MFIVs close by 16.64 seconds and 21.64 seconds, respectively. At 52.02 seconds, the pressure difference between the steam generators reaches the analysis setpoint of 352 psid (24.75 kg/cm^{2}D) for isolation of AFW to the ruptured steam generator. As a result, AFW to the ruptured steam generator is isolated. 812

The pressurizer empties at 68.16 seconds. At 72.18 seconds, the pressurizer pressure drops below safety injection setpoint of 1555 psia (109.32 kg/cm^{2}A). Within 30 seconds of SIAS, the operable HPSI pump is loaded on an emergency diesel generator and reaches full speed and the HPSI valves are fully open. At 89.76 seconds, voids begin to form in the upper head of the reactor vessel. Safety injection boron begins to reach the core at 178.52 seconds. At 267.69 seconds, the maximum core reactivity (-0.488% △p) occurs. The values of DNBR remain above 10 during this transient. At a maximum of 30 minutes the operator, via the appropriate emergency procedure, initiates plant cooldown by manual control of the atmospheric dump valves, assuming that offsite power has not been restored. Shutdown cooling system operation is initiated when the RCS reaches 350 °F (176.7 °C) and 410 psia (28.82 kg/cm^{2}A). 812 812 812

Case 4: Large Steamline Break Zero Power Operation with Offsite Power Available (SLBZP)

The dynamic behavior of the salient NSSS parameters following the SLBZP is presented in Figures 15.1.5-49 through 15.1.5-64. Table 15.1.5-4 summarizes the major events, times, and results of this transient.

At 10.90 seconds, after initiation of the steamline break, the steam generator pressure drops below the low steam-generator pressure trip and MSIS setpoint of 789 psia (55.47 kg/cm^{2}A). "Delete" 812

“Delete”
At 12.05 seconds, the reactor trip breakers open.
The MSIS initiates closure of the MSIVs and MFIVs at 12.05 seconds.
The MSIVs and MFIVs close by 17.05 seconds and 22.05 seconds
respectively. “Delete” The pressurizer empties at 65.18 seconds. At 68.46
seconds, the pressurizer pressure drops below safety injection
setpoint of 1555 psia (109.32 kg/cm^{2}A). Within 30 seconds of SIAS, the 812
operable HPSI pump reaches full speed and the HPSI valves are fully
open. At 69.24 seconds, the pressure difference between the steam
generators reaches the analysis setpoint of 352 psid (24.75 kg/cm2D)
for isolation of AFW to the ruptured steam generator. As a result,
AFW to the ruptured steam generator is isolated at 82.38 seconds voids
begin to form in the upper head of the reactor vessel. Safety
injection boron begins to reach the core at 178.13 seconds. At 292.92
seconds, the maximum core reactivity (-0.513% △p) occurs. The values 812
of DNBR remains above 10 during the post-trip approach-to-criticality
portion of this transient. At a maximum of 30 minutes the operator, 645
via the appropriate emergency procedure, initiates plant cooldown by
manual control of the MSIV bypass valves associated with the
unaffected steam generator and turbine bypass valves. Shutdown cooling
system operation is initiated when the RCS reaches 350 °F (176.7 °C)
and 410 psia (28.82 kg/cm^{2}A).

Case 5: Large Steamline Break Outside Containment During Full Power Operation with Offsite Power Available (SLBFPD)

The dynamic behavior of the salient NSSS parameters following a typical limiting SLBFPD is presented in Figures 15.1.5-65 through 15.1.5-73. Table 15.1.5-5 summarized the major events, times and results for this transient.

The consequences of this transient--fraction of fuel rods predicted to experience DNB--are the same for a spectrum of break sizes, due to the protective action of the core protection calculators(CPCs). See the discussion in Section 15C.3.2. The largest break size yields the minimum DNBR. Therefore, the transient presented here is that which results from the double-ended break of a main steamline.

At 6.96 seconds, a trip signal is initiated by the CPCs on variable 812
over power of 121%. Simultaneously, AFW flow is assumed to be 0
initiated to the affected steam generator. At 7.06 seconds the reactor trip breakers open. At 9.11 seconds, a minimum transient DNBR of 1.2440 is calculated to occur, after which DNBR rapidly increases, 812
as shown in Figure 15.1.5-73. At 13.54 seconds, the steam-generator pressure drops below the MSIS setpoint of 789 psia (55.47 kg/cm^{2}A).

At 14.45 seconds, voids begin to form in the upper head of the reactor vessel.

The MSIS initiates closure of the MSIVs and MFIVs at 14.69 seconds. The operable MSIVs and MFIVs close by 19.69 seconds and 24.69 812
seconds, respectively.

“Delete”

Subsequently, the events of this transient follow a sequence similar to those of the SLBFP (Case 2). Since the cooldown is less severe, the potential for post-trip degradation in fuel cladding performance is less for this case (SLBFPD) than for Case 2 (SLBFP). At a maximum of 30 minutes the operator, using the appropriate emergency procedure, initiates plant cooldown by manual control of the turbine bypass valves. Shutdown cooling system operation is initiated when the RCS reaches 350 °F (176.7 °C) and 410 psia (28.82 kg/cm^{2}A).

At the point of the minimum transient DNBR no more than 1.1% of the
fuel rods are “Delete” to experience DNB. All fuel pins which 812
experience DNB are conservatively assumed to fail. All of the 0
activity in the fuel gap for fuel rods that are assumed to fail is assumed to be uniformly mixed with the reactor coolant. The activity

in the fuel clad gap is assumed to be 8% of I-131, 10% of Kr-85 and 5% of 741
other iodines and noble gases accumulated in the fuel at the end of core life,
assuming continuous full power operation. This results in a primary
coolant activity of 758 μCi/gm dose equivalent I131. Assuming 0.5 gpm 812
(1.89 L/min) steam generator tube leakage, during a period of 2 hours
after initiation of the SLBFPD, the integral leakage from the RCS
through the affected steam generator is 500 lbm (227 kg), which is 741
assumed to be released to the atmosphere with a DF of 1. This mass
release results in a contribution to the inhalation thyroid dose at
the exclusion area boundary (EAB) of 45.9 rem. 812

The total steam released from the affected steam generator is 850,900 812
lbm (385,962 kg), which includes the steam release through affected steam
generator during 30 minutes plus the steam flow used to remove the decay heat
and sensible heat by unaffected steam generator supply for 2 hours after
initiation of the event. The affected steam generator will empty in
2 hours; therefore all the mass release from the affected steam
generator to the atmosphere has a DF of 1. The calculated inhalation
thyroid dose is not more than 8.2 rem for the blowdown originating 812
from the secondary system fluid discharge from the affected steam
generator.

Less than 127,300 lbm (57,742 kg) of steam from the unaffected steam 812
generator will be released through the steamline break. During the
SLBFPD the MSIVs will isolate the unaffected steam generator and
prevent it from emptying. Therefore, a DF of 100 is assumed in
calculating iodine activity released from the unaffected steam
generator. The resulting contribution to the inhalation thyroid dose
at the EAB is less than 12 mrem. 812

The foregoing doses are calculated by the methods outlined in Subsection 15.0.4. Table 15.1.5-11 presents the major assumptions, parameters, and calculational methods used to evaluate the radiological consequences for this transient.

In summary, the total 2-hour inhalation thyroid dose at the EAB as a consequence of the SLBFPD is no more than 54.2 rem. 812

Case 6: Large Steamline Break Outside Containment from Zero Power Operation with Loss-of-Offsite Power (SLBZPLOOPD)

424

Case 6 is included in Case 3, since the break of the latter can be either inside or outside of containment. The figures, tables, and discussion for Case 3 apply to Case 6.

Assuming 0.5 gpm (1.89 L/min) steam-generator tube leakage, during a period of 2 hours after initiation of the SLBZPLOOPD the integral leakage from the RCS through the affected steam generator is 500 lbm (227 kg), which is assumed to be released to the atmosphere with a DF of 1. This mass release results in a contribution to the inhalation thyroid doses at the EAB of:

1. 0.0484 rem, assuming technical specification primary coolant activity;

2. 2.91 rem, assuming a preaccident iodine spike; or

812

3. 4.21 rem, assuming an event generating iodine spike.

The total steam released from the affected steam generator is 921,900 lbm (418,167 kg), which includes the steam release through affected steam generator during 30 minutes plus the steam flow used to remove the decay heat and sensible heat by unaffected steam generation for 2 hours after initiation of the event. The affected steam generator 424 will empty in 2 hours; therefore, all the mass release from the affec3 ted steam generator to atmosphere has a DF of 1.

The calculated inhalation thyroid dose is 8.92 rem for the blowdown 812
steam originating from the affected steam generator.

Less than 50,200 lbm (22,771 kg) of steam from the unaffected steam 812
generator will be released through the atmospheric dump valves and
through the steamline break within 2 hours. During the SLBZPLOOPD the
MSIVs will isolate the unaffected steam generator and prevent it from 424
emptying. Therefore, a DF of 100 is assumed in calculating iodine
activity released from the unaffected steam generator. The resulting
contribution to the inhalation thyroid dose at the EAB is 0.00486 rem. 812

The foregoing doses are calculated by the methods outlined in
Subsection 15.0.4. Table 15.1.5-11 presents the major assumptions,
parameters, and calculational methods used to evaluate the 424
radiological consequences for this transient.

In summary, the total 2-hour inhalation thyroid dose at the EAB as a
consequence of the SLBZPLOOPD is no more than 13.1 rem. 812

15.1.5.4 Conclusions

For the large steamline break in combination with a single failure and stuck CEA, with or without a loss-of-offsite power, fission power remains sufficiently low following reactor trip to preclude fuel damage as a result of post-trip return to power.

For a large steamline break during zero power operation in combination with a loss of offsite power and Technical Specification tube leakage, the 2-hour inhalation thyroid dose at the EAB is well within 10 CFR 100 guidelines:

a. 8.97 rem, assuming Technical Specification primary coolant activity;

b. 11.8 rem, assuming a preaccident iodine spike; or 812

c. 13.1 rem, assuming an event generated iodine spike.

The maximum potential for radiological releases due to fuel failure occurs for full power steamline breaks outside containment in combination with a stuck CEA. For these cases the maximum potential for degradation in fuel cladding 424 performance occurs prior to and during reactor trip. The fraction of fuel predicted to experience DNB for these events is no more than 1.1%. With the assumption of 0.5 gpm (1.89 L/min) steam generator tube leakage and a 741 assumption of 1.1% fuel failure the 2-hour inhalation thyroid dose at the EAB is calculated to be no more than 54.2 rem, which is well within the 10 CFR 100 812 guideline value.

The total body gamma dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0- to 2-hour dose at the EAB and for the duration of the accident at the outer boundary of the LPZ. The results are listed in Table 15.1.5-12.

Potential fuel failure is sufficiently limited to ensure that the core will remain in place and intact with no loss of core cooling capabilities.

TABLE 15.1.5-1

SEQUENCE OF EVENTS FOR A LARGE STEAMLINE BREAK DURING FULL POWER OPERATION WITH CONCURRENT LOSS-OF-OFFSITE POWER (SLBFPLOOP)

Time (sec)	Event	Setpoint or Value
0.0	Steamline break and loss-of-offsite power occur	--
1.0	CPC trip signal generated, RCP shaft speed (%) AFW is assumed to be initiated to the ruptured steam generators	95
1.1	Trip breakers open	--
8.82	Voids begin to form in RV upper head	--
9.97	Steam generator pressure reaches main steam isolation signal (msis) analysis setpoint, psia (kg/cm^2a)	789 (55.47)
16.12	MSIVs close completely	--
21.12	MFIVs close completely	--
35.88	Isolation of AFW to affected SG due to high △P, psid (kg/cm^2D)	352 (24.75)
195.31	Pressurizer empties	--
312.62	Pressurizer pressure reaches safety injection actuation signal (SIAS) analysis setpoint, psia (kg/cm^2A)	1555 (109.32)
343.77	Safety injection flow begins	--
519.78	Safety injection boron begins to reach reactor core	--
662.42	Maximum transient reactivity, $10^{-2}\,\Delta\rho$	-0.131
1800	Operator initiates cooldown	--

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TABLE 15.1.5-2

SEQUENCE OF EVENTS FOR A LARGE STEAMLINE BREAK DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFP)

Time (sec)	Event	Setpoint or Value	
0.0	Steamline break occurs	--	
4.68	CPC variable over power trip signal generated, percent of full power (%), "Delete"	103	812
4.78	Trip breakers open	--	812
10.98	Voids begin to form in RV upper head	--	812
13.96	Steam generator pressure reaches main steam isolation signal analysis setpoint, psia (kg/cm^2A)	789 (55.47)	812
20.11	MSIVs close completely	--	812
25.11	MFIVs close completely	--	812
120.62	Pressurizer empties	--	812
136.32	AFW is assumed to be initiated to the affected SG	--	812
158.12	Pressurizer pressure reaches safety injection actuation signal analysis setpoint, psia (kg/cm^2A)	1555 (109.32)	812
189.27	Safety injection flow begin	--	812
280.3	Safety injection boron begins to reach reactor core	--	812
382.42	Maximum transient reactivity, $10^{-2}\triangle\rho$	-0.154	
	"delete"		645
1800	Operator initiates cooldown	--	

TABLE 15.1.5-3

SEQUENCE OF EVENTS FOR A LARGE STEAMLINE BREAK DURING ZERO POWER OPERATION WITH CONCURRENT LOSS-OF-OFFSITE POWER (SLBZPLOOP AND SLBZPLOOPD)

Time (sec)	Event	Setpoint or Value	
0.0	Steamline break and loss-of-offsite power occur.	--	
1.0	CPC trip signal generated, RCP shaft speed (%)	95	
	"Delete"		812
1.1	Trip breakers open	--	
10.49	Steam generator pressure reaches main steam isolation signal analysis setpoint, psia (kg/cm^2A)	789 (55.47)	
16.64	MSIVs close completely	--	
21.64	MFIVs close completely	--	
52.02	Isolation of AFW to affected SG due to high △P, psid (kg/cm^2D)	352 (24.75)	
68.16	Pressurizer empties	--	812
72.18	Pressurizer pressure reaches safety injection actuation signal analysis setpoint, psia (kg/cm^2A)	1555 (109.32)	
89.76	Voids begin to form in RV upper head	--	
103.33	Safety injection flow begins	--	
178.52	Safety injection boron begins to reach reactor core	--	
267.69	Maximum transient reactivity, 10^{-2} △ρ	-0.488	
1800	Operator initiates cooldown	--	

TABLE 15.1.5-4

SEQUENCE OF EVENTS FOR A LARGE STEAMLINE BREAK DURING ZERO POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBZP)

Time (sec)	Event	Setpoint or Value	
0.0	Steamline break occur.	--	
10.90	Steam generator pressure reaches reactor trip analysis setpoint, psia (kg/cm^2A), and main steam isolation signal analysis setpoint. “Delete”	789 (55.47)	
11.95	Low steam generator pressure reactor trip signal generated	--	
12.05	Trip breakers open	--	
17.05	MSIVs close completely	--	
22.05	MFIVs close completely	--	
65.18	Pressurizer empties	--	812
68.46	Pressurizer pressure reaches safety injection actuation signal analysis setpoint, psia (kg/cm^2A)	1555 (109.32)	
69.24	Isolation of AFW to affected SG due to high $\triangle P$, psid (kg/cm^2D)	352 (24.75)	
82.38	Voids begin to form in RV upper head	--	
99.61	Safety injection flow begins	--	
178.13	Safety injection boron begins to reach reactor core	--	
292.92	Maximum transient reactivity, $10^{-2} \triangle \rho$	-0.513	
1800	Operator initiates cooldown	--	

TABLE 15.1.5-5

SEQUENCE OF EVENTS FOR A LARGE STEAMLINE BREAK OUTSIDE CONTAINMENT DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFPD)

Time (sec)	Event	Setpoint or Value
0.0	Steamline break occurs	--
6.96	CPC variable over power trip signal generated, percent of full power (%), AFW is assumed to be initiated to the ruptured SG	121
7.06	Trip breakers open	--
9.11	Minimum transient DNBR	1.2440
13.54	Steam generator pressure reaches main steam isolation signal analysis setpoint, psia (kg/cm^2A)	789 (55.47)
14.45	Voids begin to form in RV upper head	--
19.69	MSIVs close completely	--
24.69	MFIVs close completely	--
85.05	Pressurizer pressure reaches safety injection actuation signal (SIAS) analysis setpoint, psia (kg/cm^2A)	1555 (109.32)
116.20	Safety injection flow begins	--
226.15	Safety injection boron begins to reach reactor core	--
327.11	Maximum transient reactivity, $10^{-2}\triangle\rho$	-3.239
1800	Operator initiates cooldown	--

812

TABLE 15.1.5-6

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAMLINE BREAK DURING FULL POWER OPERATION WITH CONCURRENT LOSS-OF-OFFSITE POWER (SLBFPLOOP)

Parameter		Assumed Value	
Initial Core Power Level, MWt		2871.3	
Initial Core Inlet Coolant Temperature, °F(°C)		570 (298.9)	
Initial Core Mass Flow Rate, 10^6 lbm/hr (kg/hr)		111.2 (50.44)	
Initial Pressurizer Pressure, psia (kg/cm^{2}A)		2325 (163.45)	
Initial Pressurizer Water Volume, ft^3 (m^3)		1030 (29.17)	
Doppler Coefficient		Most Negative (Fig.15.1.5-74)	
Moderator Coefficient		Most Negative (Fig.15.1.5-75)	
Axial Shape Index		+0.3	
CEA Worth for Trip, $10^{-2}\triangle p$		-9.4	
Initial Steam Generator Inventory, lbm (kg),			812
	Affected	194,467 (88,209)	
	Intact	194,467 (88,209)	
One High-Pressure Safety Injection Pump		Inoperative	
Core Burnup		End of Cycle	
Blowdown Fluid		Saturated Steam	
Blowdown Area for Each Steamline, ft^2 (m^2)		0.942 (0.0875)	

"Delete" 812

Amendment 812
2018.05.30

TABLE 15.1.5-7

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAMLINE BREAK DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFP)

Parameter	Assumed Value	
Initial Core Power Level, MWt	2871.3	
Initial Core Inlet Coolant Temperature, °F(°C)	570 (298.9)	
Initial Core Mass Flow Rate, 10^6 lbm/hr (kg/hr)	111.2 (50.44)	
Initial Pressurizer Pressure, psia (kg/cm^{2}A)	2325 (163.45)	
Initial Pressurizer Water Volume, ft^3 (m^3)	1030 (29.17)	
Doppler Coefficient	Most Negative (Fig.15.1.5-74)	
Moderator Coefficient	Most Negative (Fig.15.1.5-75)	
Axial Shape Index	+0.3	
CEA Worth for Trip, 10^{-2} $\triangle$p	-9.4	
Initial Steam Generator Inventory, lbm (kg),		
Affected	194,467 (88,209)	812
Intact	194,467 (88,209)	
One Main Steam Isolation Valve on Intact Steam Generator	Inoperative	
Core Burnup	End of Cycle	
Blowdown Fluid	Saturated Steam	
Blowdown Area for Each Steamline, ft^2 (m^2)	0.942 (0.0875)	

TABLE 15.1.5-8

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAMLINE BREAK DURING ZERO POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBZPLOOP AND SLBZPLOOPD)

Parameter		Assumed Value	
Initial Core Power Level, MWt		10	
Initial Core Inlet Coolant Temperature, °F(°C)		572 (300)	
Initial Core Mass Flow Rate, 10^6 lbm/hr (kg/hr)		110.8 (50.26)	
Initial Pressurizer Pressure, psia (kg/cm^{2}A)		2325 (163.45)	
Initial Pressurizer Water Volume, ft^3 (m^3)		1030 (29.17)	
Doppler Coefficient		Most Negative (Fig.15.1.5-74)	
Moderator Coefficient		Most Negative (Fig.15.1.5-75)	812
Axial Shape Index		+0.6	812
CEA Worth for Trip, $10^{-2}\triangle p$		-6.0	812
Initial Steam Generator Inventory, lbm (kg),			812
	Affected	282,934 (128,337)	812
	Intact	282,934 (128,337)	812
One High-Pressure Safety Injection Pump		Inoperative	
Core Burnup		End of Cycle	
Blowdown Fluid		Saturated Steam	
Blowdown Area for Each Steamline, ft^2 (m^2)		0.942 (0.0875)	

“Delete” 812

TABLE 15.1.5-9

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAMLINE BREAK DURING ZERO POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBZP)

Parameter		Assumed Value	
Initial Core Power Level, MWt		10	
Initial Core Inlet Coolant Temperature, °F(°C)		572 (300)	
Initial Core Mass Flow Rate, 10^6 lbm/hr (kg/hr)		110.8 (50.26)	
Initial Pressurizer Pressure, psia (kg/cm^{2}A)		2325 (163.45)	
Initial Pressurizer Water Volume, ft^3 (m^3)		1030 (29.17)	
Doppler Coefficient		Most Negative (Fig.15.1.5-74)	
Moderator Coefficient		Most Negative (Fig.15.1.5-75)	812
Axial Shape Index		+0.6	812
CEA Worth for Trip, 10^{-2} △p		-6.0	812
Initial Steam Generator Inventory, lbm (kg),			812
	Affected	282,934 (128,337)	812
	Intact	282,934 (128,337)	812
One High-Pressure Safety Injection Pump		Inoperative	
Core Burnup		End of Cycle	
Blowdown Fluid		Saturated Steam	
Blowdown Area for Each Steamline, ft^2 (m^2)		0.942 (0.0875)	

“Delete” 812

TABLE 15.1.5-10

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAMLINE BREAK OUTSIDE CONTAINMENT DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFPD)

Parameter	Assumed Value	
Initial Core Power Level, MWt	2871.3	
Initial Core Inlet Coolant Temperature, °F(°C)	570 (298.9)	
Initial Core Mass Flow Rate, 10^6 lbm/hr (kg/hr)	135.7 (61.57)	
Initial Pressurizer Pressure, psia (kg/cm^{2}A)	2325 (163.45)	
Initial Pressurizer Water Volume, ft^3 (m^3)	1030 (29.17)	
Doppler Coefficient	Least Negative (Fig.15.1.5-74)	
Moderator Coefficient	Most Negative (Fig.15.1.5-76)	
Axial Shape Index		
Scram Rod Worth Insertion Curve	+0.4	645
Minimum DNBR	+0.3	645
Radial Peaking Factor, F_R	2.2909	812
CEA Worth for Trip, $10^{-2}\triangle\rho$	-9.4	812
Initial Steam Generator Inventory, lbm (kg),		812
Affected	87,522 (39,699)	812
Intact	87,522 (39,699)	812
One Main Steam Isolation Valve on Inlet Steam Generator	Inoperative	
Core Burnup	End of Cycle	
Blowdown Fluid	Saturated Steam	
Blowdown Area for Each Steamline, ft^2 (m^2)	0.942 (0.0875)	

TABLE 15.1.5-11 (Sh. 1 of 4)

PARAMETERS USED IN EVALUATING THE 2-HOUR RADOLOGICAL CONSEQUENCES OF STEAMLINE BREAKS OUTSIDE CONTAINMENT UPSTREAM OF MSIV

Parameter	Value: SLBFPD (Case 5)	Value: SLBZPLOOPD (Case 6)	
A. Data and Assumptions Used to Evaluate the Radioactive Source Term			
a. Power Level, MWt	2871.3	10	
b. Burnup, MWD/MTU	82,000	82,000	
c. Percent of Fuel Assumed to Experience DNB, %	1.1	0	83
d. Reactor Coolant Activity Before Event, μCi/gm, dose eq. I-131	0.8	0.8	812
e. Secondary System Activity Before Event μCi/gm, dose eq. I-131	0.08	0.08	812
f. Primary System Liquid Inventory, lbm (kg)	495,230 (224,636)	500,870 (227,195)	812
g. Steam Generator Inventory, lbm (kg)			812
- Affected Steam Generator	84,685 (38,413)	282,934 (128,339)	812
- Intact Steam Generator	84,685 (38,413)	282,934 (128,339)	812

See footnotes on last page of table.

TABLE 15.1.5-11 (Sh. 2 of 4)

Parameter	Value: SLBFPD (Case 5)		Value: SLBZPLOOPD (Case 6)		
B. Data and Assumptions Used to Evaluate Activity Released from the SECONDARY System	0-2 hours	0-36 hours	0-2 hours	0-36 hours	
a. Primary to Secondary Leak Rate, gpm (L/min)	0.5 (1.8927)	0.5[1] (1.8927)	0.5 (1.8927)	0.5[1] (1.8927)	812
b. Total Mass Release from the Affected Steam Generator, lbm (kg)	850,900 (385,962)	1,480,800 (671,680)	921,900 (418,167)	1,492,600 (677,032)	812
c. Total Mass Release from the Intact Steam Generator, lbm (kg)	127,300 (57,742)	127,300 (57,742)	50,200 (22,771)	50,200 (22,771)	
d. Reactor Coolant System Activity After Event, Ci					
Isotope					
I-131		1.36E+05			
I-132		1.20E+05			
I-133		1.74E+05			
I-134		1.99E+05			
I-135		1.64E+05			
Kr-85m		3.28E+04			
Kr-85		2.37E+03			741
Kr-87		6.62E+04			
Kr-88		9.35E+04			
Xe-131m		9.47E+02			
Xe-133		1.73E+05			
Xe-135		5.65E+04			
Xe-138		1.62E+05			

1) After 8hr, the primary to secondary leak rate is 0.25gpm (0.94635L/M) 812

TABLE 15.1.5-11 (Sh. 3 of 4)

Parameter	Value SLBFPD (Case 5)	SLBZPLOOPD (Case 6)	
e. Percent of Core Fission Products Assumed Released to Reactor Coolant	I-131 8 Kr-85 10 Other Nuclides 5	***	741
f. Iodine Decontamination Factor in the Affected Steam Generator	1.0	1.0	
g. Iodine Decontamination Factor in the Intact Steam Generator	100	100	
h. Credit for Radioactive Decay in Transit to Doss Point	No	No	
i. Loss-of-offsite Power	No	Yes	
C. Dispersion Data			
1. Distance to Exclusion Area Boundary, m	700	700	
2. Distance to Low Population Zone Outer Boundary, m	5633	5633	
3. Atmospheric Dispersion Factor, sec/m^3			
- At EAB	7.050×10^{-4}	7.050×10^{-4}	741
- At LPZ Outer Boundary	4.468×10^{-5}	4.468×10^{-5}	741

TABLE 15.1.5-11 (Sh. 4 of 4)

Parameter	SLBFPD (Case 5)	SLBZPLOOPD (Case 6)
D. Dose Data		
1. Method of Dose Calculation	Subsection 15.0.4	Subsection 15.0.4
2. Dose Conversion Assumptions	Subsection 15.0.4	Subsection 15.0.4

NOTE:

(1) Except for case assuming preaccident iodine spike (see note 2).

(2) The following three sub-cases are presented:

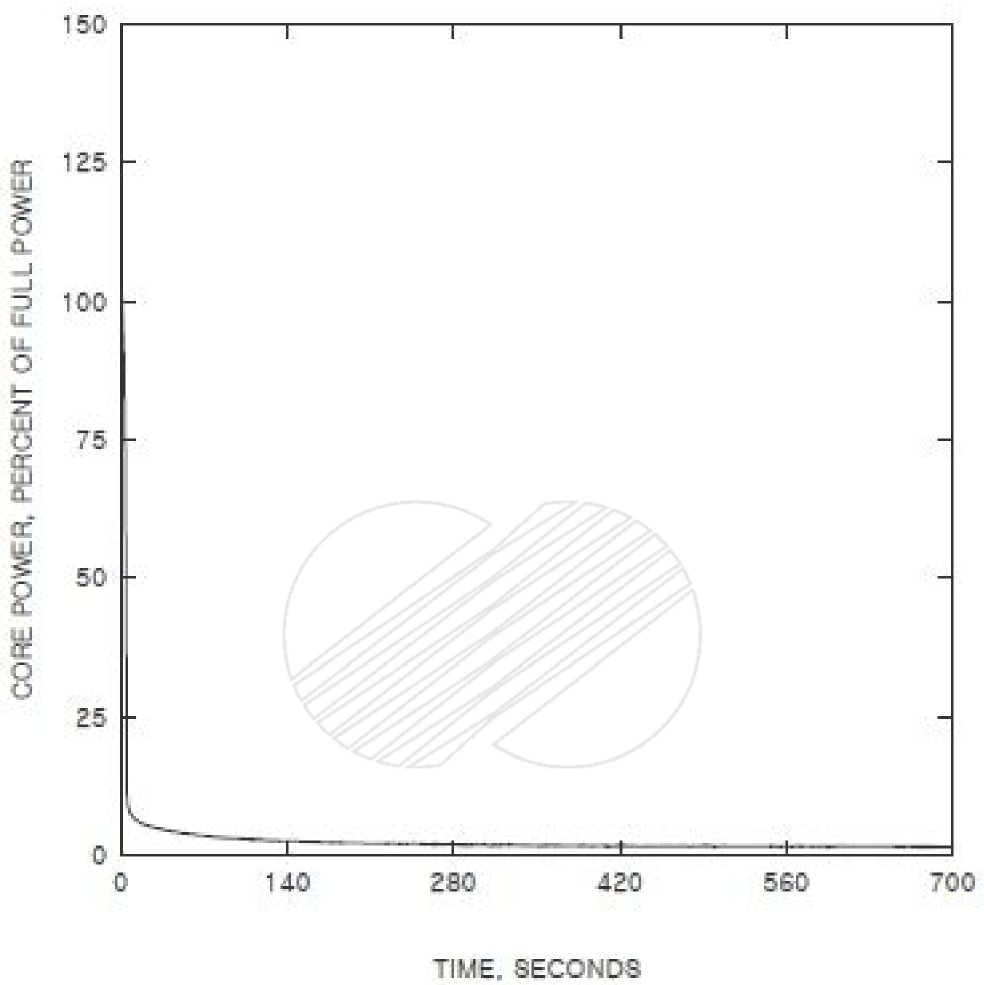
Sub-case	RCS activity after event, μCi/gm	
a) Technical Specification activity	0.8	812
b) Preaccident iodine spike (PIS)	48	
c) Event generated iodine spike (GIS)	550 (Max. activity at 8 hour after event)	

* Numbers in parenthesis refer to the power of ten; e.g., 1.034 (+5) = 1.034 × 10^5

TABLE 15.1.5-12

RADIOLOGICAL CONSEQUENCES OF STEAM PIPING FAILURES

Result	SLBFPD 1.1% Failed Fuel	SLBZPLOOPD Tech. Spec.	SLBZPLOOPD PIS	SLBZPLOOPD GIS	
					83
Exclusion Area Boundary Dose (0-2 hr), rem					
Thyroid	5.42E+01	8.97E+00	1.18E+01	1.31E+01	
Whole-body	2.37E-01	1.40E-02	1.84E-02	2.76E-02	
LPZ Boundary Dose (0-36 hr), rem					812
Thyroid	1.86E+01	9.34E-01	2.03E+00	9.45E+00	
Whole-body	1.11E-01	1.58E-03	3.73E-03	4.09E-02	



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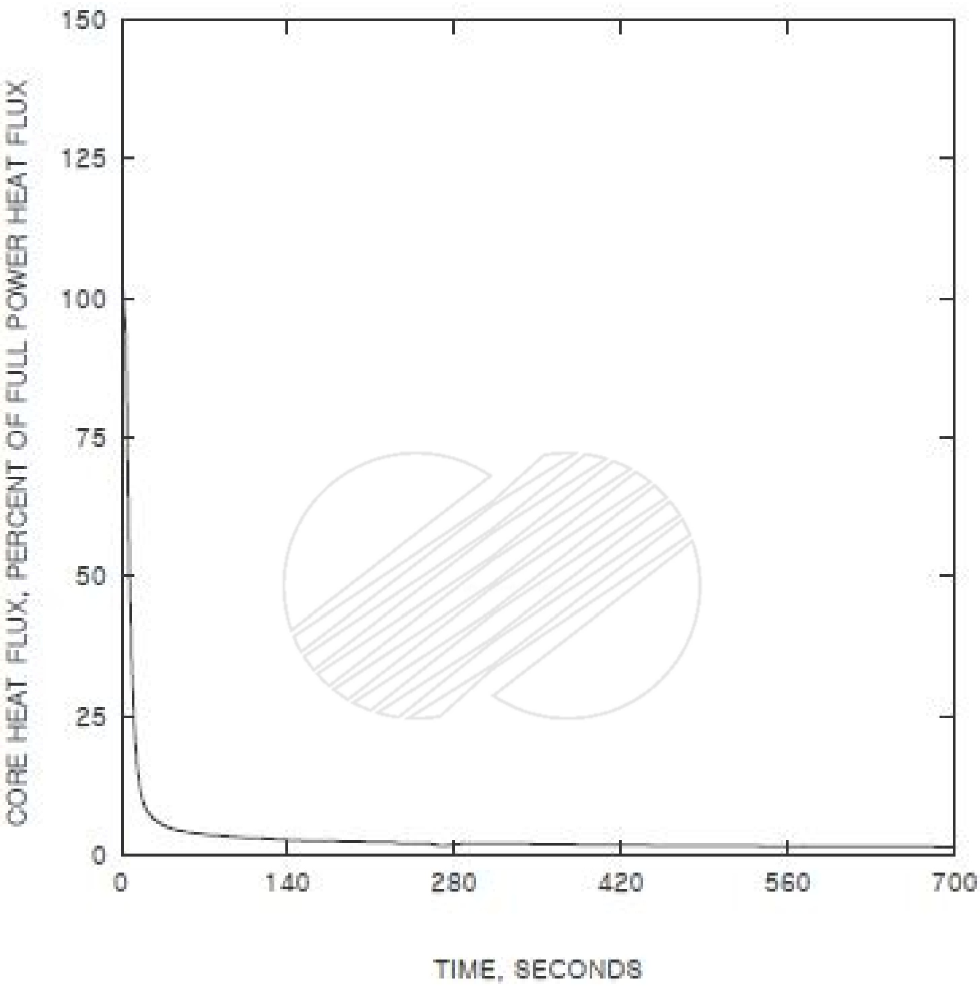
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
CORE POWER VS. TIME

Figure 15.1.5-1



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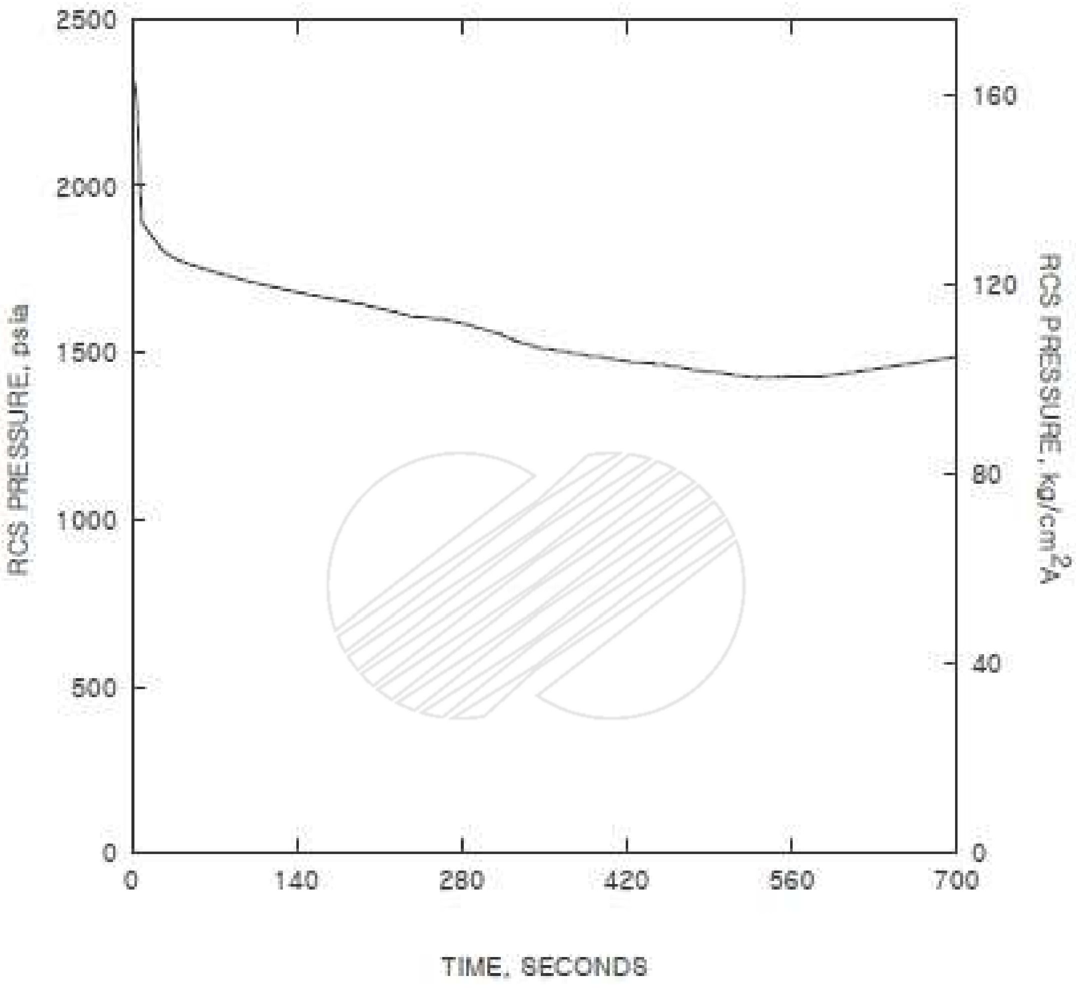
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
CORE HEAT FLUX VS. TIME

Figure 15.1.5-2



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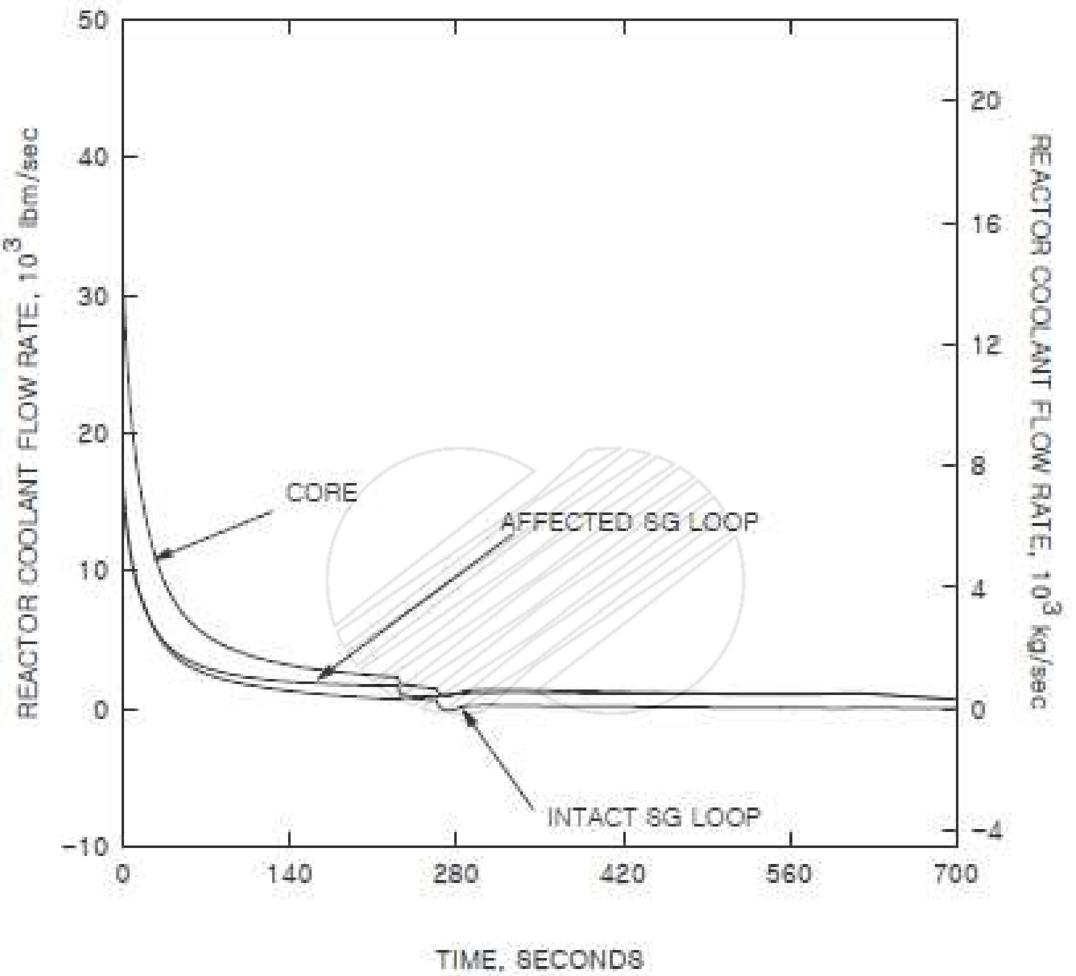
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
RCS PRESSURE VS. TIME

Figure 15.1.5-3



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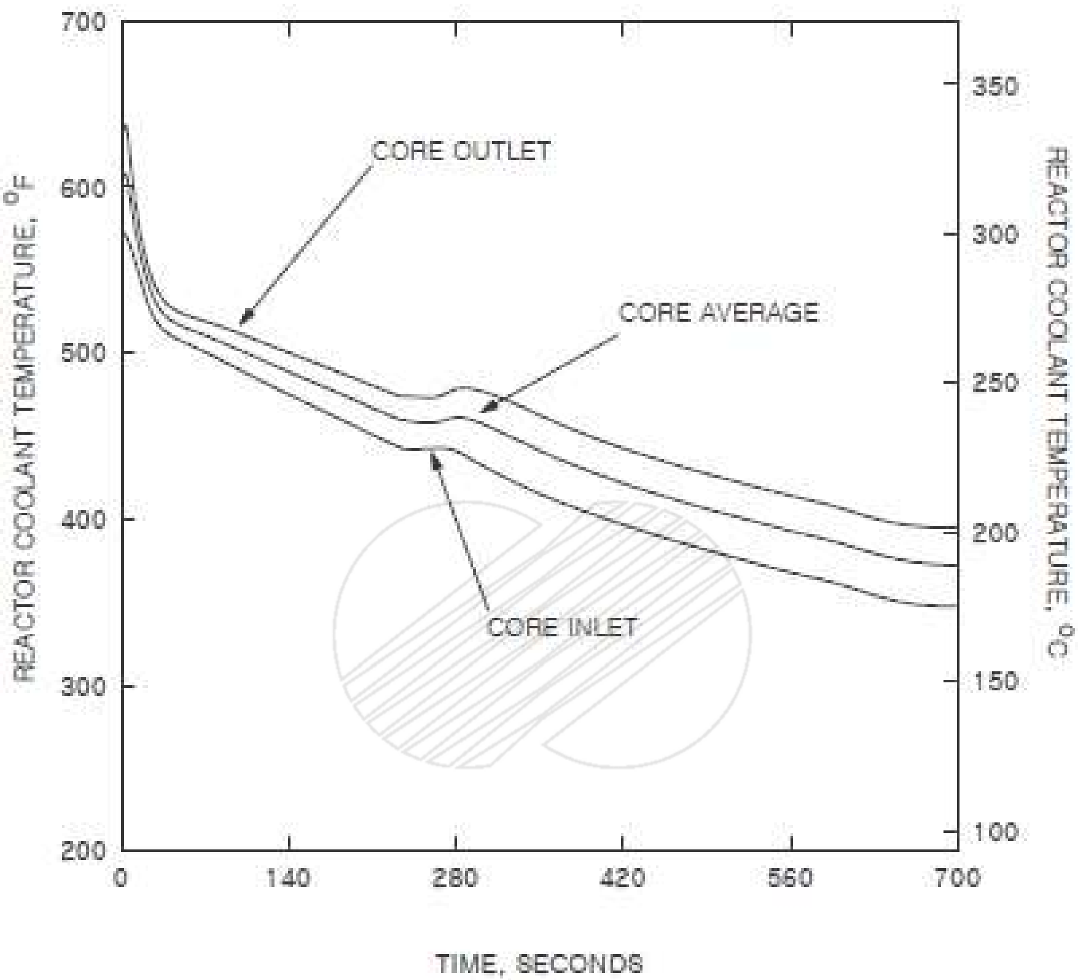
FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS-OF-OFFSITE POWER: REACTOR COOLANT FLOW RATE VS. TIME Figure 15.1.5-4	



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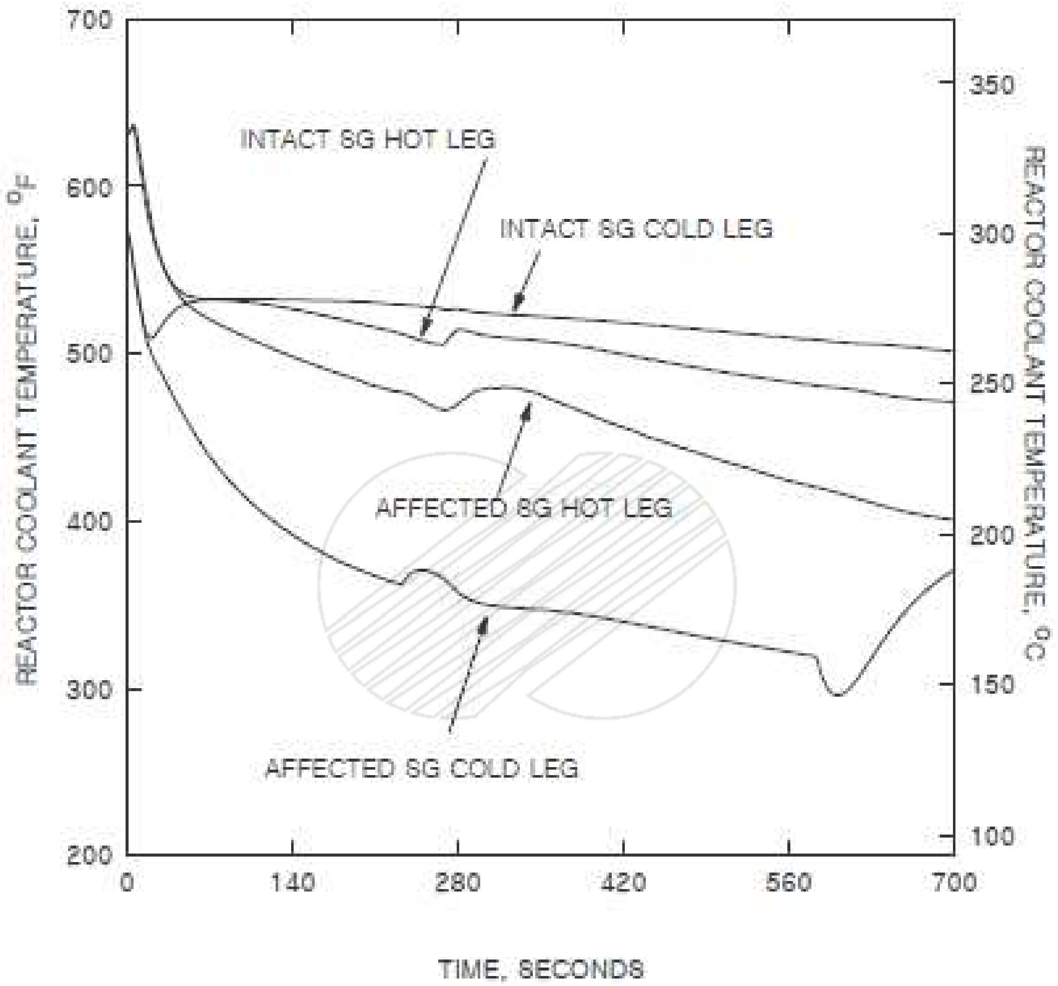
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
REACTOR COOLANT TEMPERATURES (A) VS. TIME

Figure 15.1.5-5



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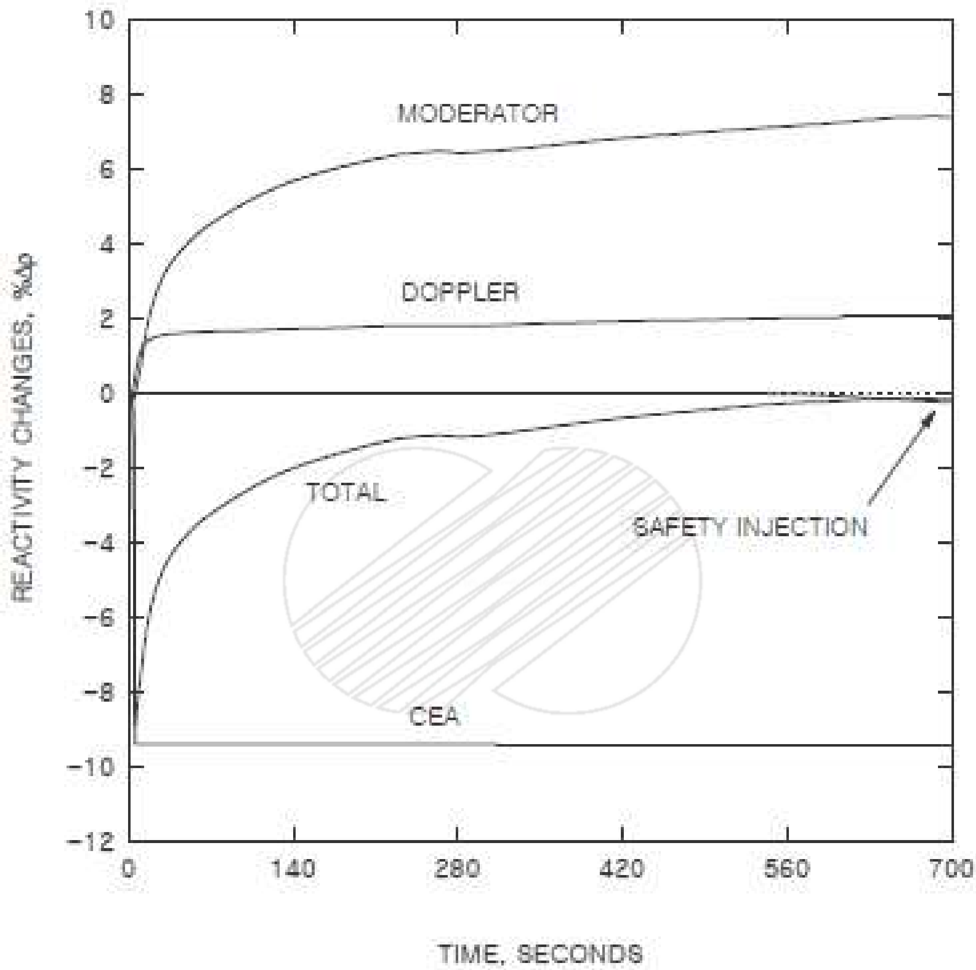
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
REACTOR COOLANT TEMPERATURES (B) VS. TIME

Figure 15.1.5-6



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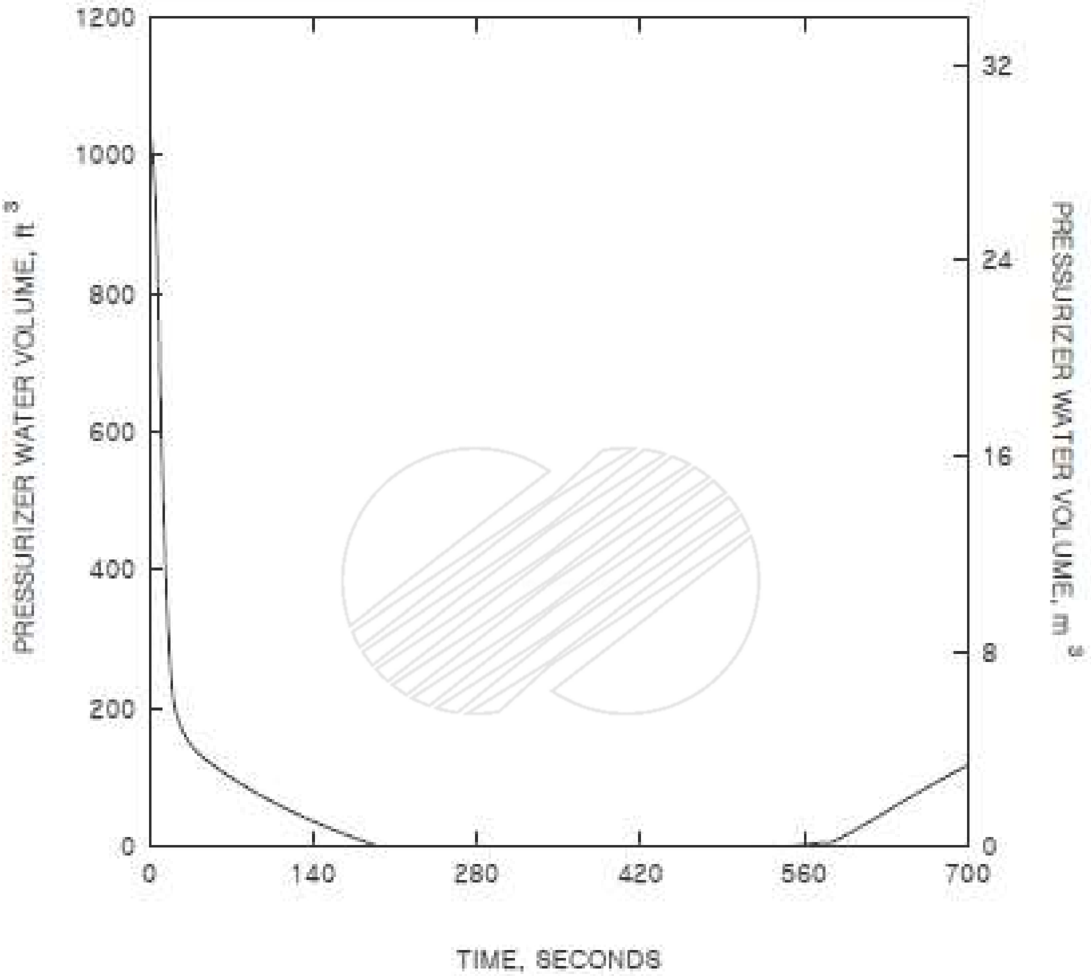
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
REACTIVITY CHANGES VS. TIME

Figure 15.1.5-7



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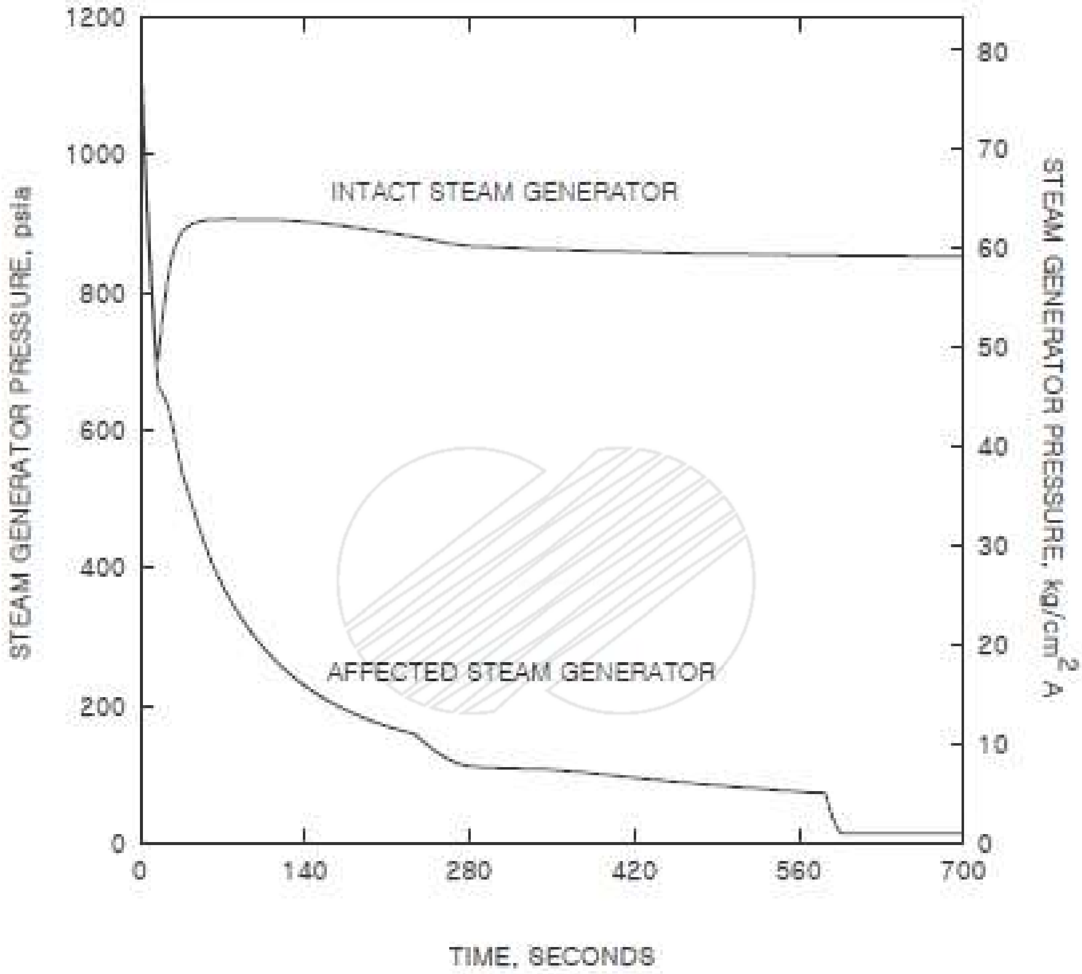
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
PRESSURIZER WATER VOLUME VS. TIME

Figure 15.1.5-8



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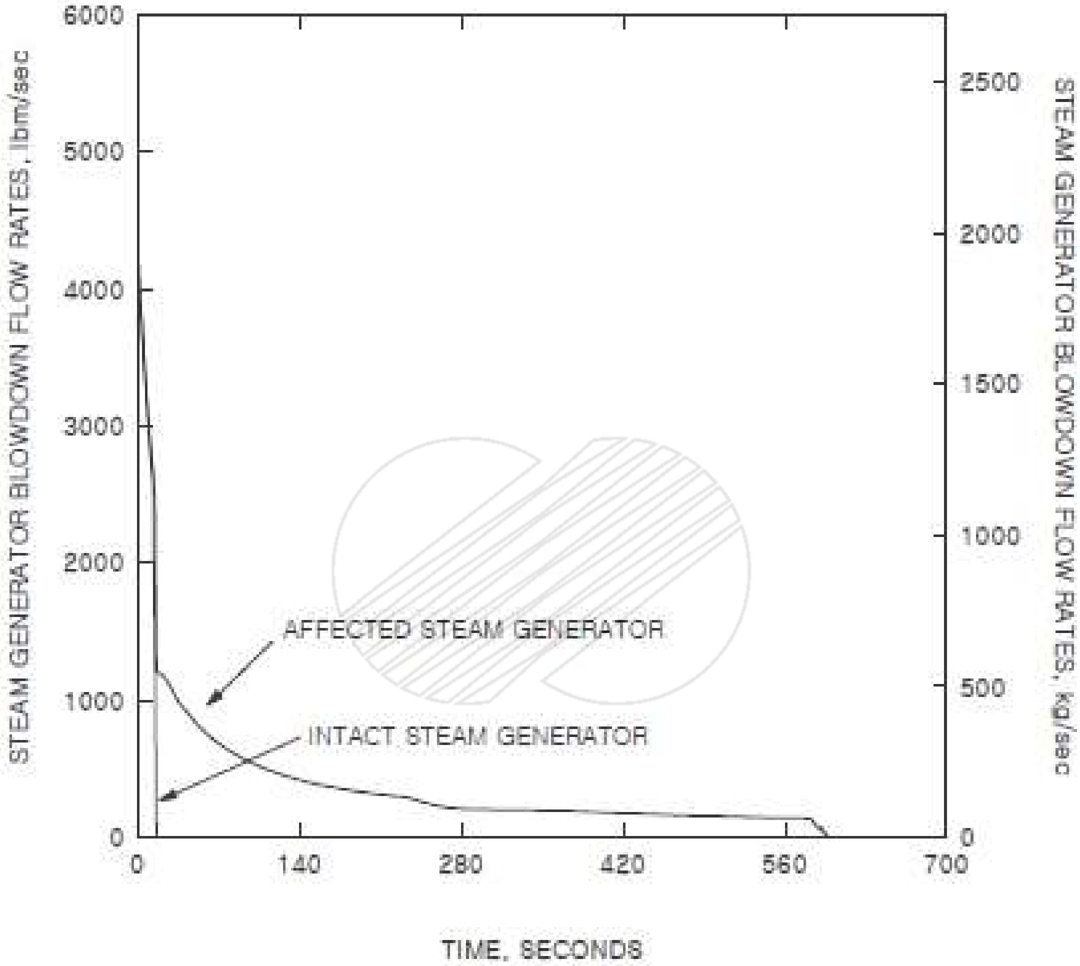
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
STEAM-GENERATOR PRESSURES VS. TIME

Figure 15.1.5-9



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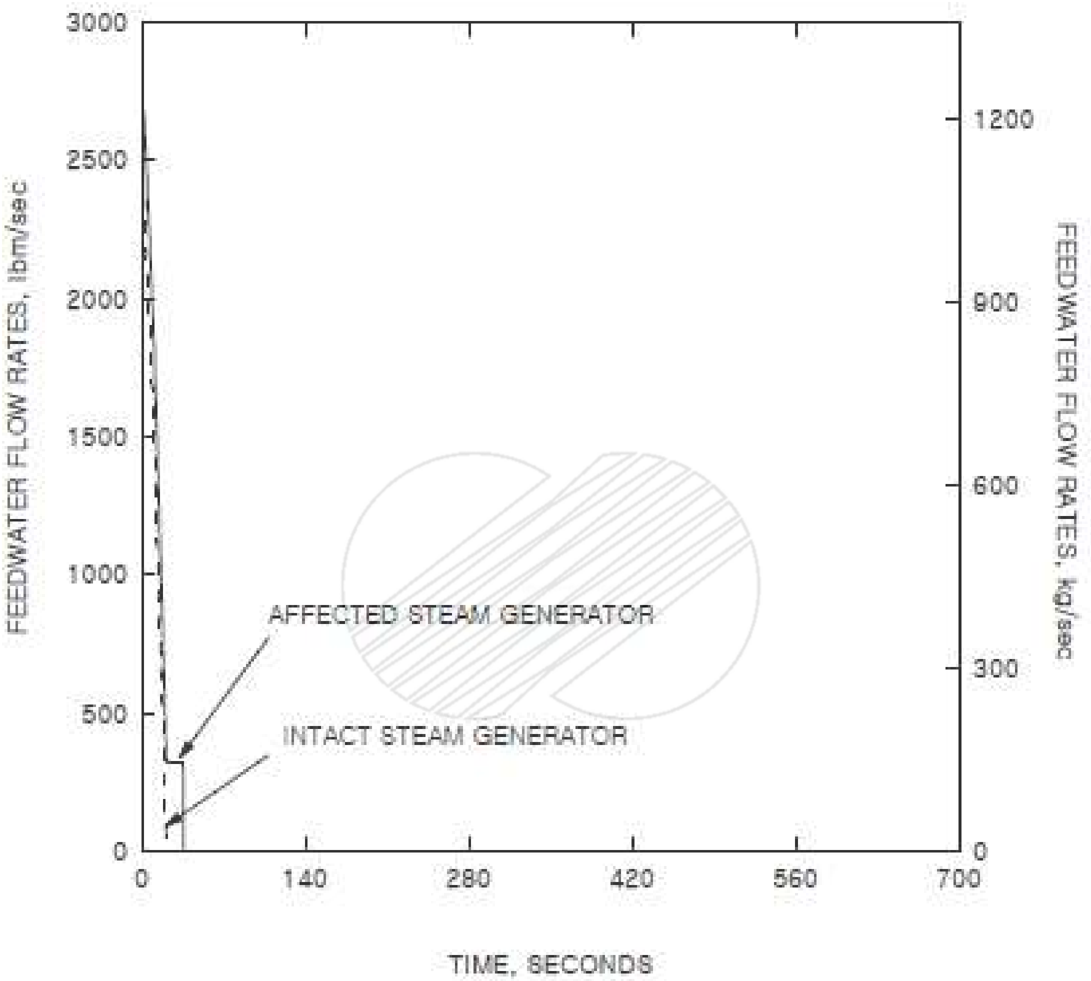
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
STEAM-GENERATOR BLOWDOWN RATES VS. TIME

Figure 15.1.5-10



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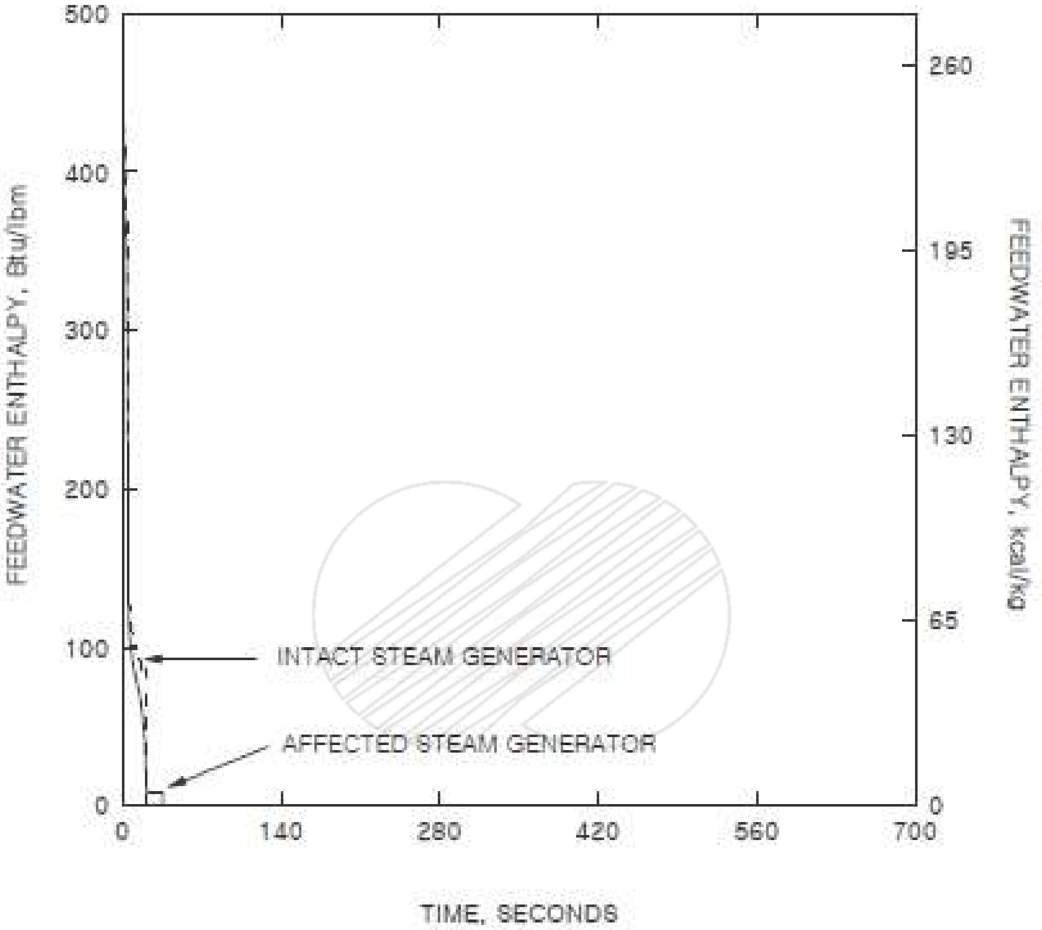
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
FEEDWATER FLOW RATES VS. TIME

Figure 15.1.5-11



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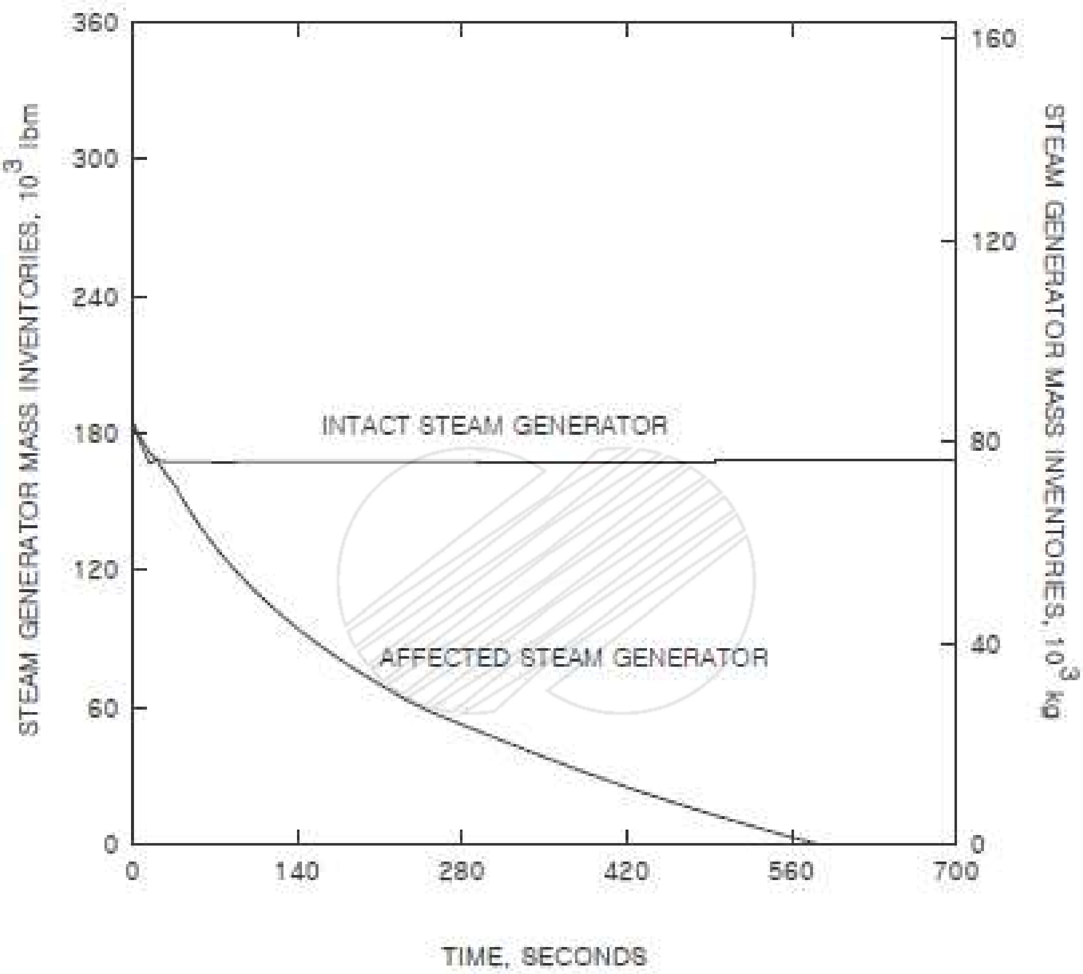
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
FEEDWATER ENTHALPY VS. TIME

Figure 15.1.5-12



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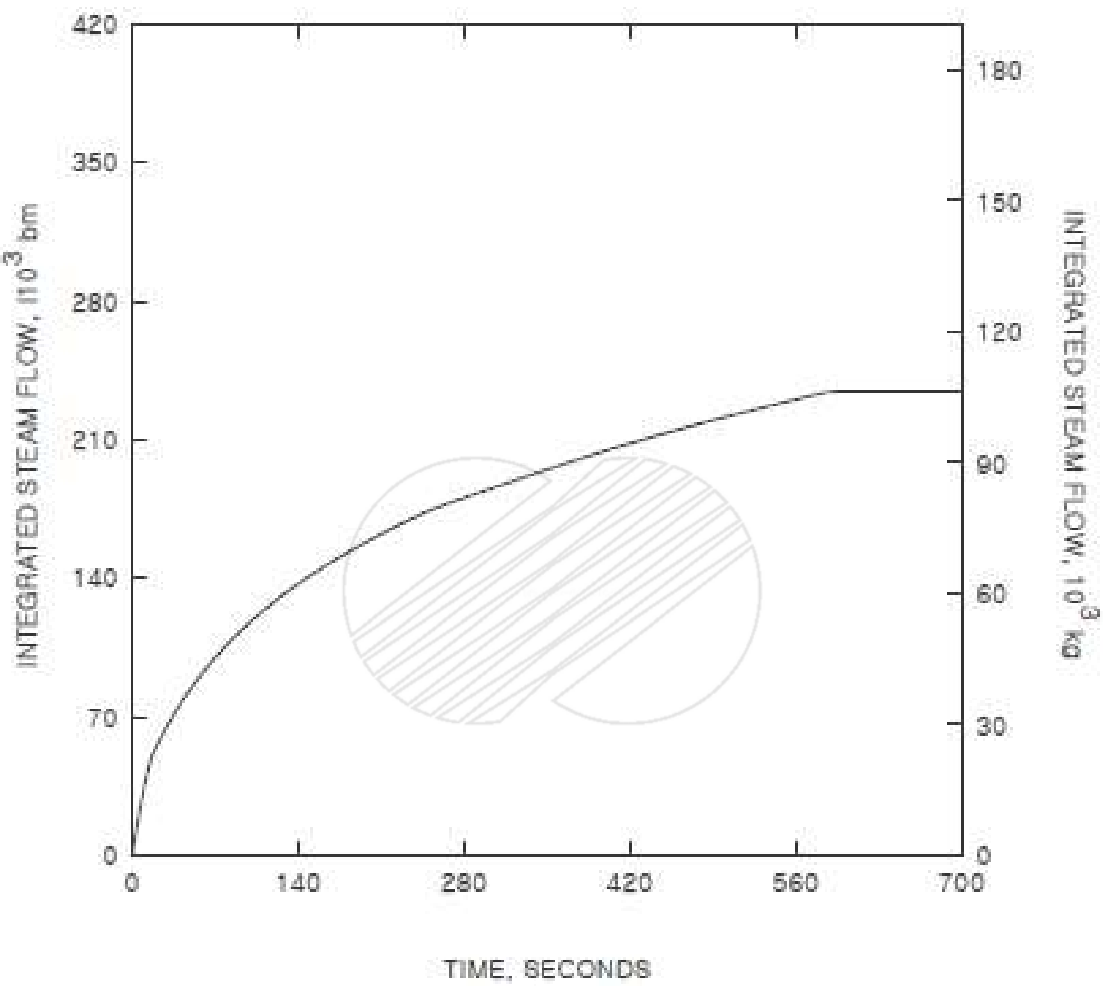
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS–OF–OFFSITE POWER:
STEAM–GENERATOR LIQUID MASS VS. TIME

Figure 15.1.5–13



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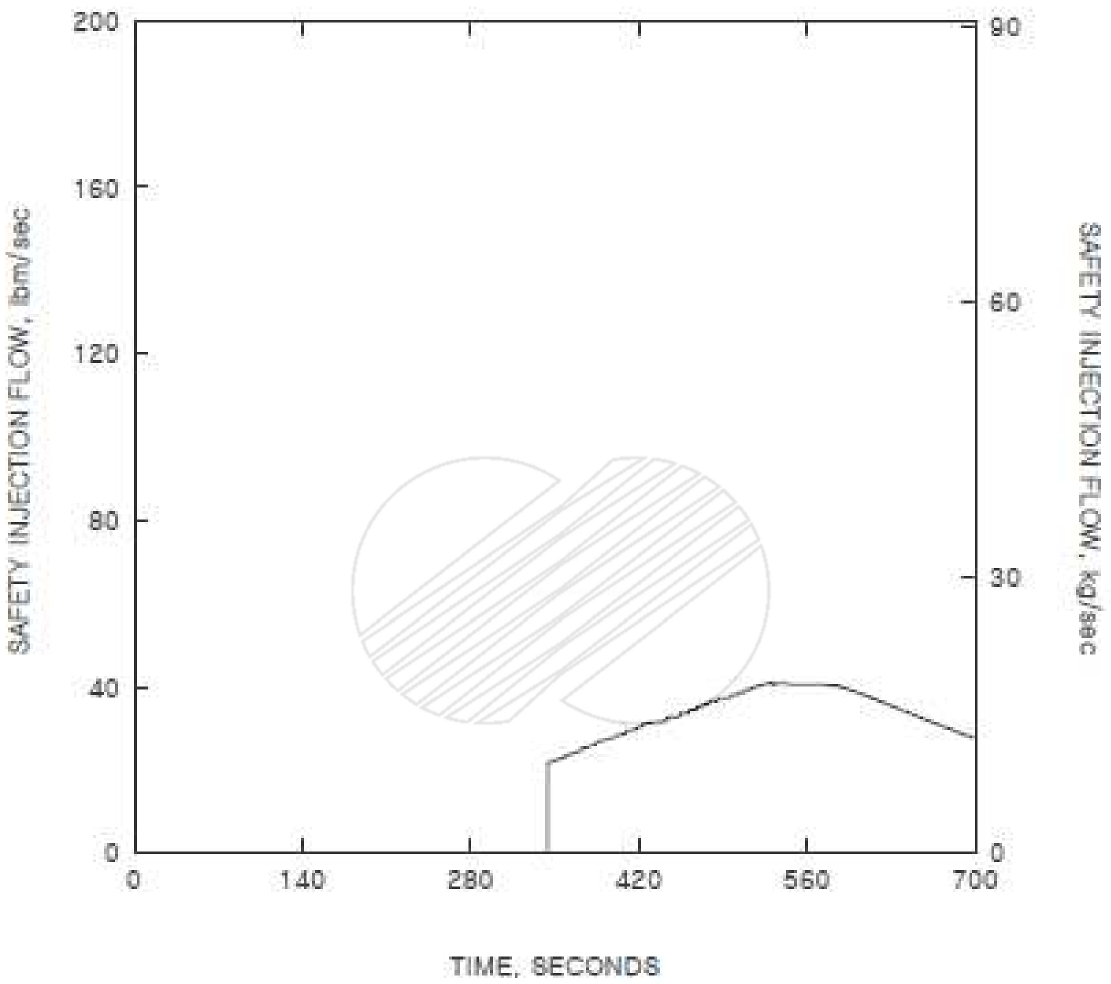
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
INTEGRATED STEAM MASS RELEASE
THRU BREAK VS. TIME

Figure 15.1.5-14



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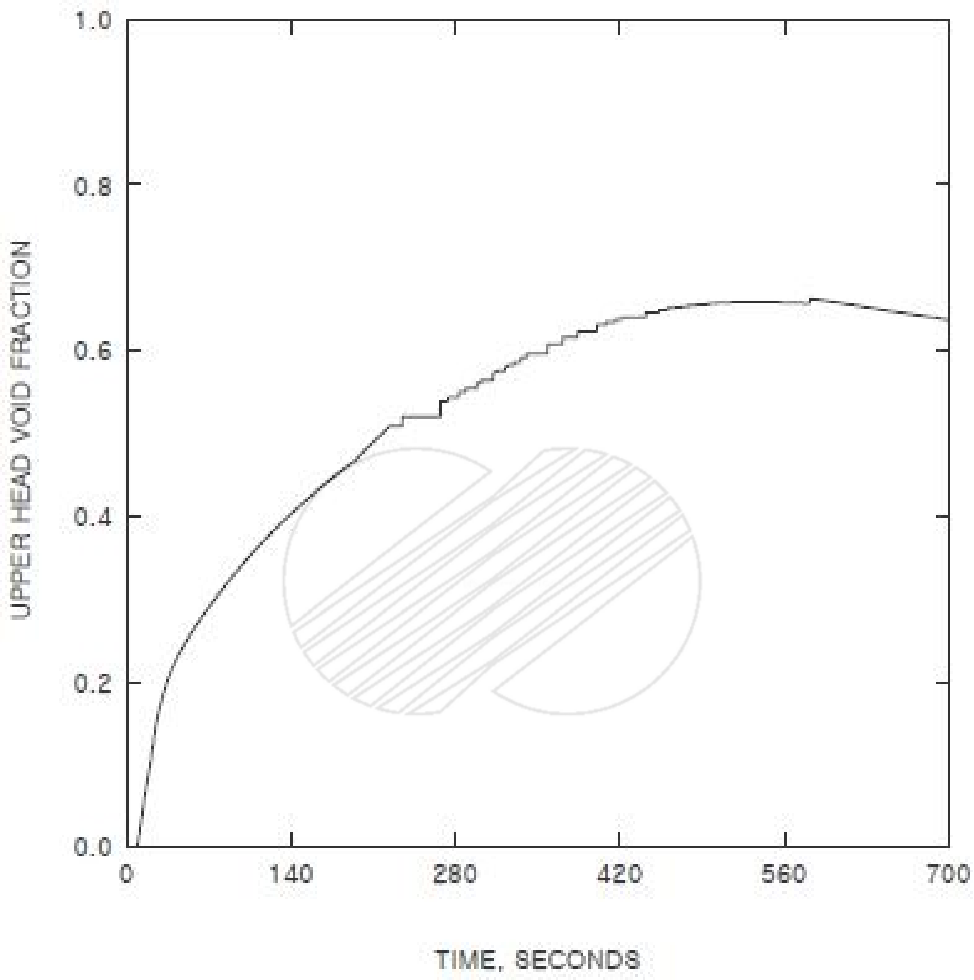
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
SAFETY INJECTION FLOW VS. TIME

Figure 15.1.5-15



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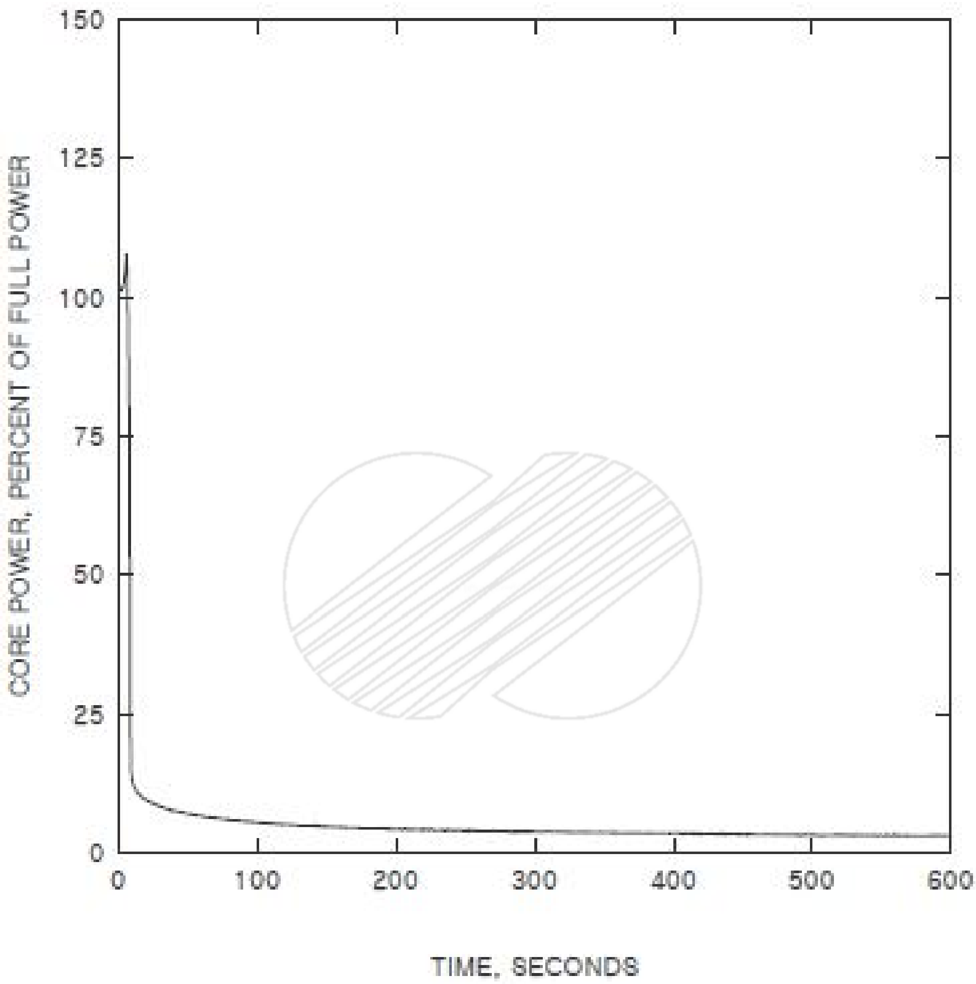
FULL POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
UPPER HEAD VOID FRACTION VS. TIME

Figure 15.1.5-16



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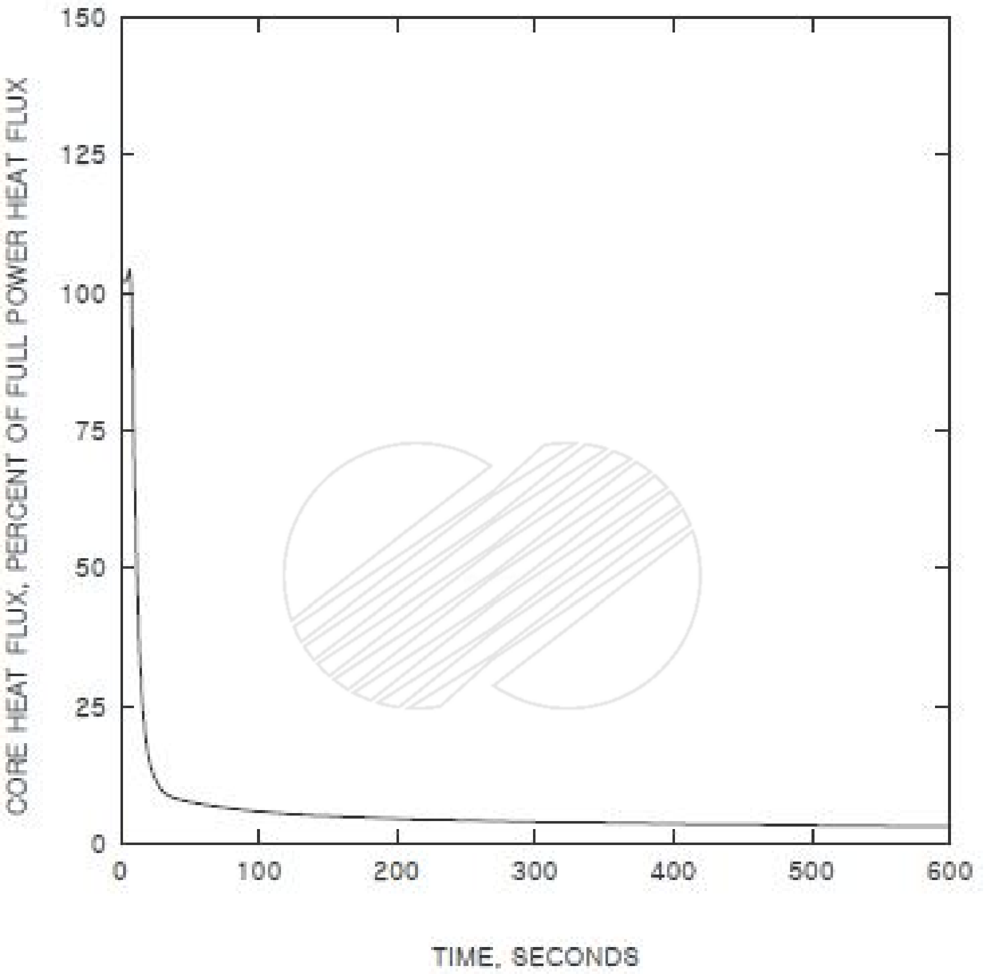
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
CORE POWER VS. TIME

Figure 15.1.5-17



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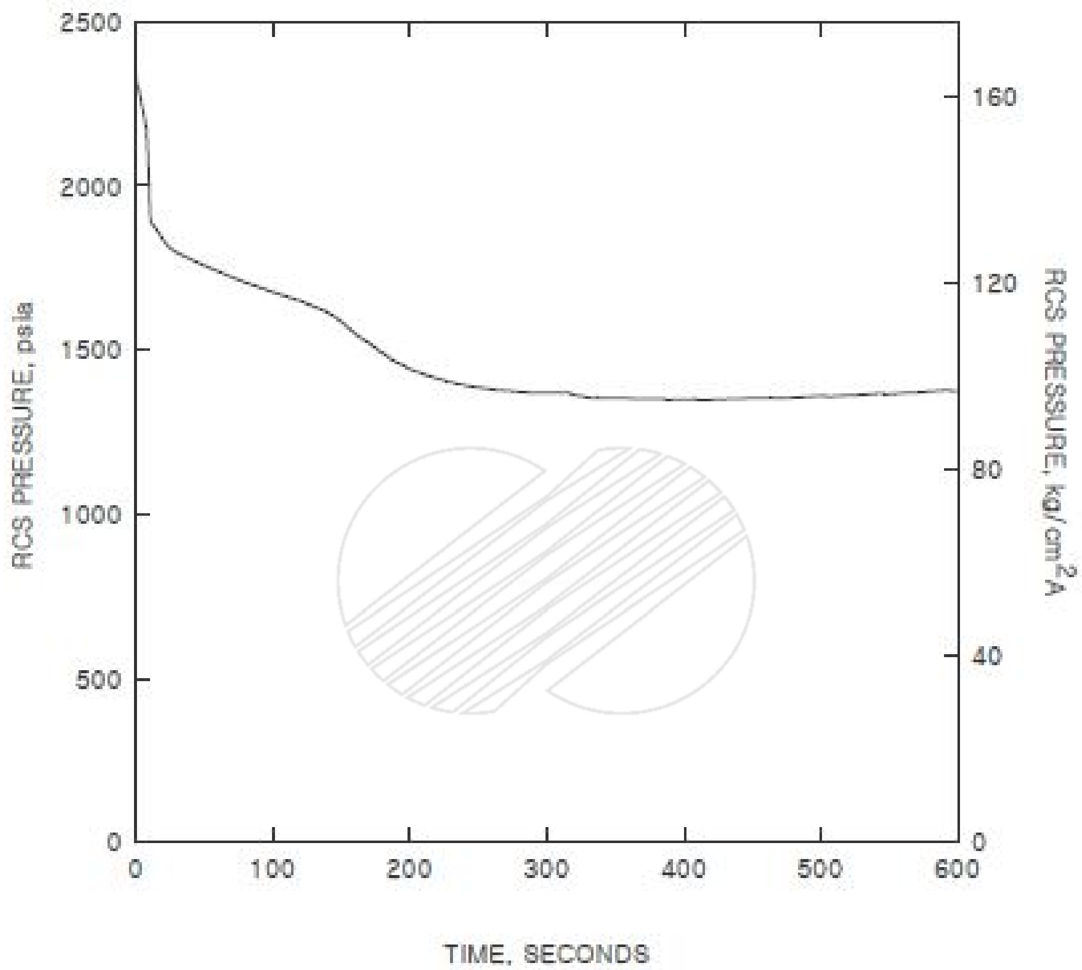
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
CORE HEAT FLUX VS. TIME

Figure 15.1.5-18



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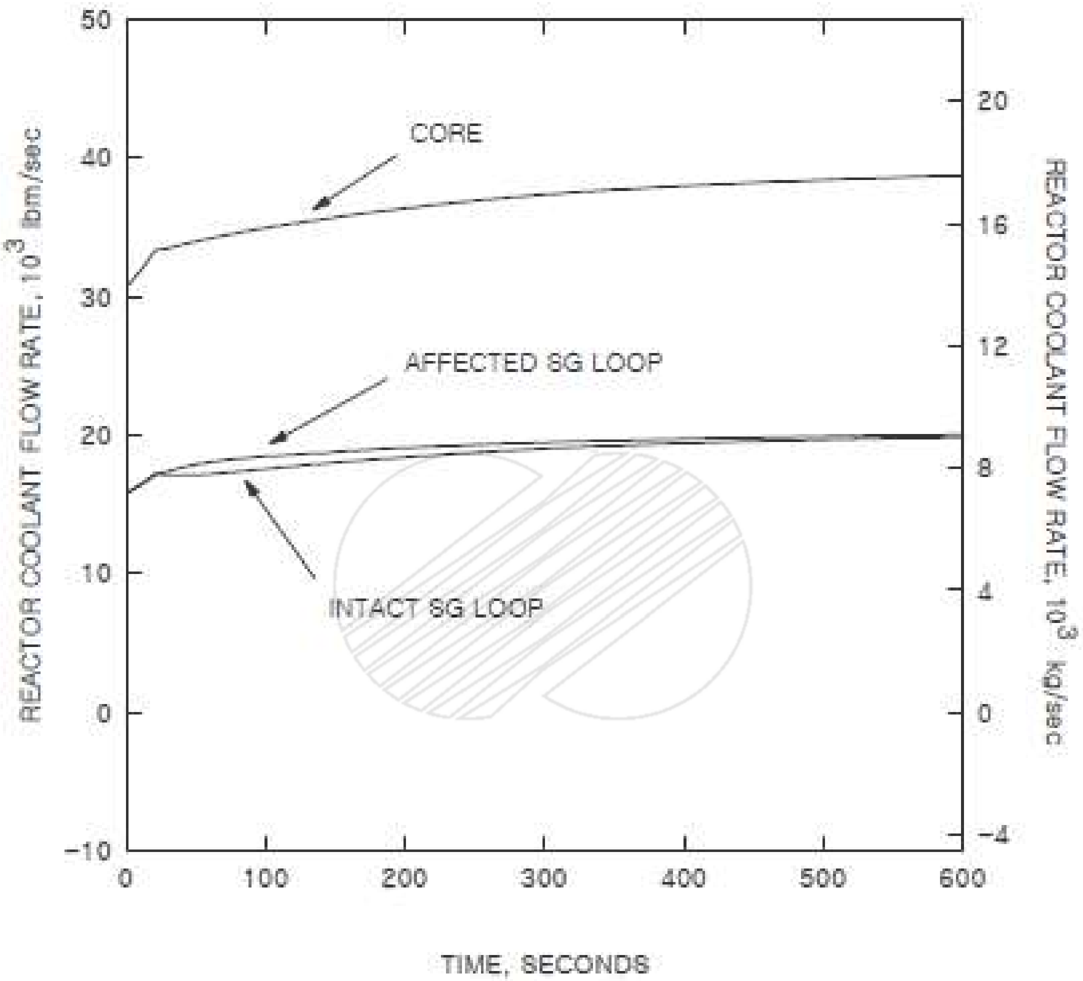
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
RCS PRESSURE VS. TIME

Figure 15.1.5-19



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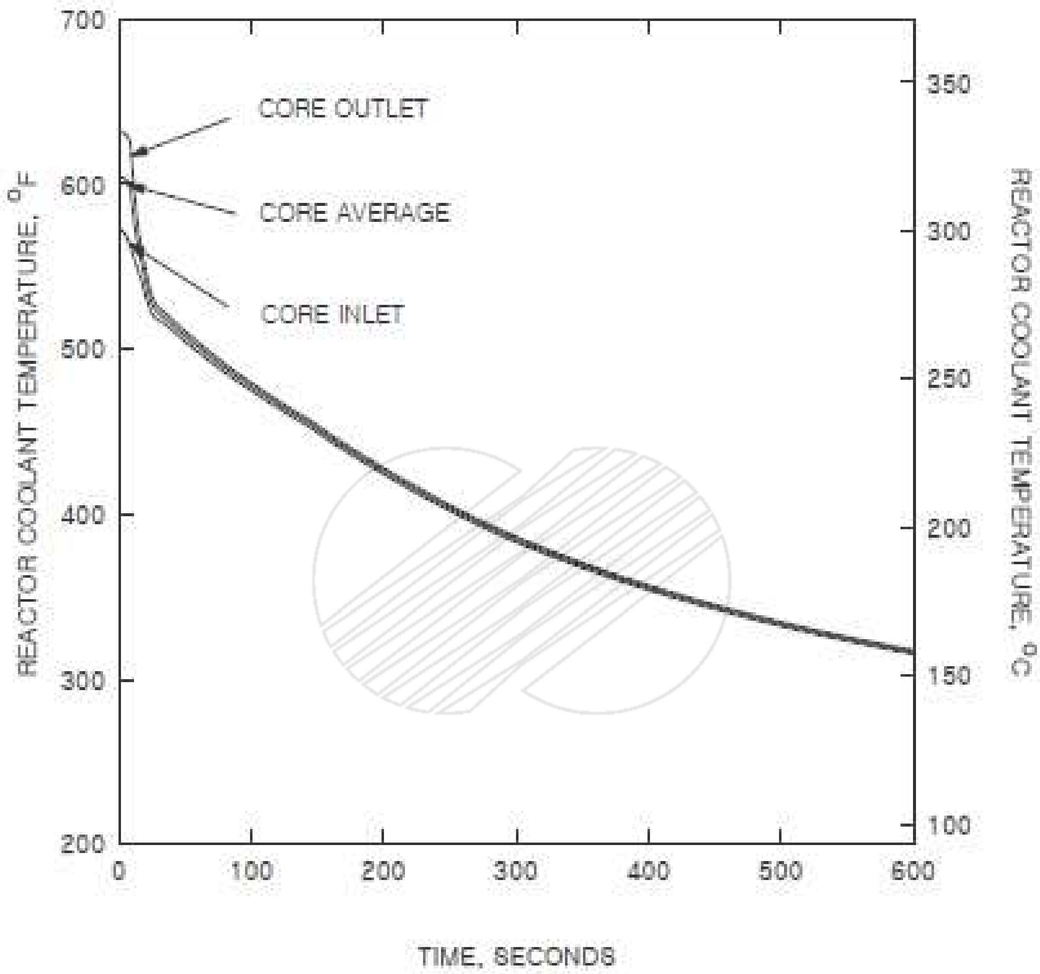
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
REACTOR COOLANT FLOW RATE VS. TIME

Figure 15.1.5-20



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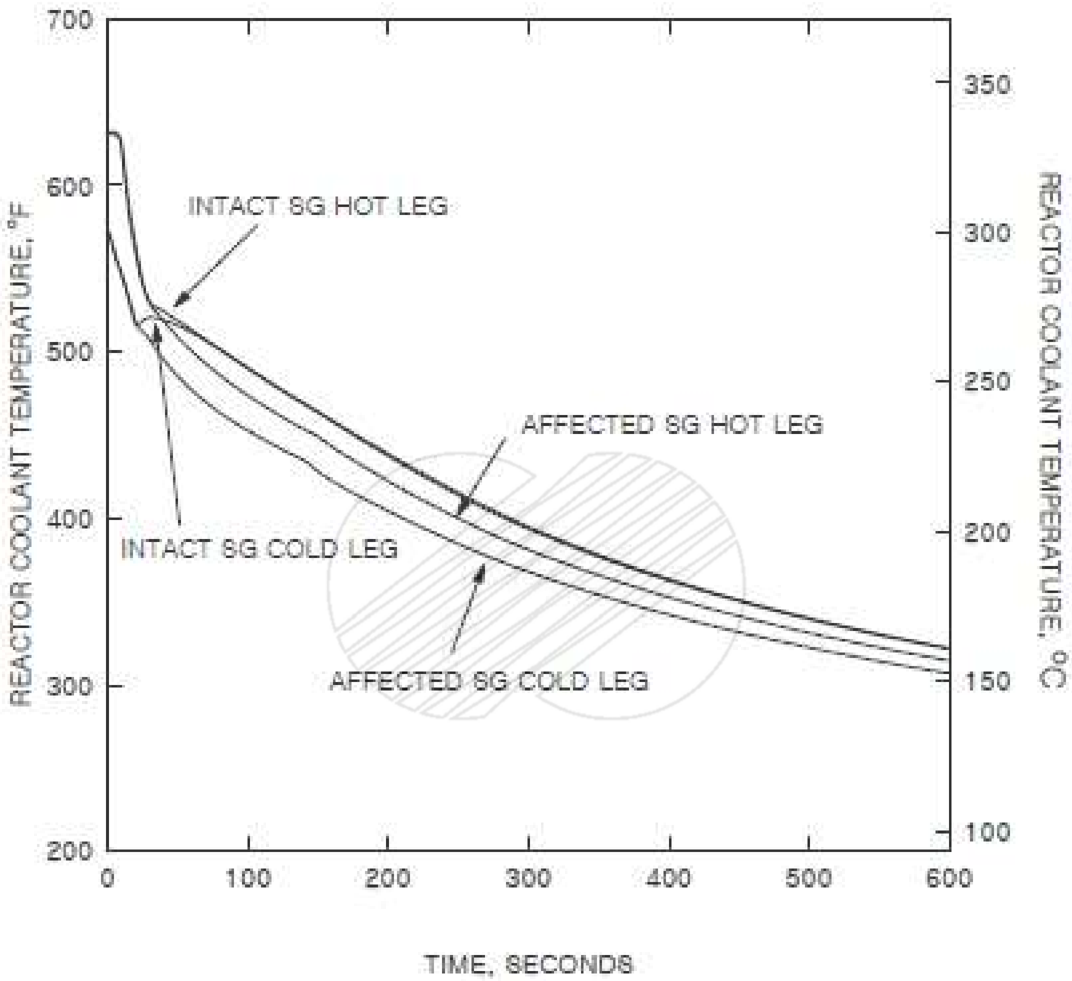
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
REACTOR COOLANT TEMPERATURES (A) VS. TIME

Figure 15.1.5-21



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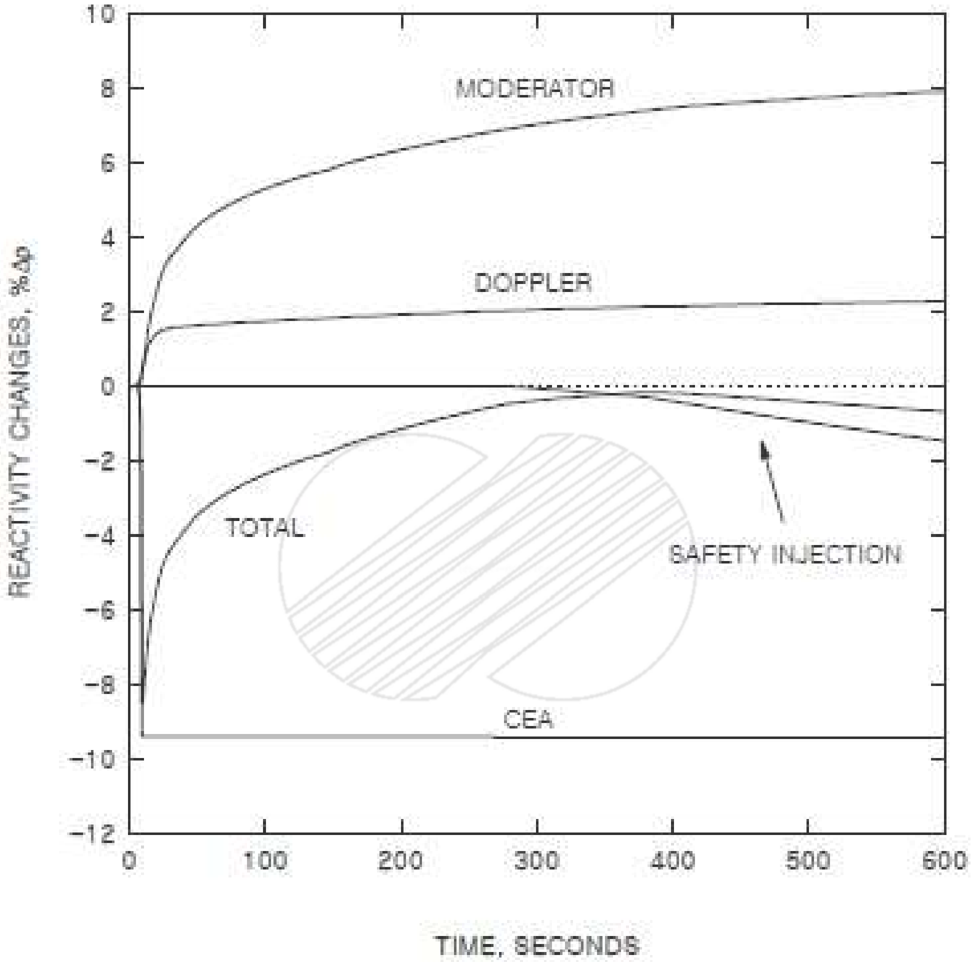
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
REACTOR COOLANT TEMPERATURES (B) VS. TIME

Figure 15.1.5-22



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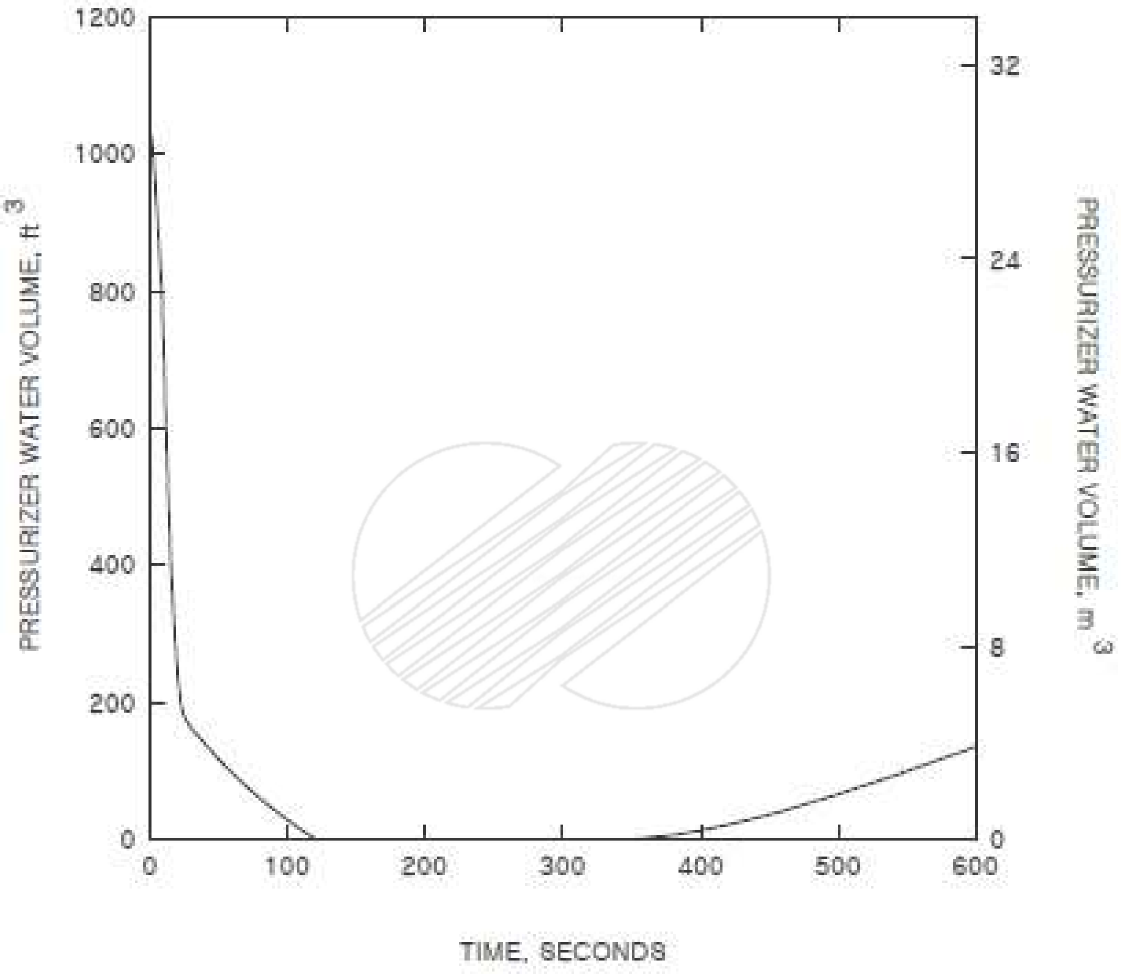
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
REACTIVITY CHANGES VS. TIME

Figure 15.1.5-23



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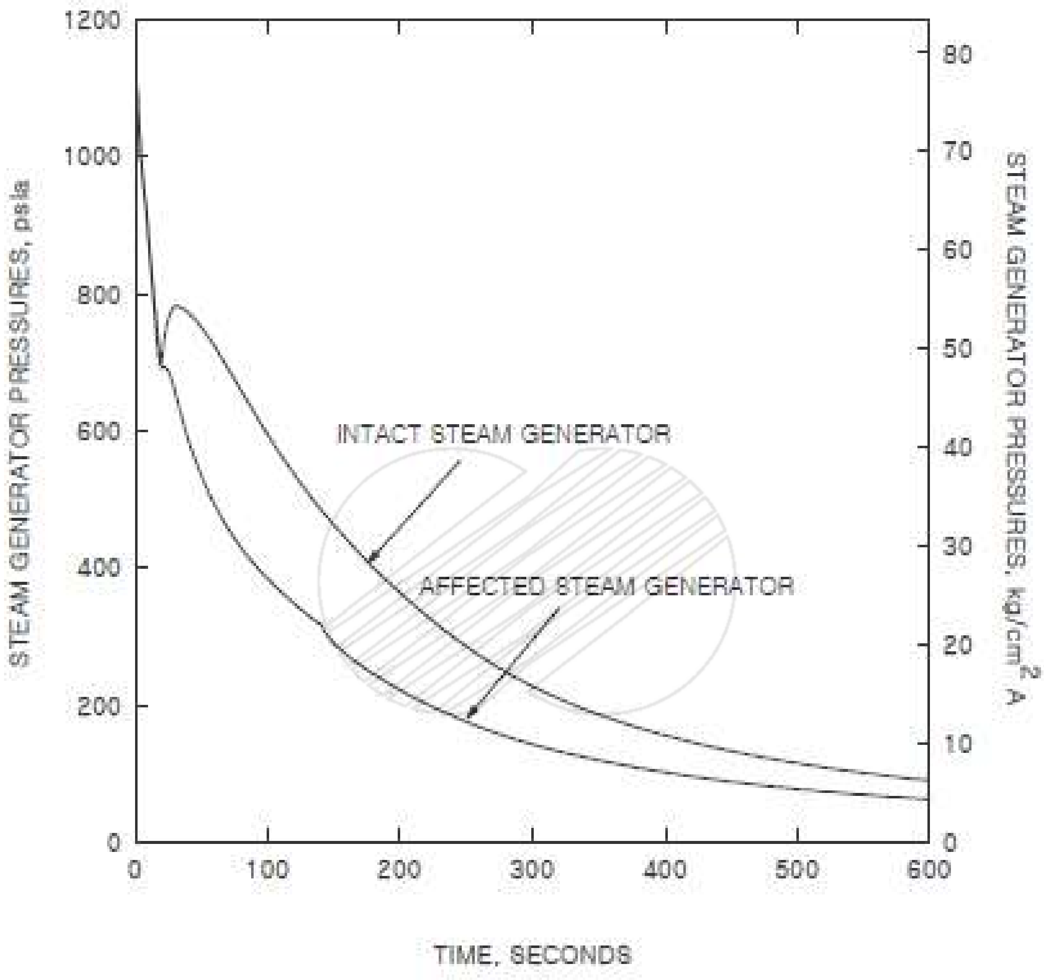
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
PRESSURIZER WATER VOLUME VS. TIME

Figure 15.1.5-24



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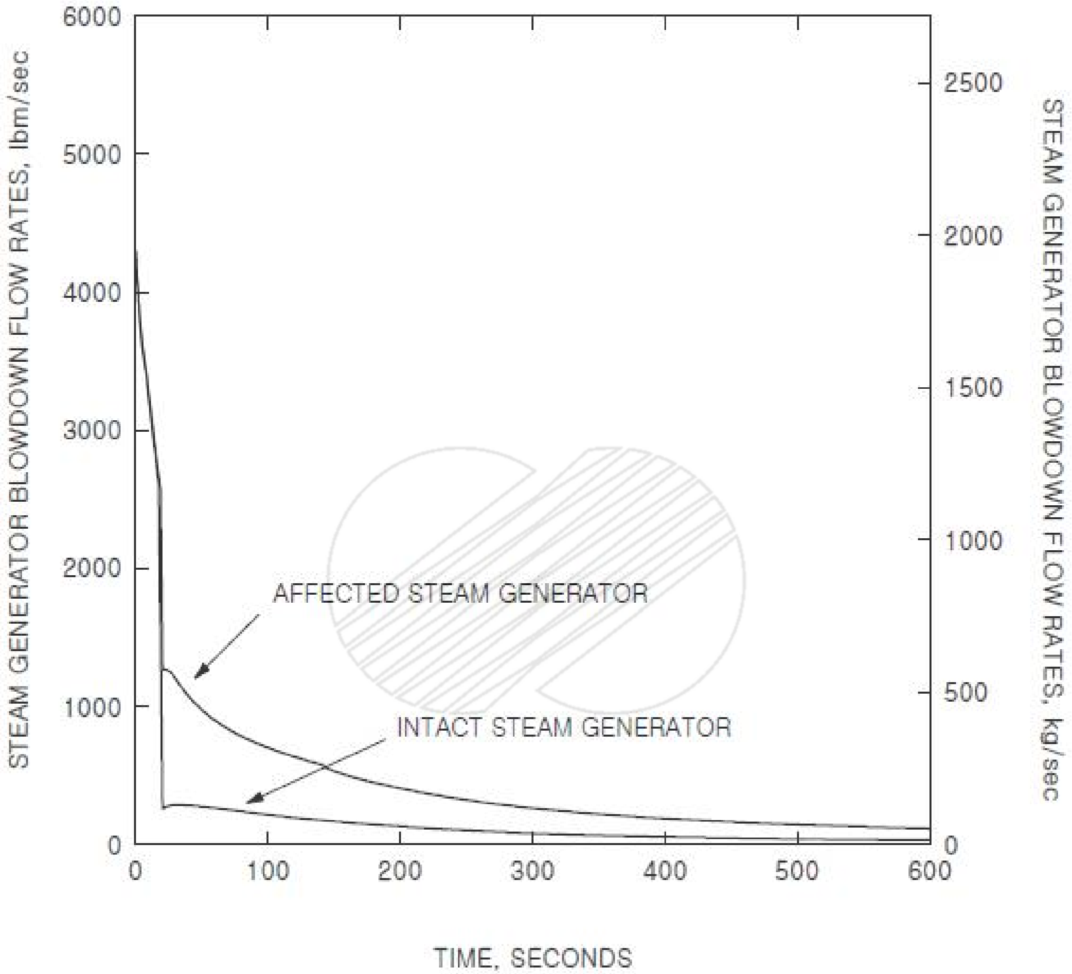
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
STEAM-GENERATOR PRESSURES VS. TIME

Figure 15.1.5-25



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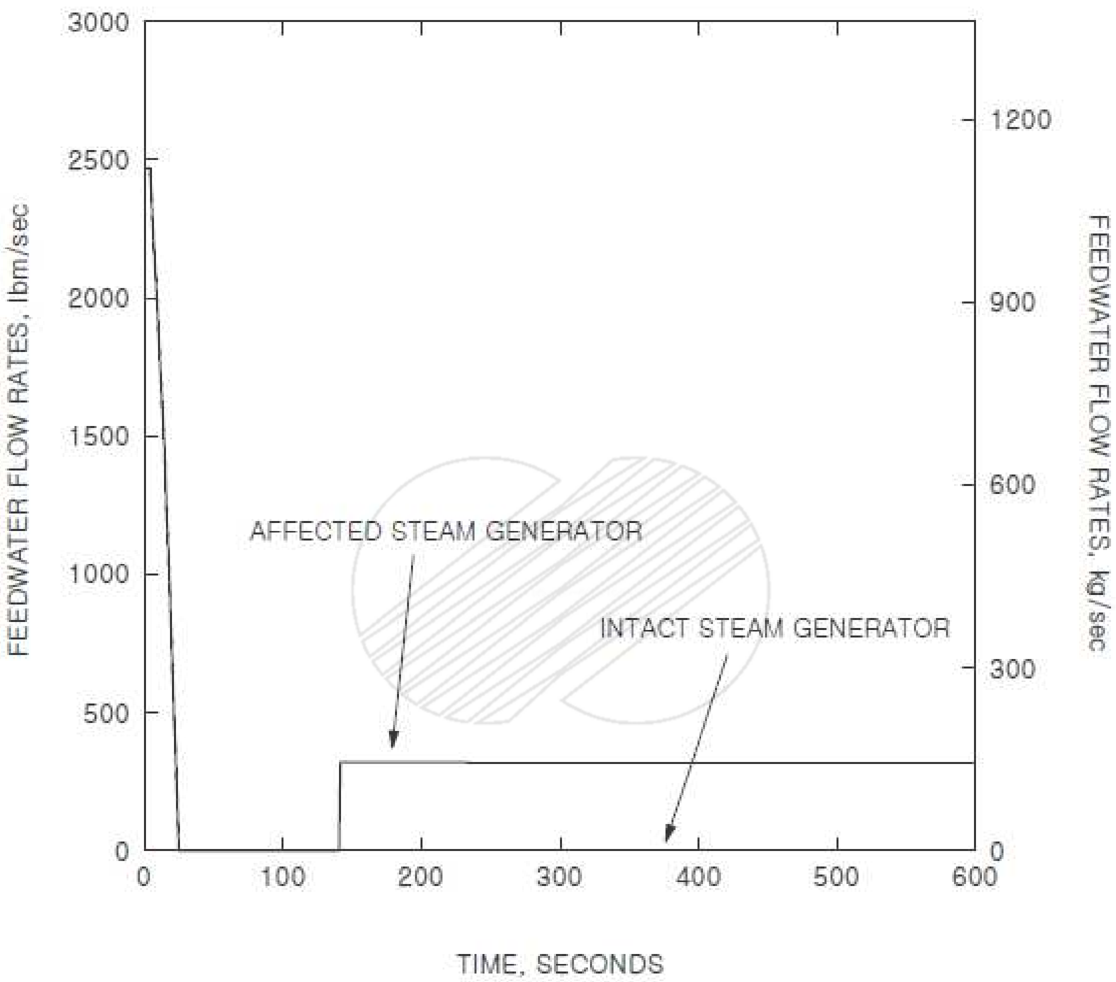
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
STEAM-GENERATOR BLOWDOWN RATES VS. TIME

Figure 15.1.5-26



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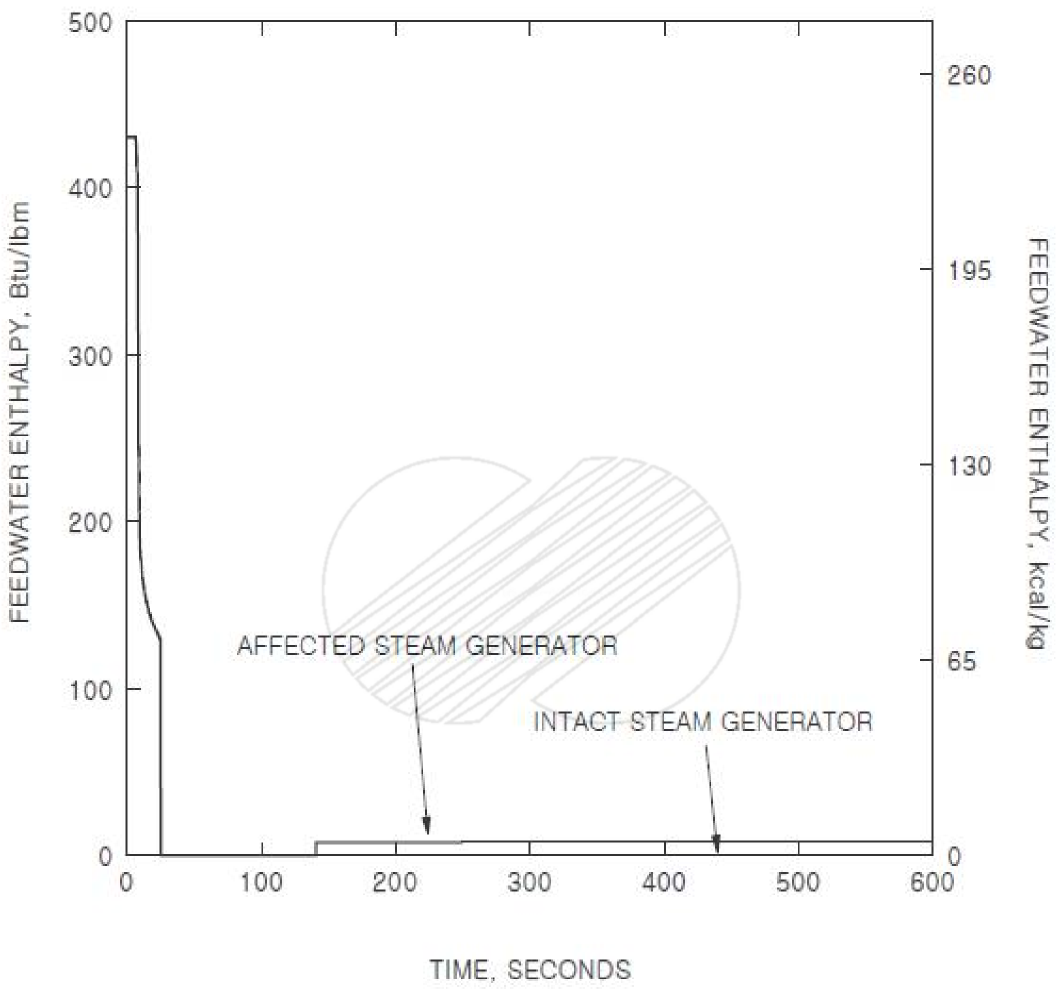
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
FEEDWATER FLOW RATES VS. TIME

Figure 15.1.5-27



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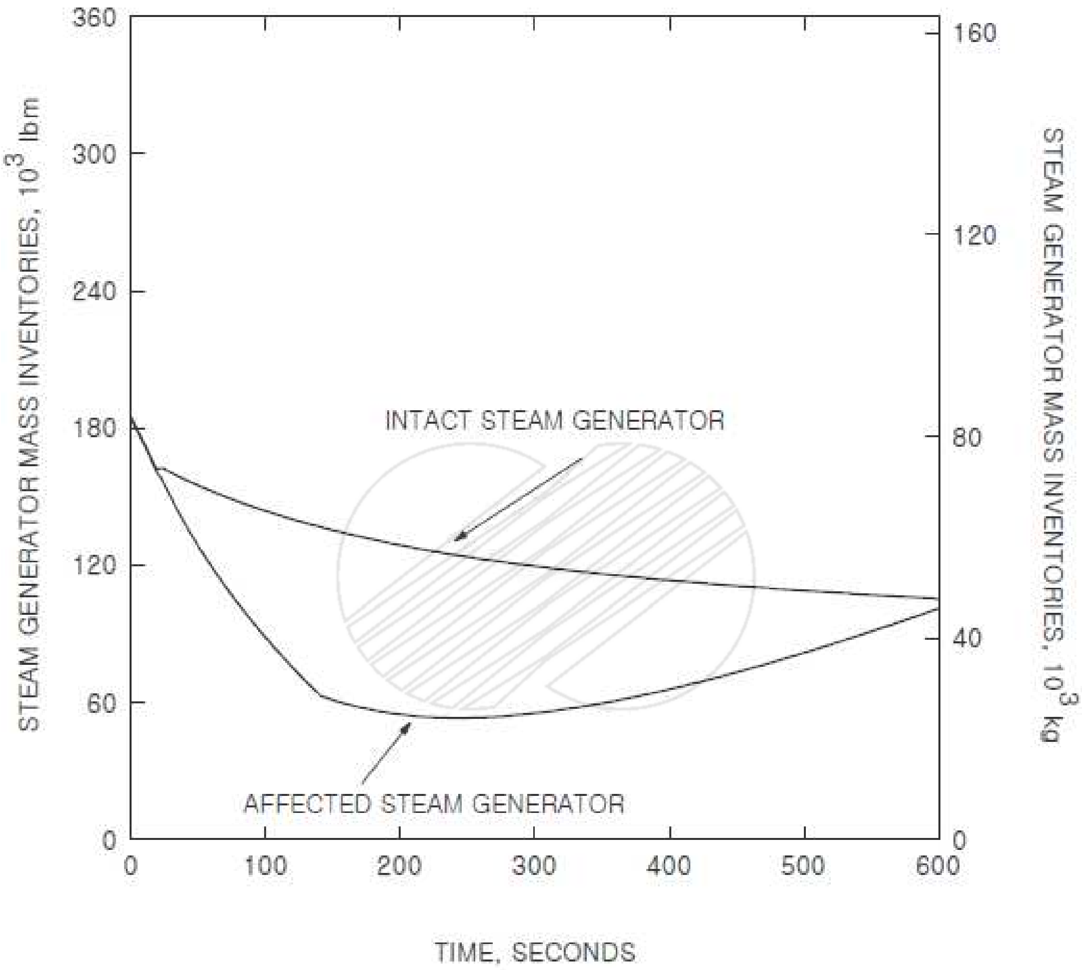
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
FEEDWATER ENTHALPY VS. TIME

Figure 15.1.5-28



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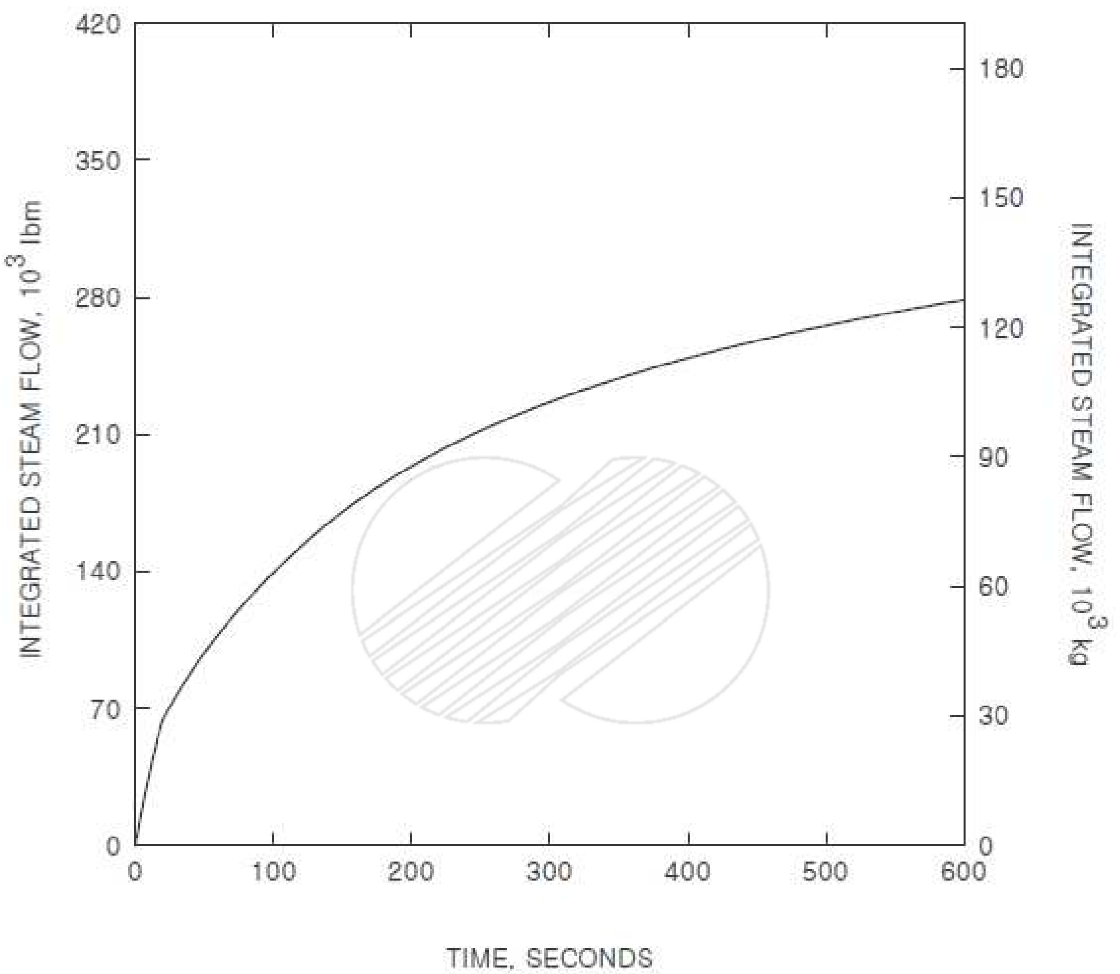
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
STEAM-GENERATOR LIQUID MASS VS. TIME

Figure 15.1.5-29



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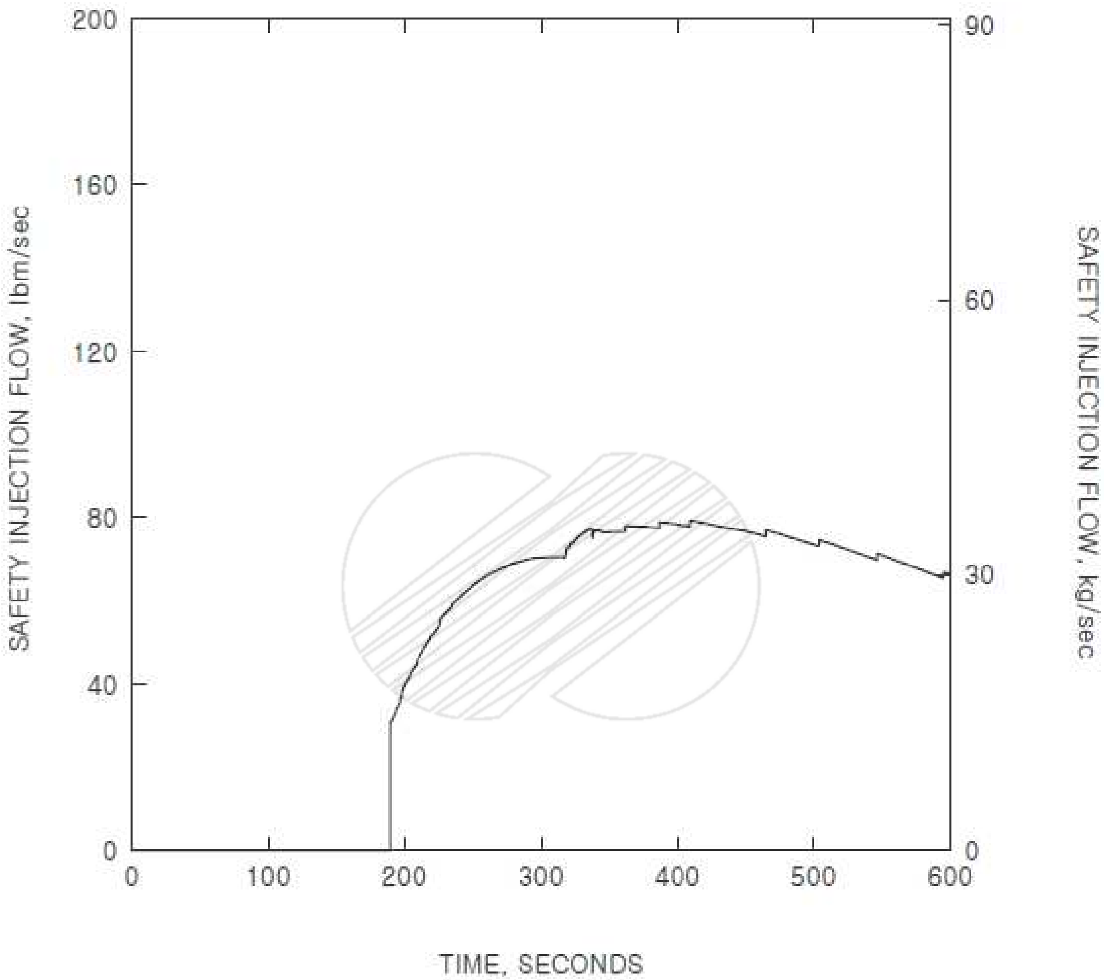
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
INTEGRATED STEAM RELEASE VS. TIME

Figure 15.1.5-30



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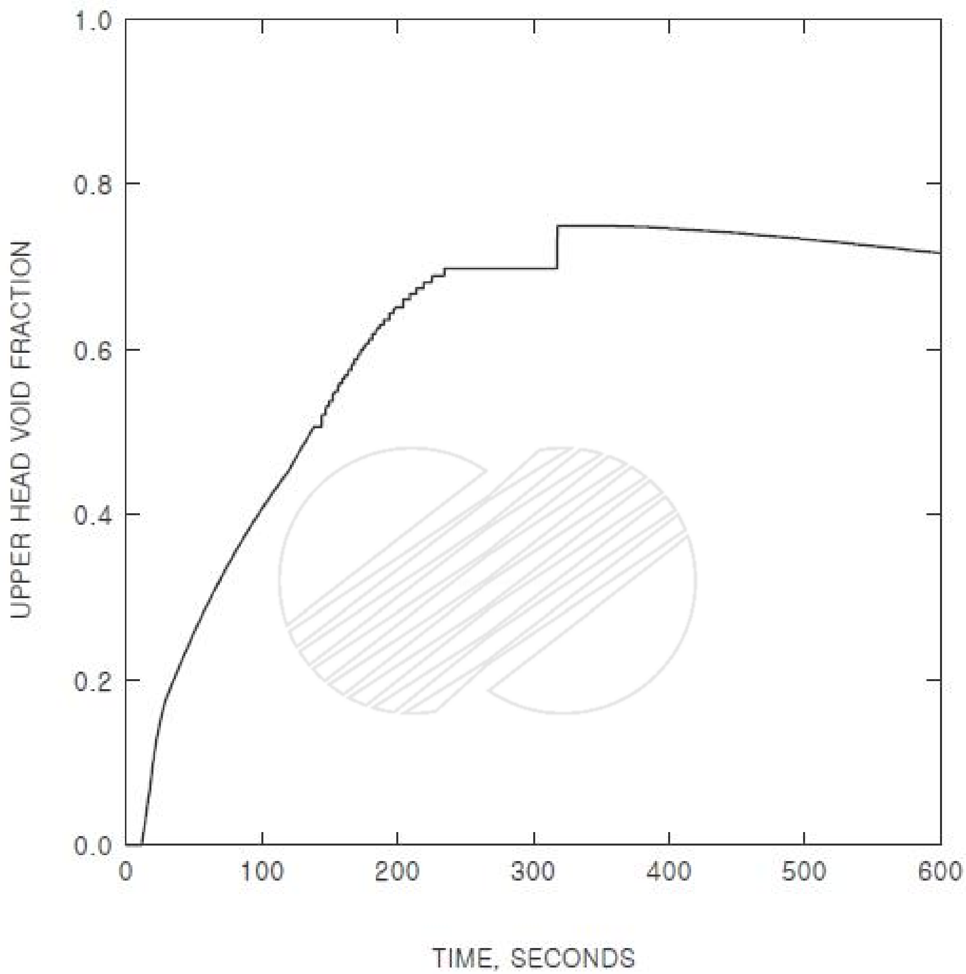
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
SAFETY INJECTION FLOW VS. TIME

Figure 15.1.5-31



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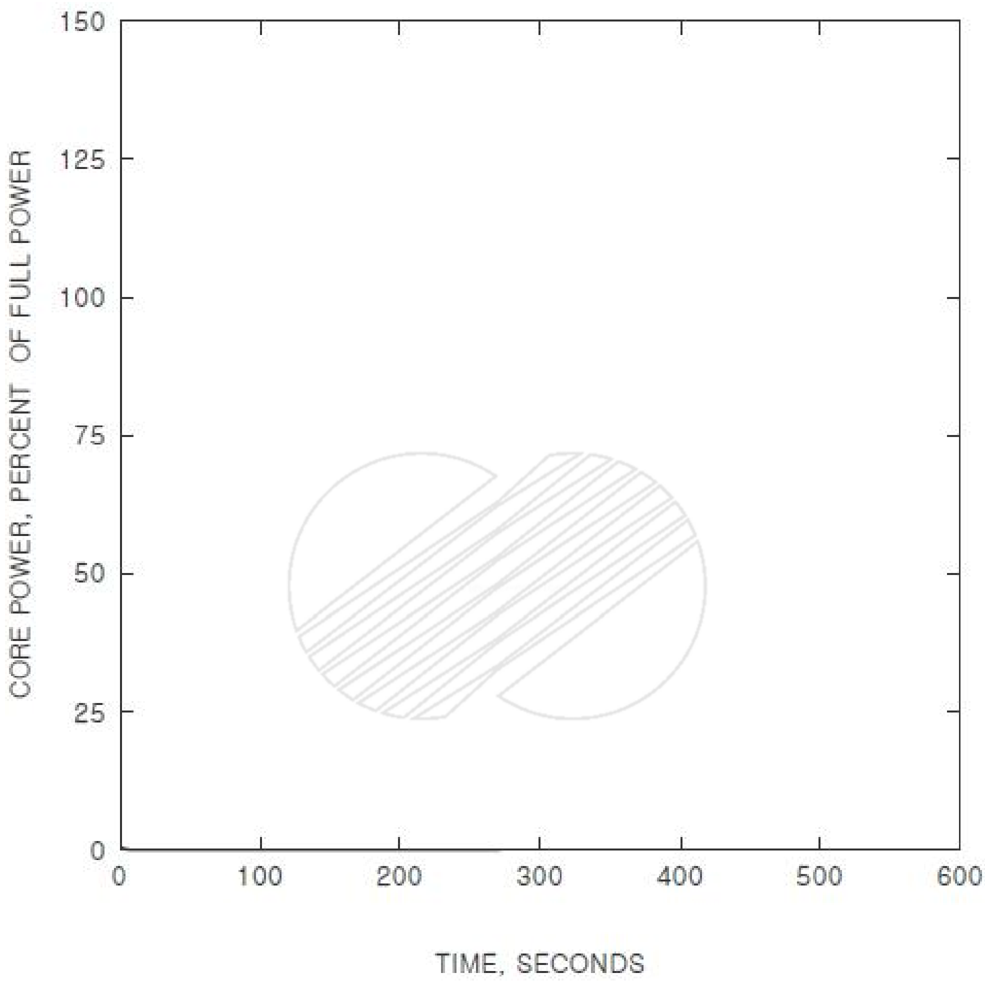
FULL POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
UPPER HEAD VOID FRACTION VS. TIME

Figure 15.1.5-32



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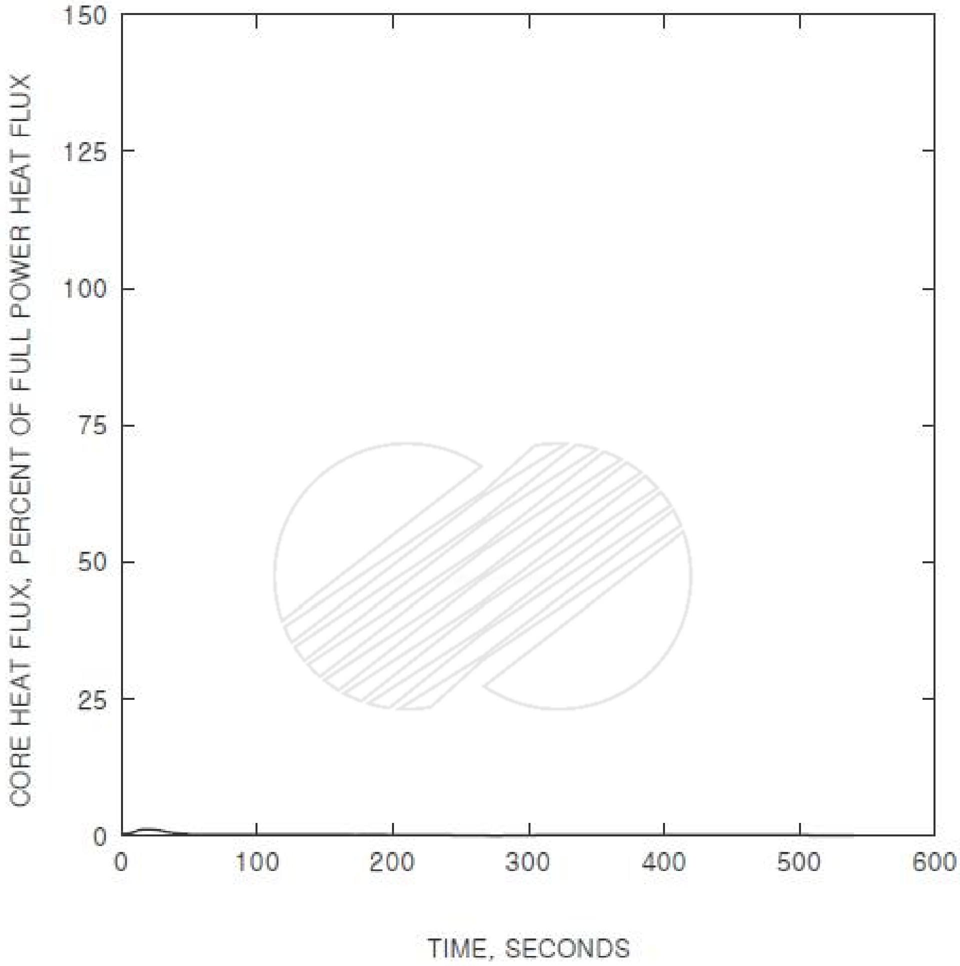
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
CORE POWER VS. TIME

Figure 15.1.5-33



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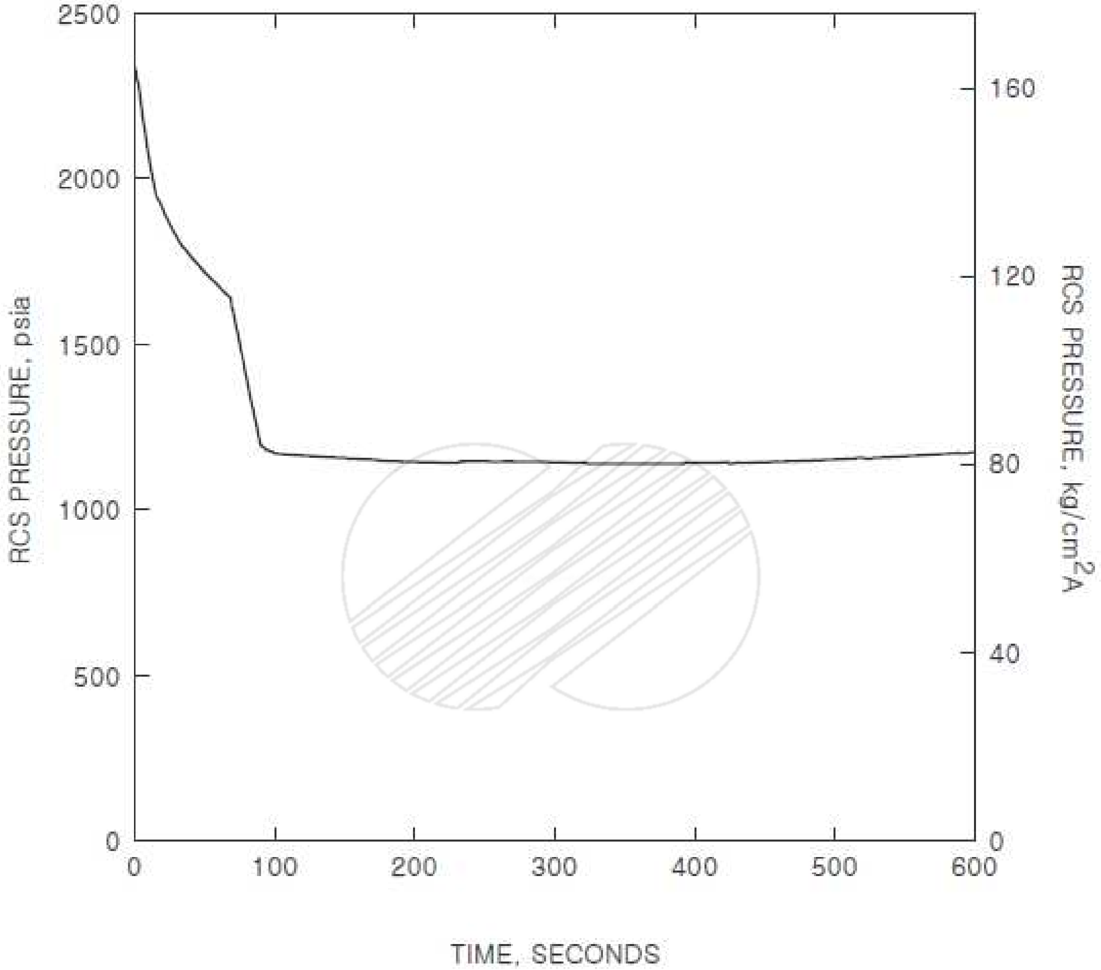
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
CORE HEAT FLUX VS. TIME

Figure 15.1.5-34



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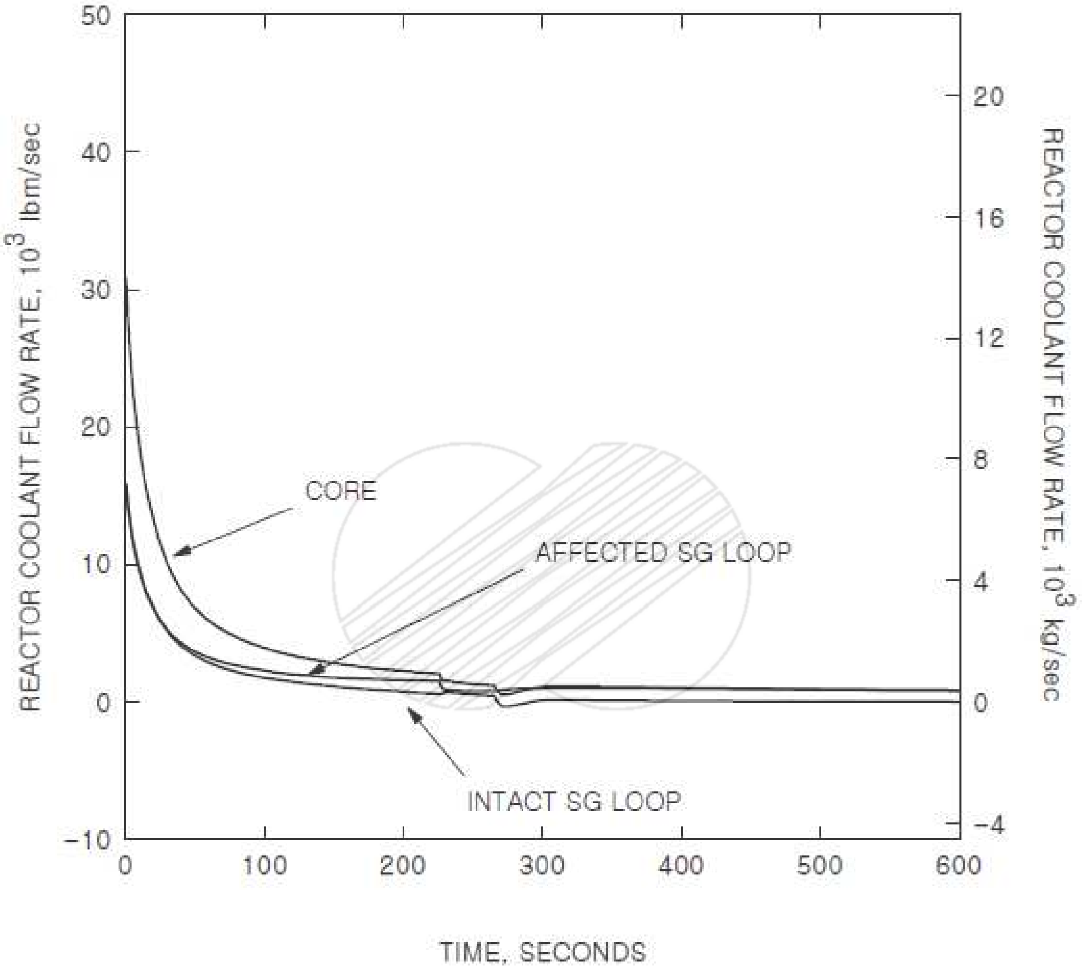
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
RCS PRESSURE VS. TIME

Figure 15.1.5-35



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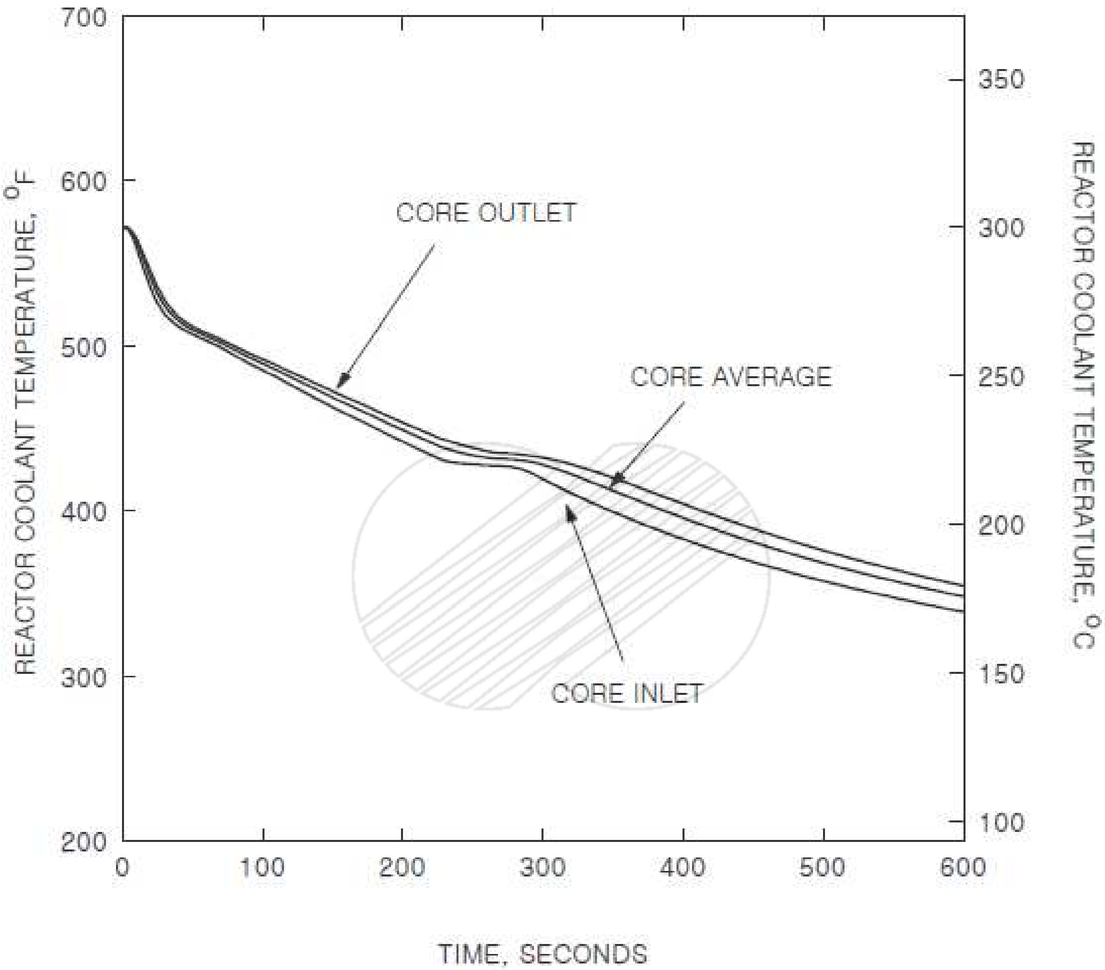
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
REACTOR COOLANT FLOW RATE VS. TIME

Figure 15.1.5-36



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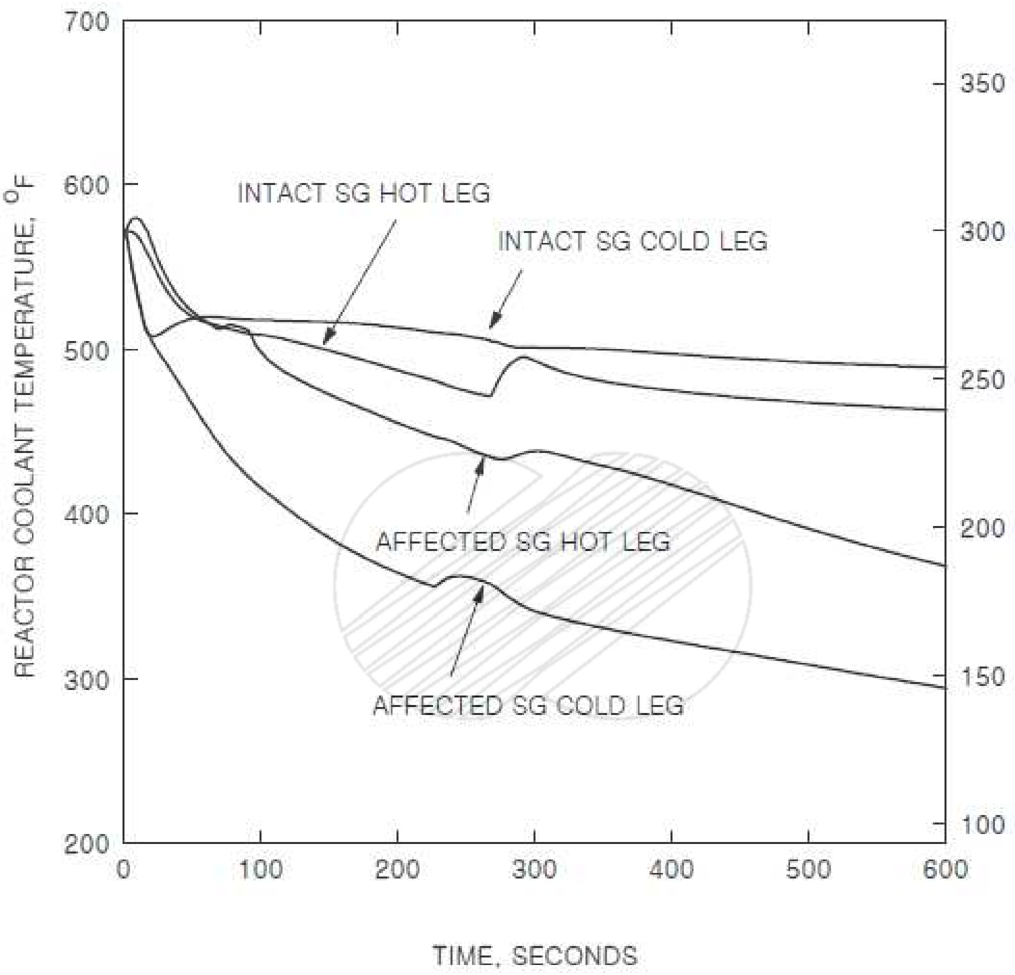
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS–OF–OFFSITE POWER:
REACTOR COOLANT TEMPERATURES (A) VS. TIME

Figure 15.1.5–37



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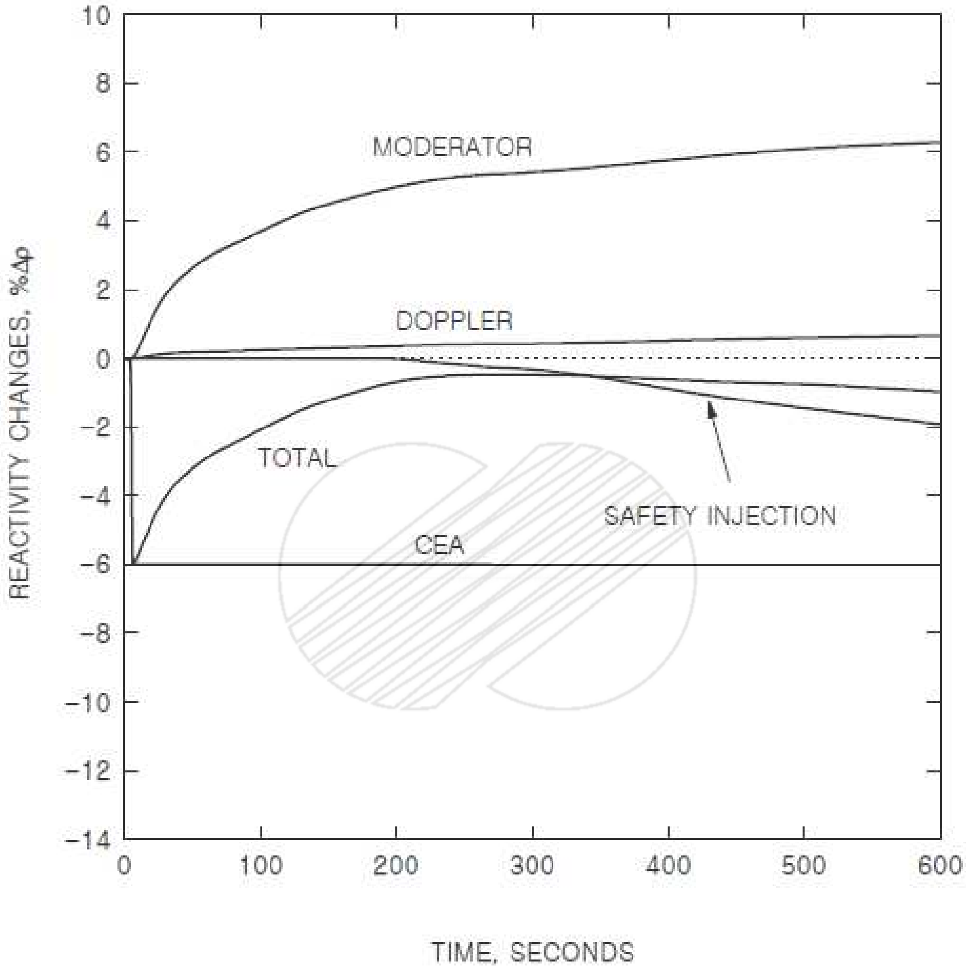
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS–OF–OFFSITE POWER:
REACTOR COOLANT TEMPERATURES (B) VS. TIME

Figure 15.1.5–38



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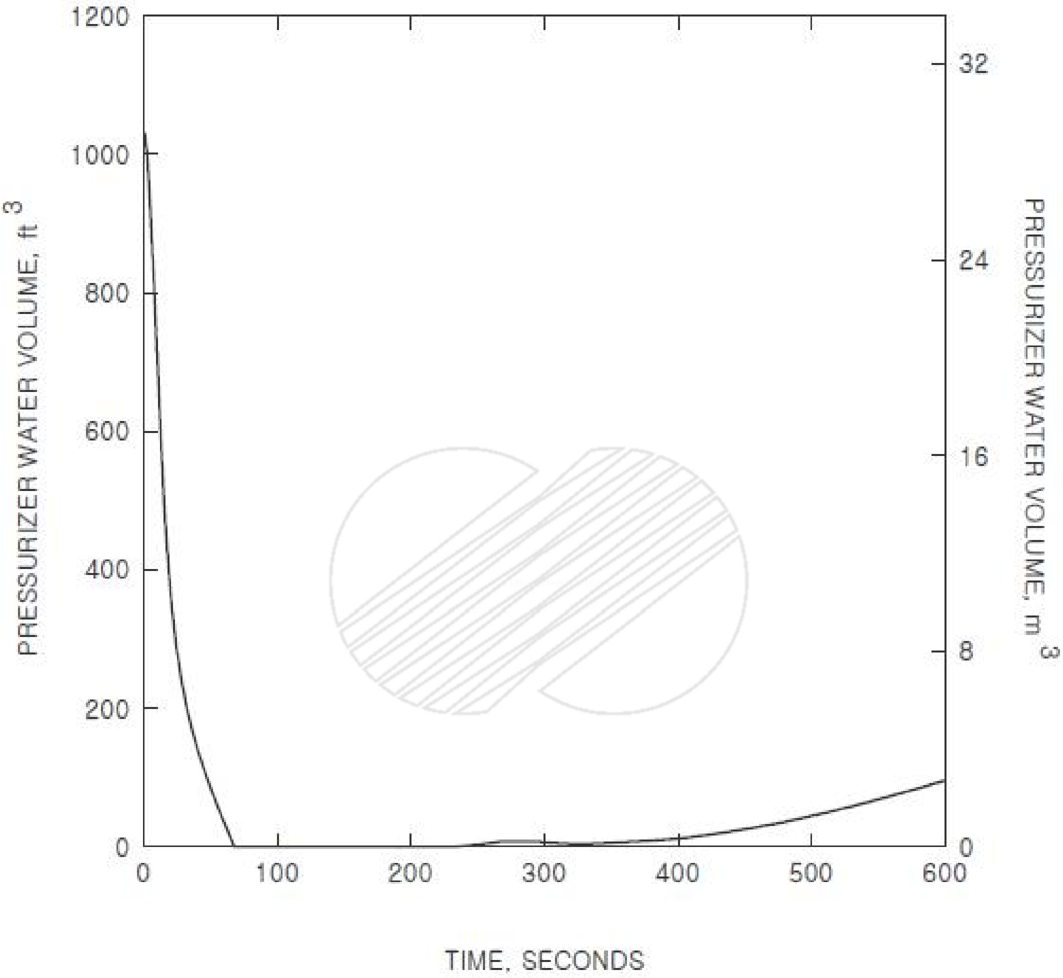
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS−OF−OFFSITE POWER:
REACTIVITY CHANGES VS. TIME

Figure 15.1.5−39



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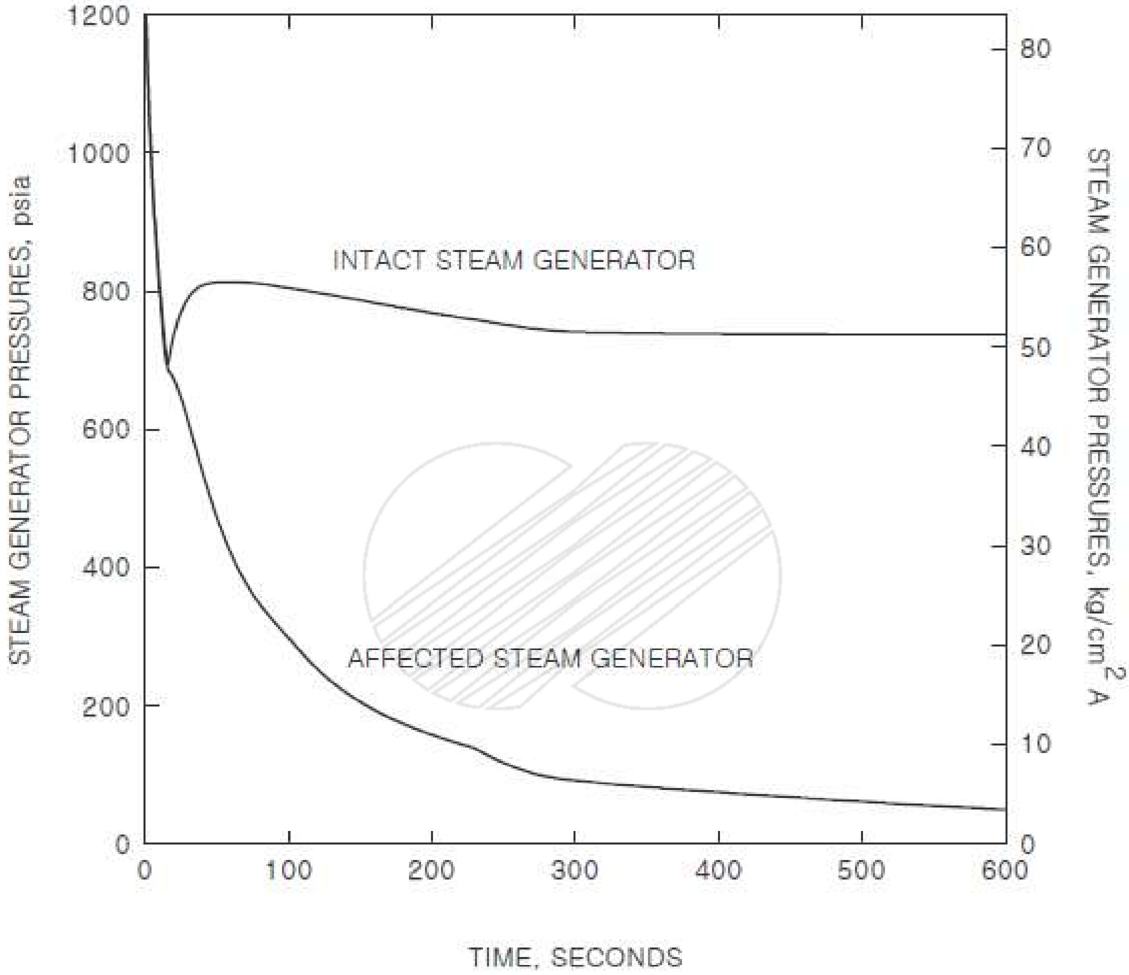
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
PRESSURIZER WATER VOLUME VS. TIME

Figure 15.1.5-40



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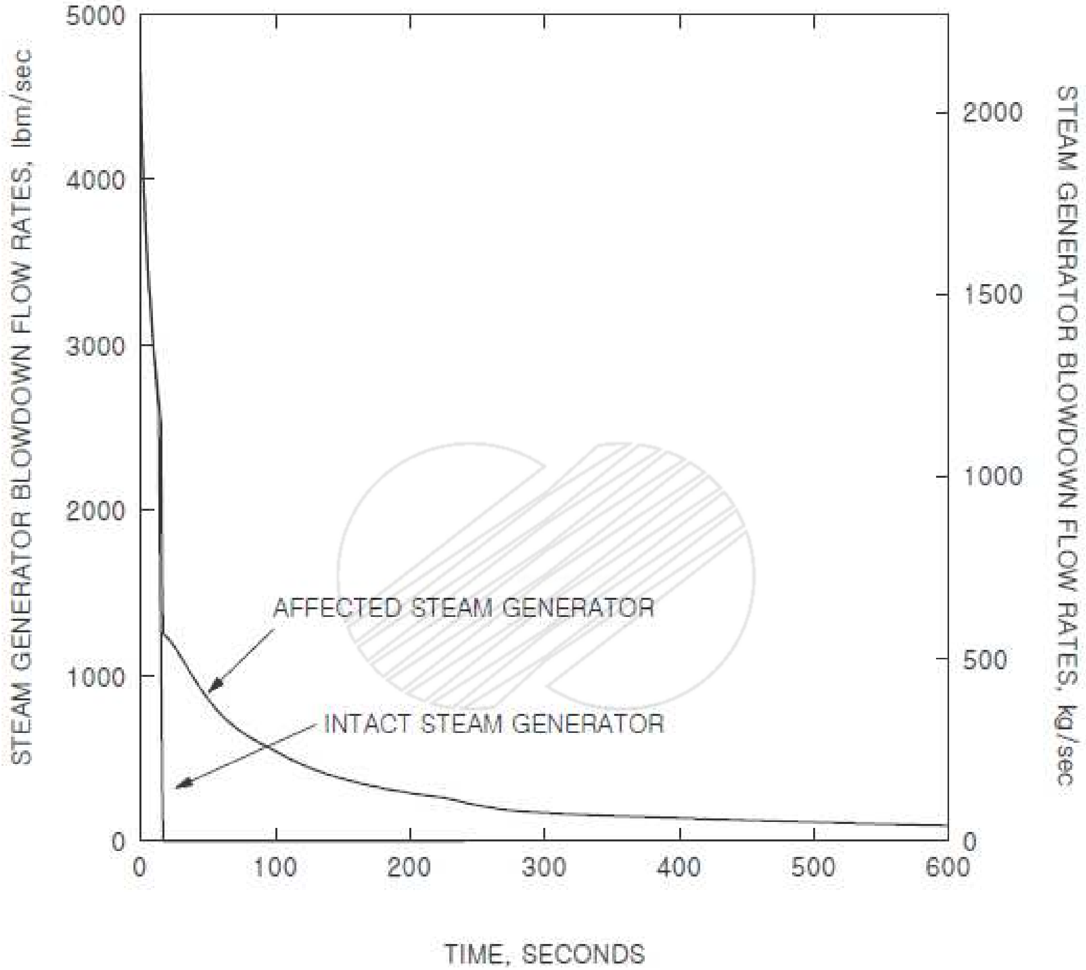
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS–OF–OFFSITE POWER:
STEAM–GENERATOR PRESSURES VS. TIME

Figure 15.1.5–41



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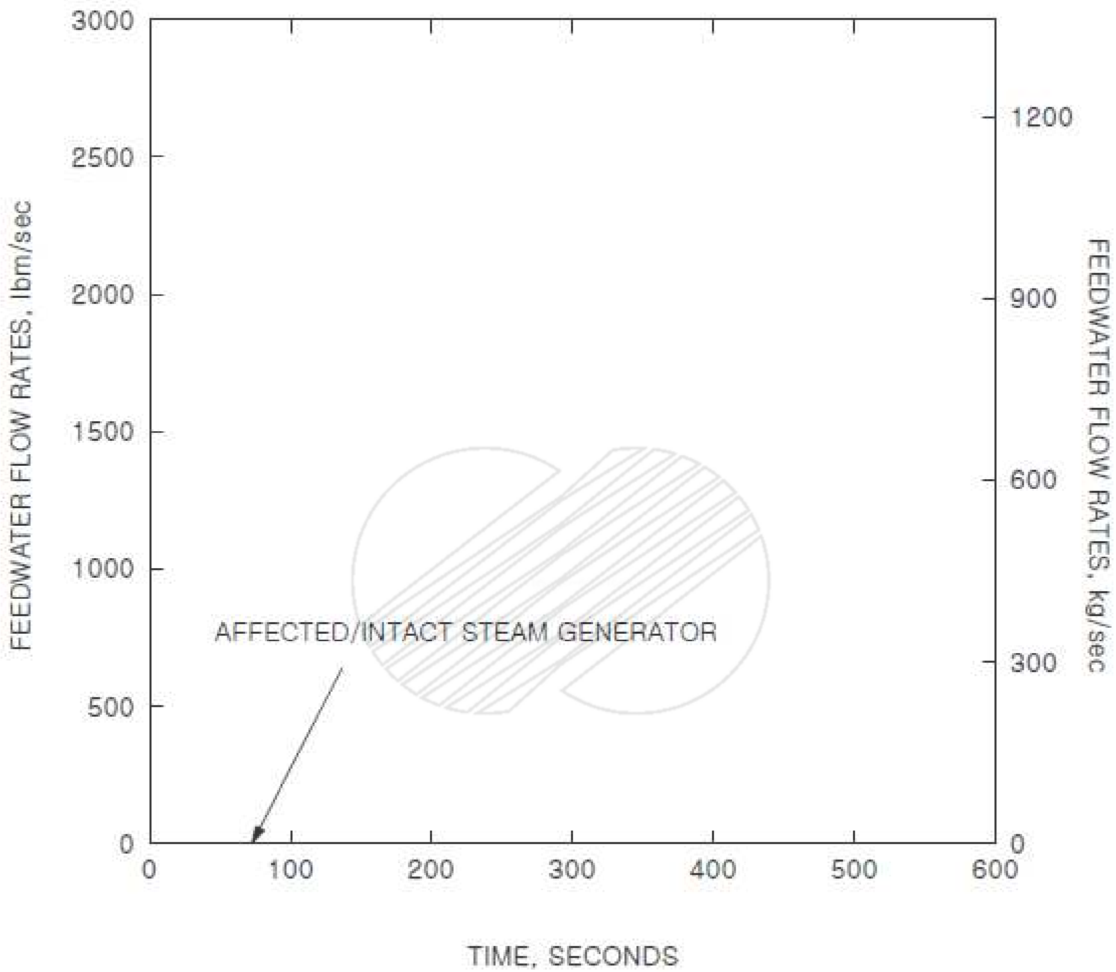
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
STEAM-GENERATOR BLOWDOWN RATES VS. TIME

Figure 15.1.5-42



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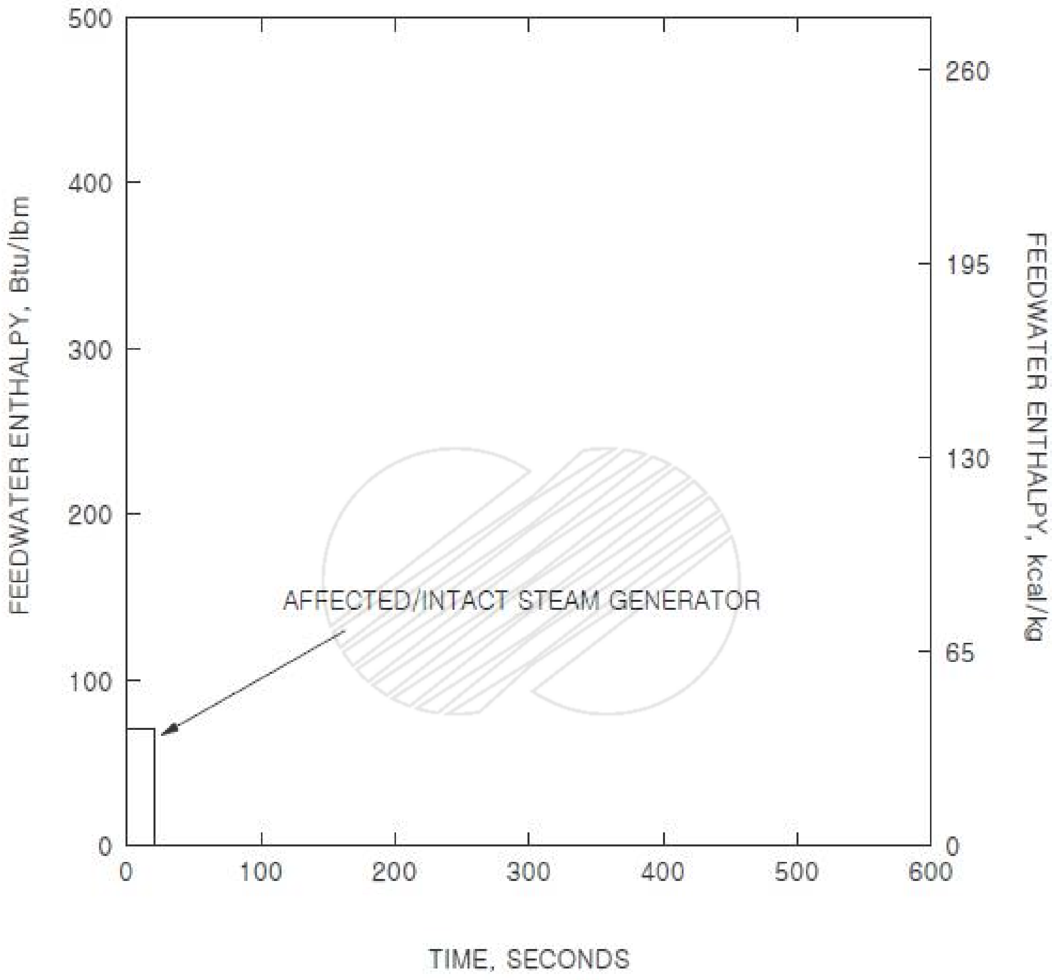
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
FEEDWATER FLOW RATES VS. TIME

Figure 15.1.5-43



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ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
FEEDWATER ENTHALPY VS. TIME

Figure 15.1.5-44

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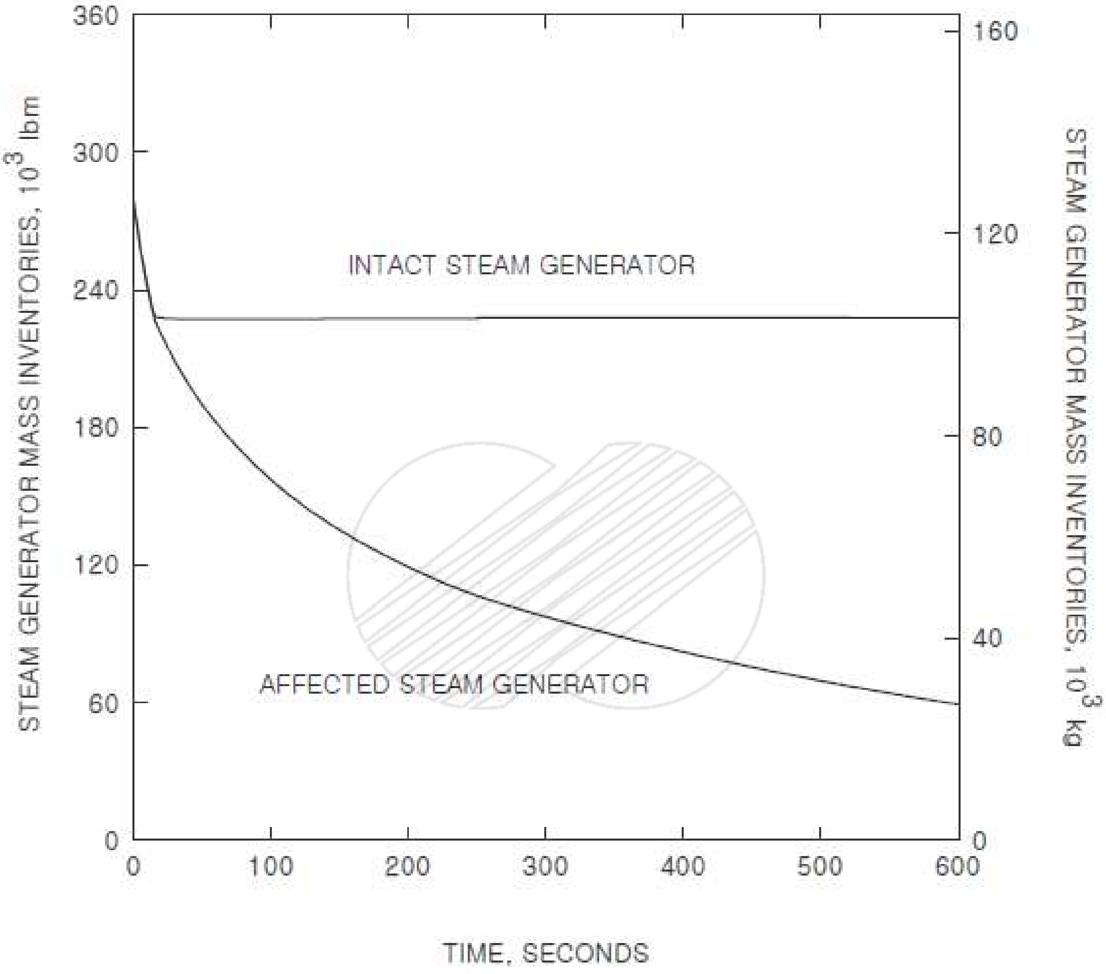

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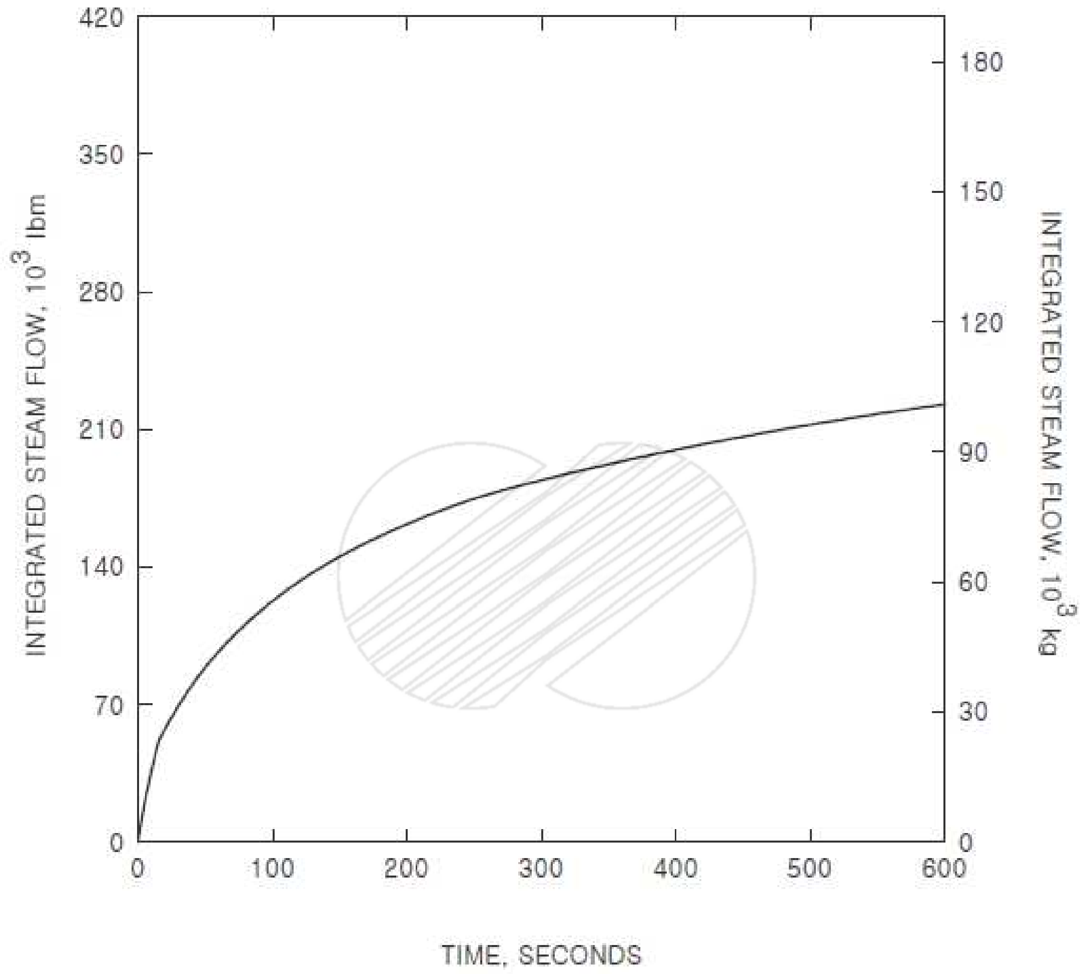
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
STEAM-GENERATOR MASS INVENTORIES VS. TIME

Figure 15.1.5-45



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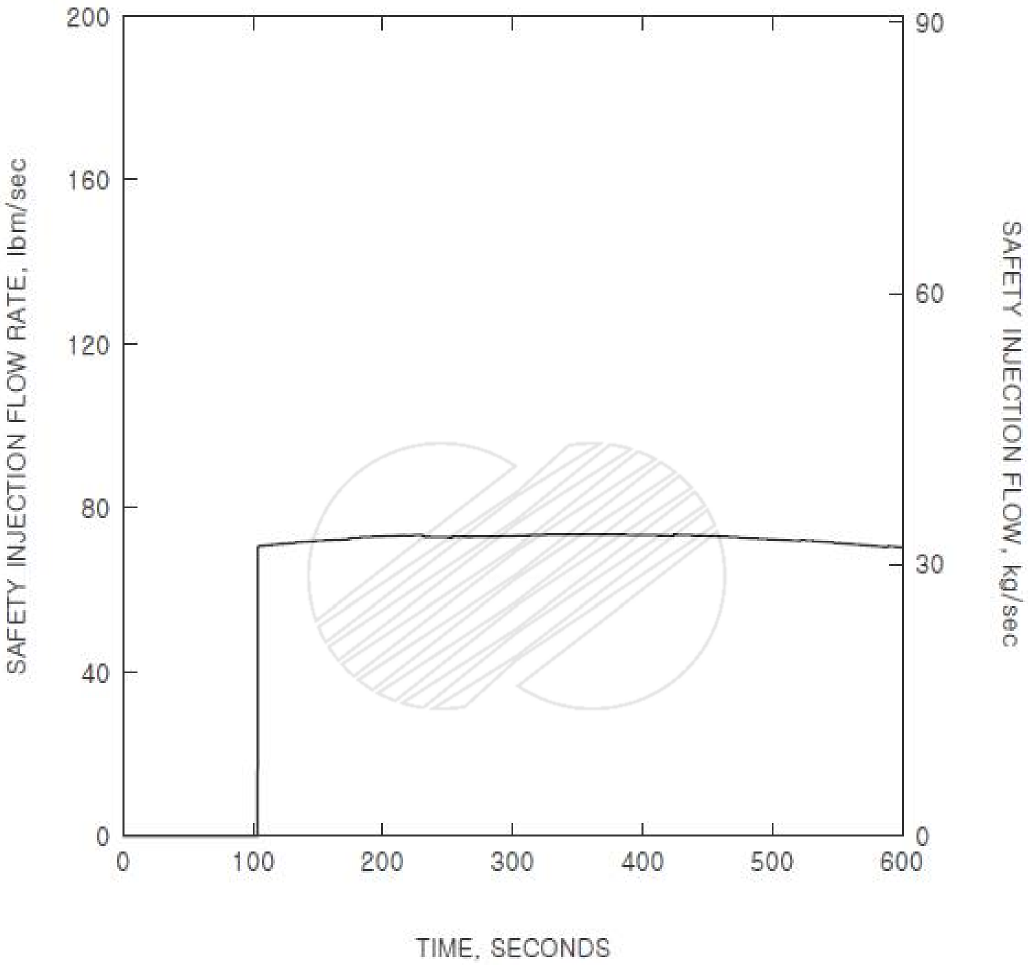
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
INTEGRATED STEAM MASS RELEASE
THROUGH BREAK VS. TIME

Figure 15.1.5-46



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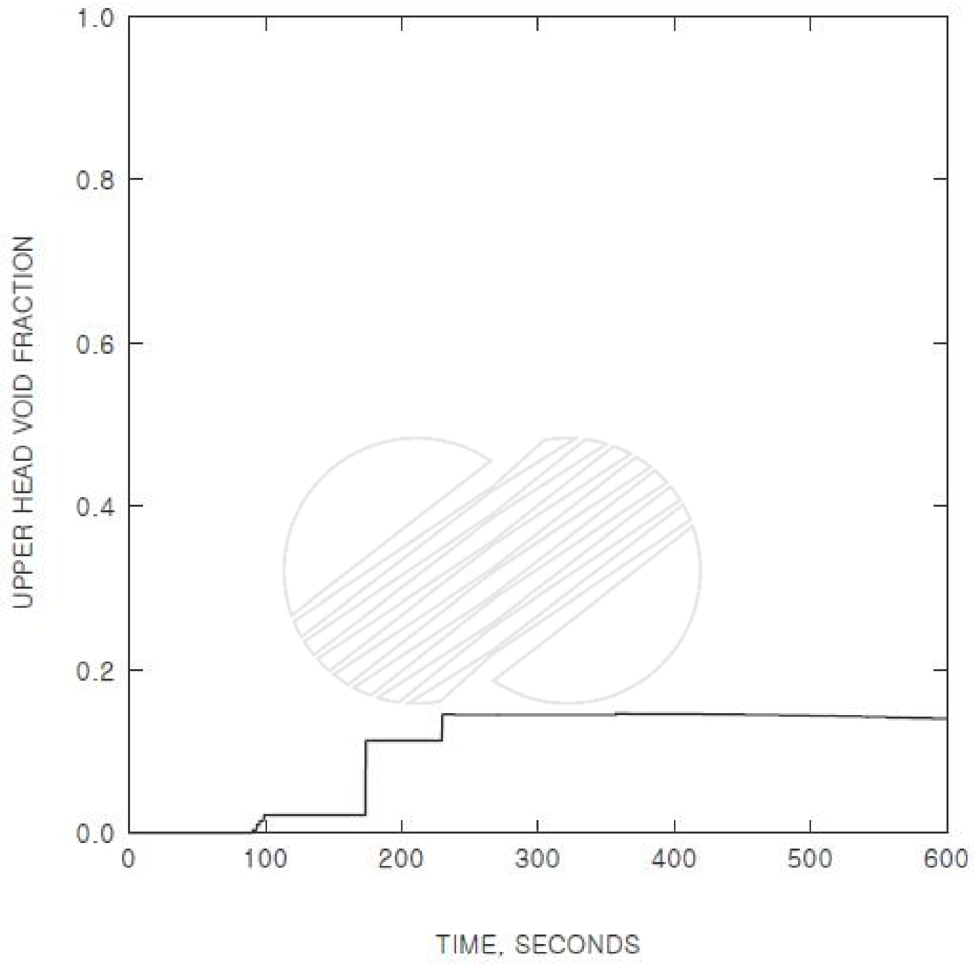
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS-OF-OFFSITE POWER:
SAFETY INJECTION FLOW VS. TIME

Figure 15.1.5-47



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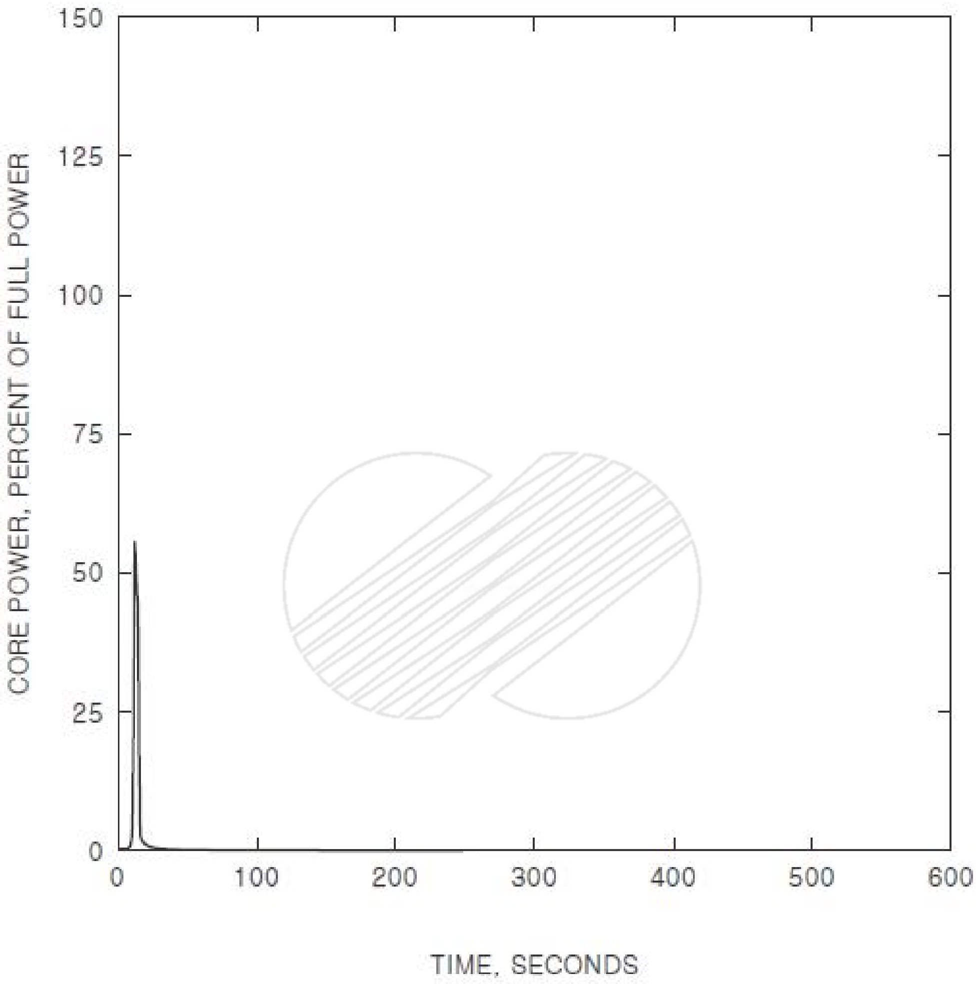
ZERO POWER LARGE STEAM LINE BREAK
WITH CONCURRENT LOSS–OF–OFFSITE POWER:
UPPER HEAD VOID FRACTION VS. TIME

Figure 15.1.5–48



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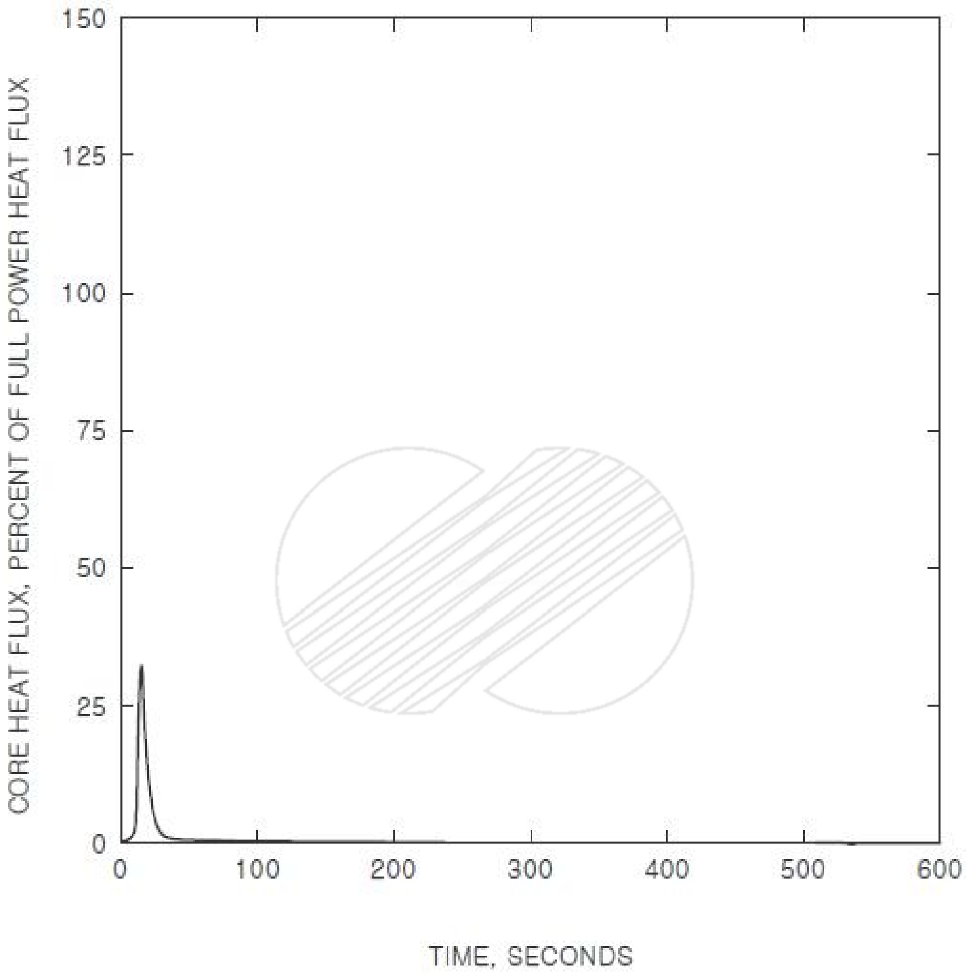
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
CORE POWER VS. TIME

Figure 15.1.5-49



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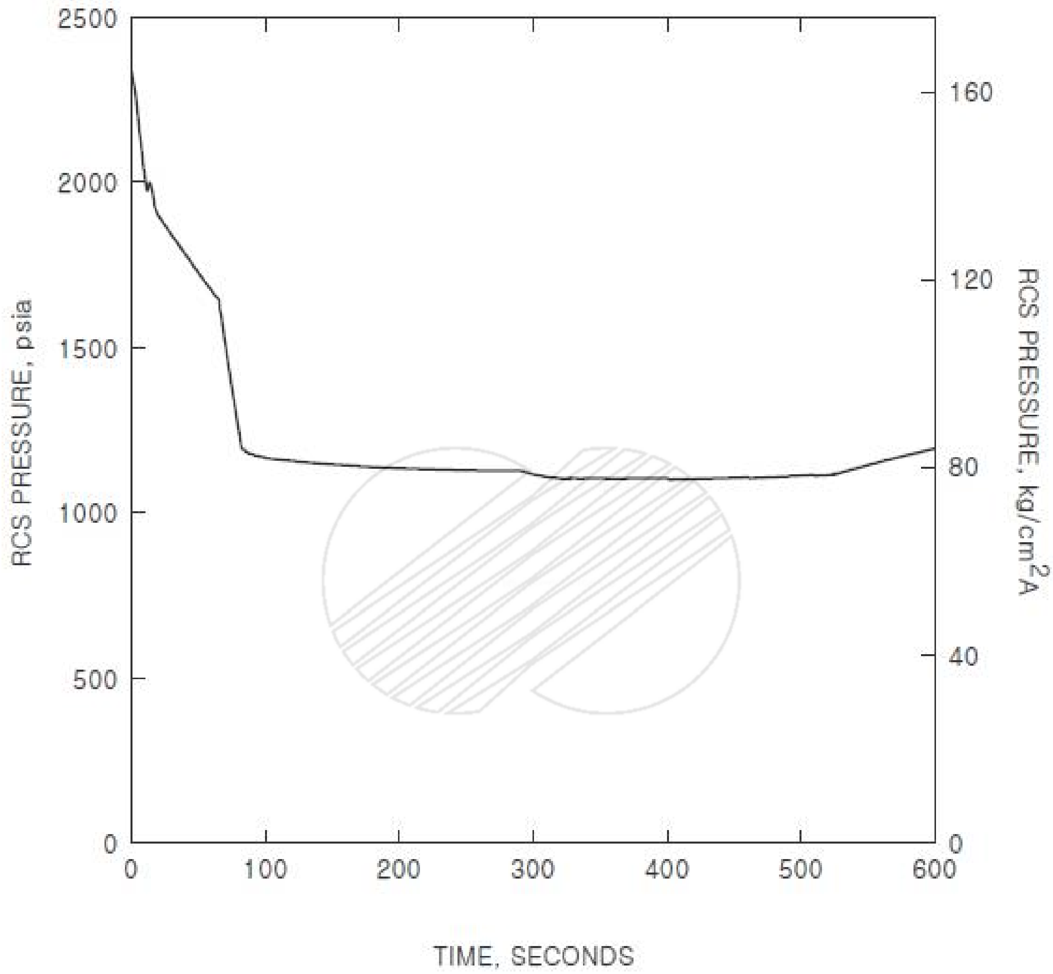
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
CORE HEAT FLUX VS. TIME

Figure 15.1.5-50



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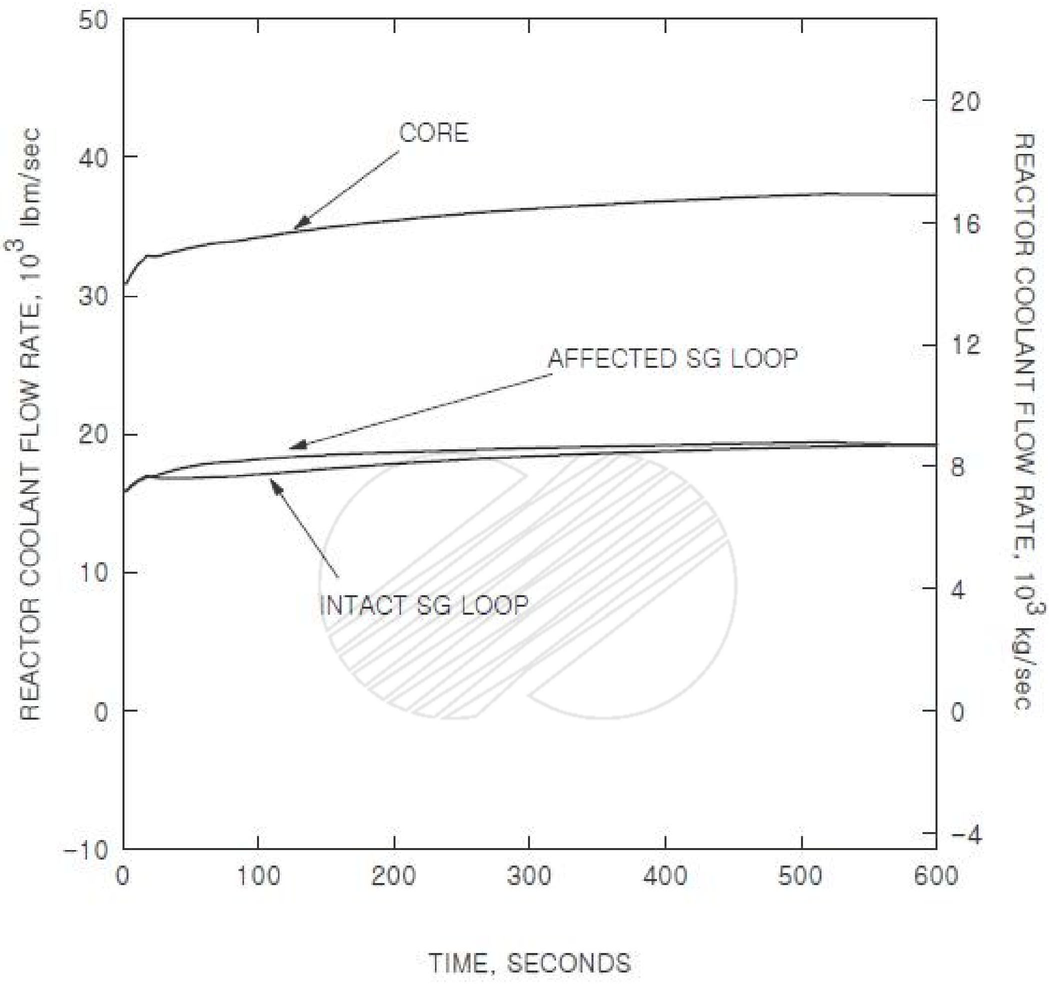
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
RCS PRESSURE VS. TIME

Figure 15.1.5-51



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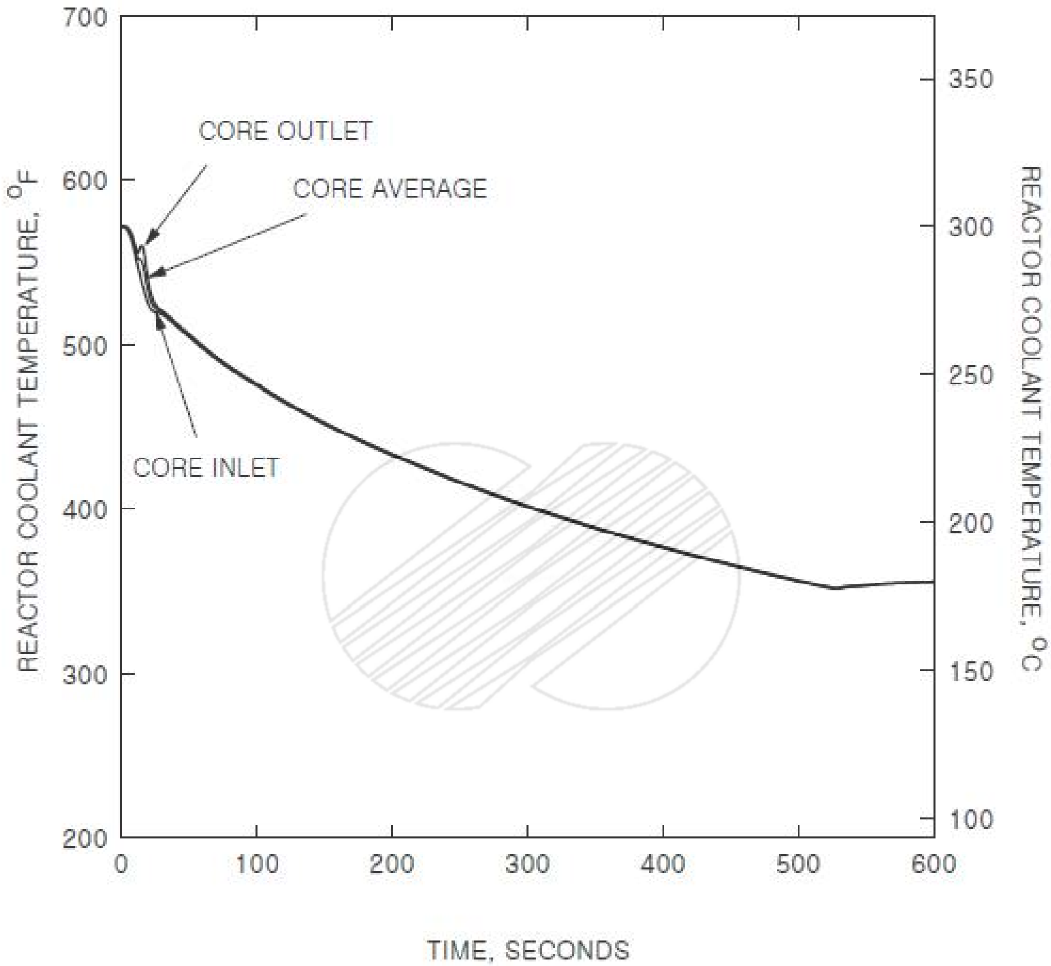
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
REACTOR COOLANT FLOW RATE VS. TIME

Figure 15.1.5-52



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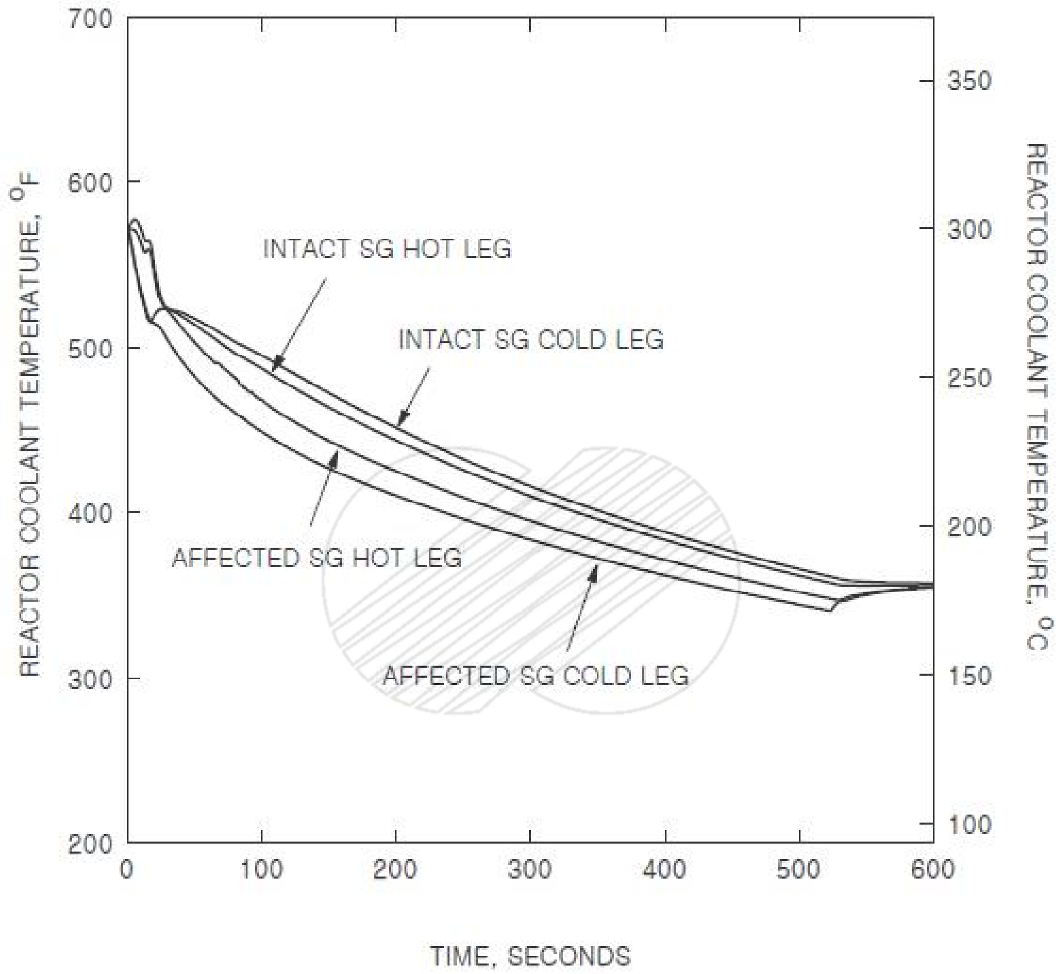
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
REACTOR COOLANT TEMPERATURES (A) VS. TIME

Figure 15.1.5-53



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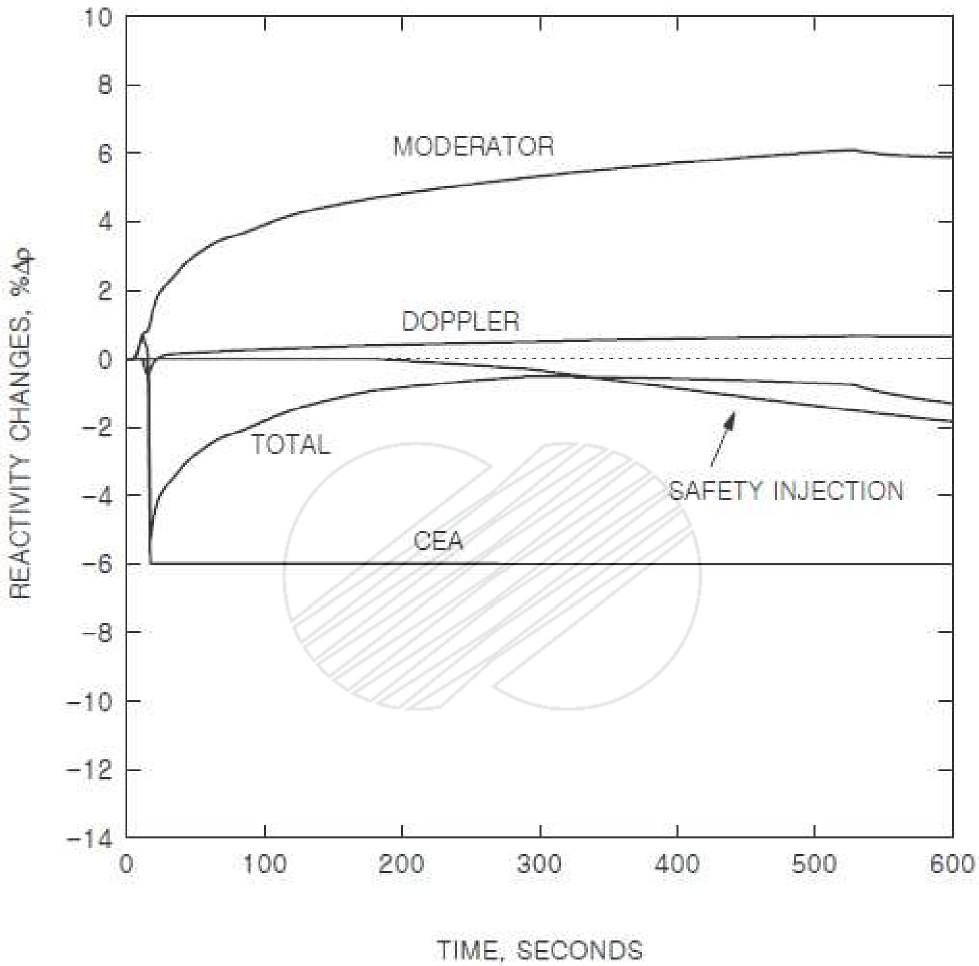
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
REACTOR COOLANT TEMPERATURES (B) VS. TIME

Figure 15.1.5-54



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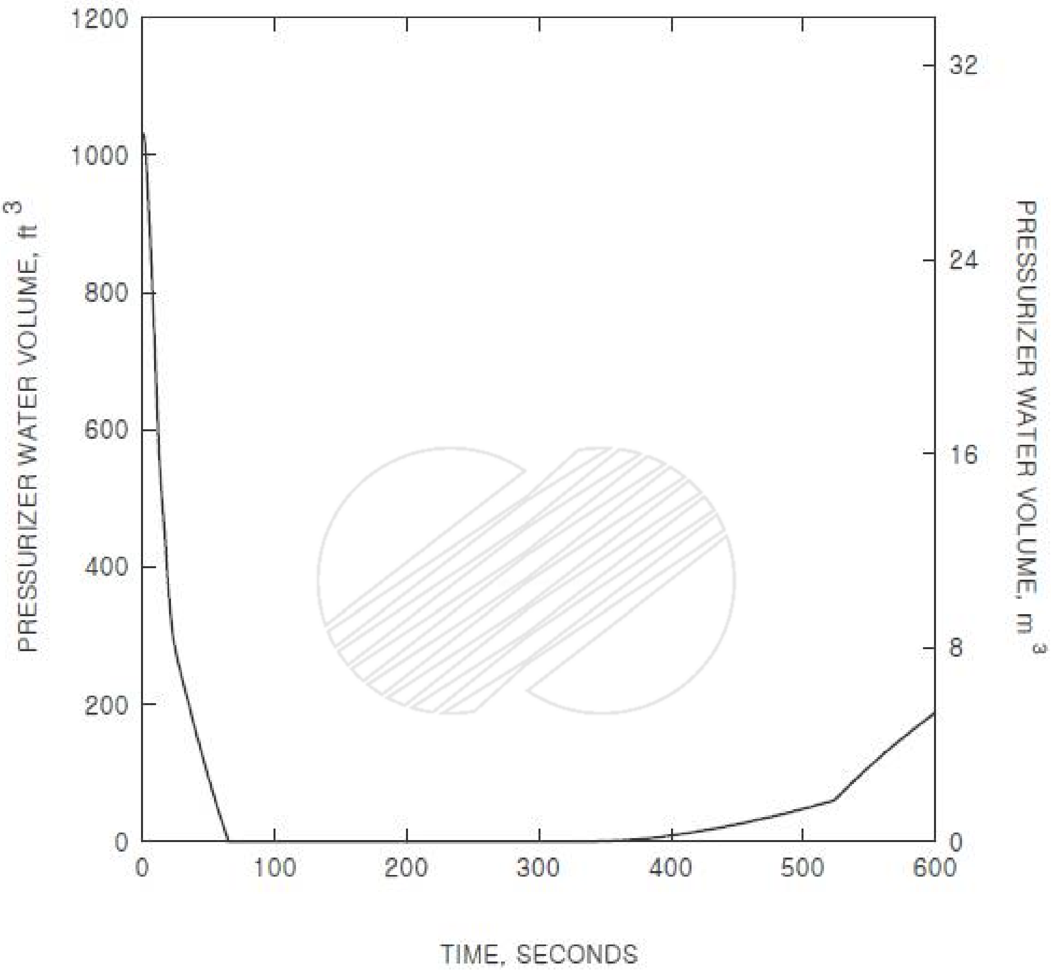
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
REACTIVITY CHANGES VS. TIME

Figure 15.1.5-55



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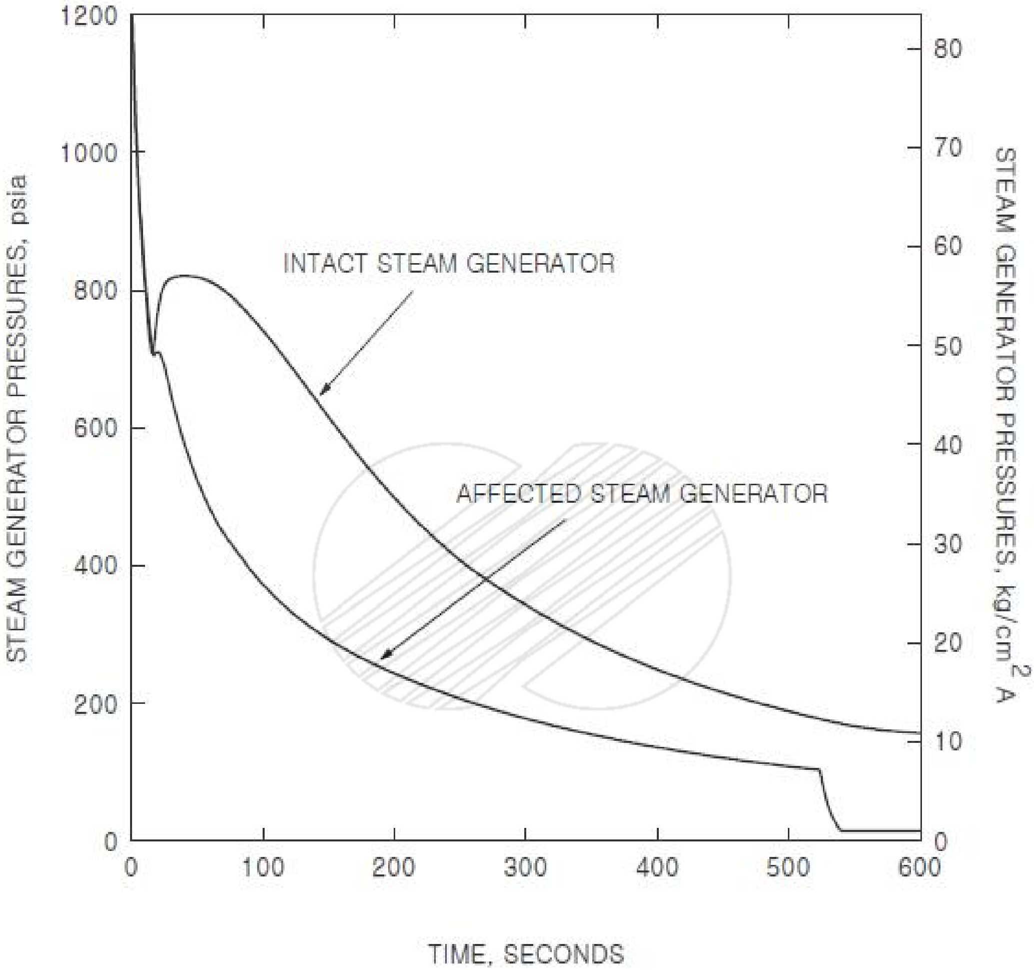
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
PRESSURIZER WATER VOLUME VS. TIME

Figure 15.1.5-56



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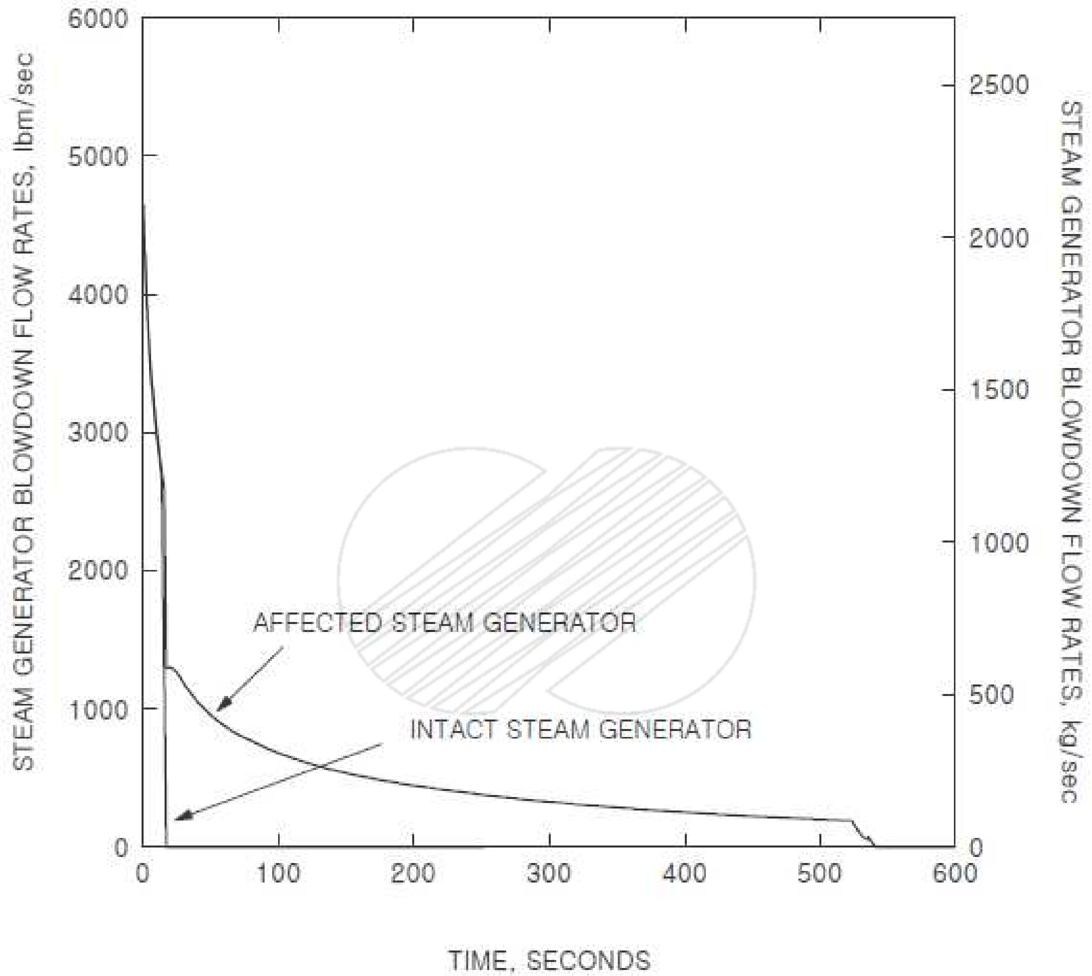
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
STEAM-GENERATOR PRESSURES VS. TIME

Figure 15.1.5-57



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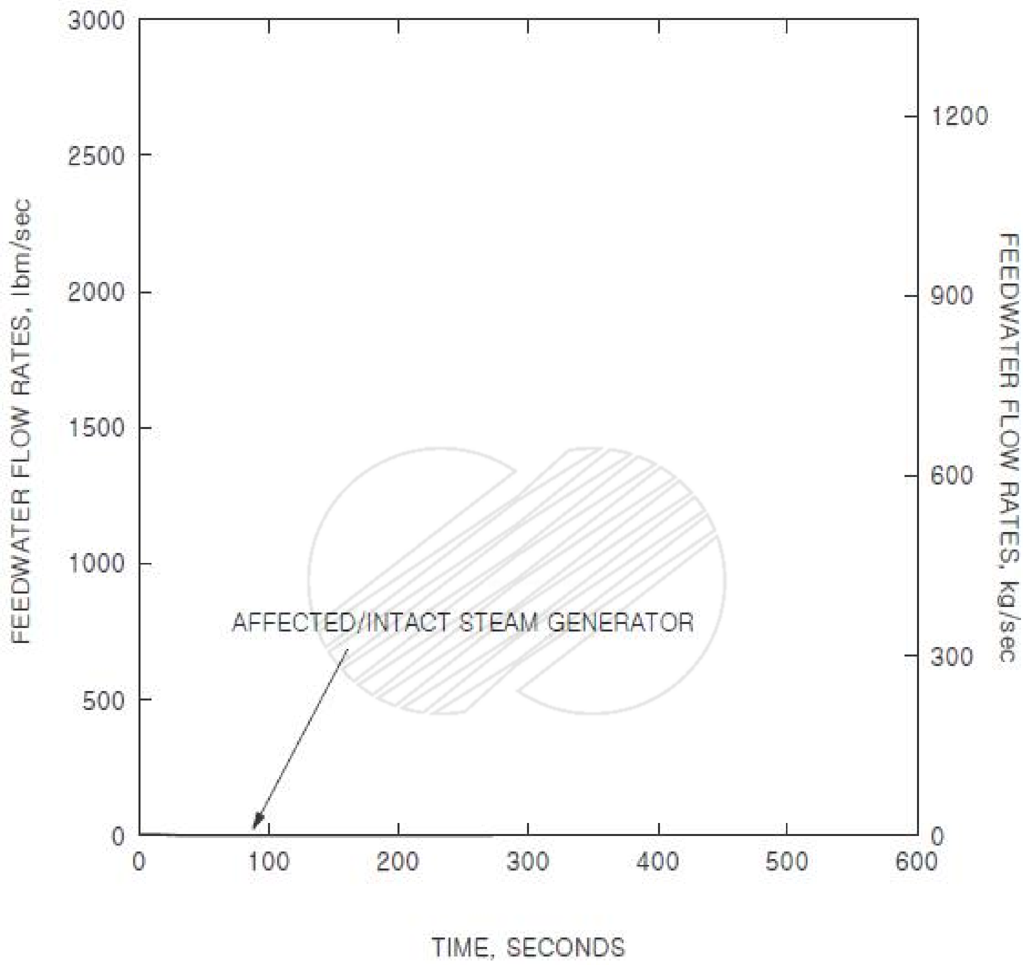
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
STEAM-GENERATOR BLOWDOWN RATES VS. TIME

Figure 15.1.5-58



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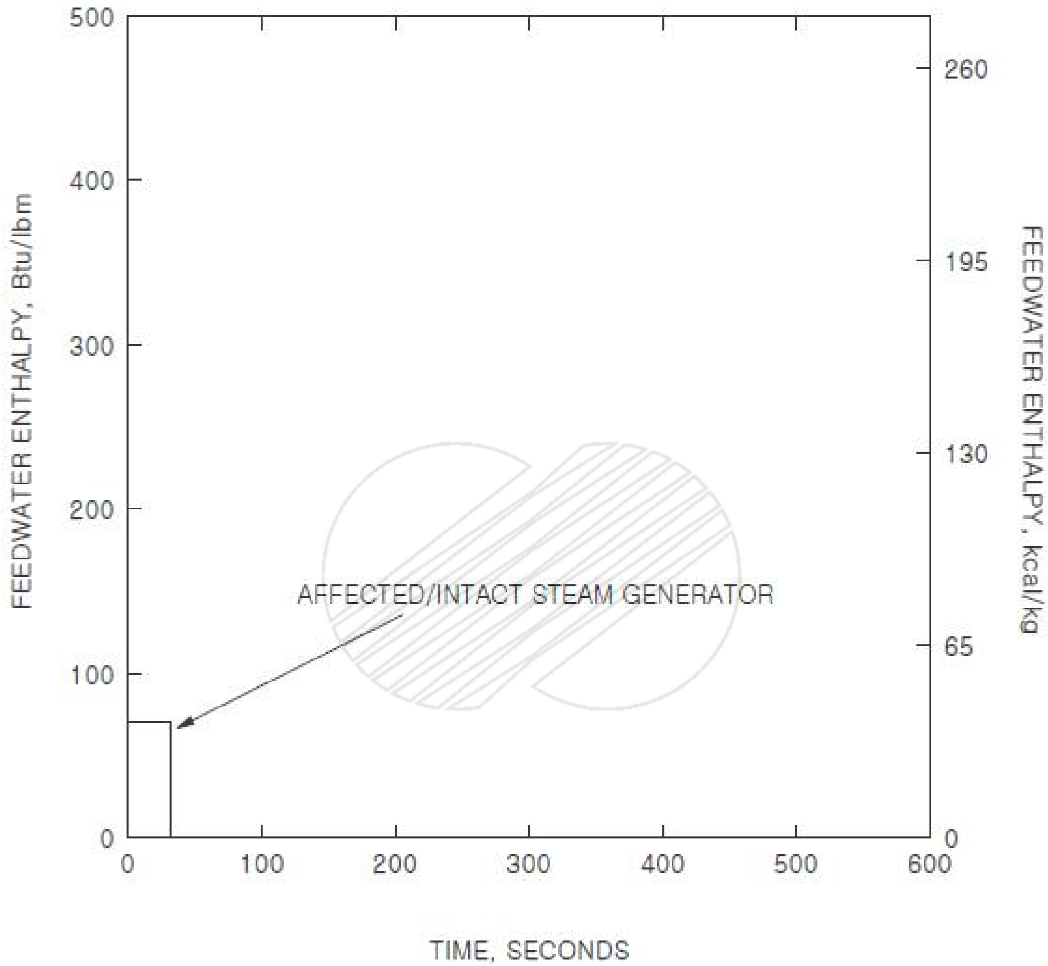
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
FEEDWATER FLOW RATES VS. TIME

Figure 15.1.5-59



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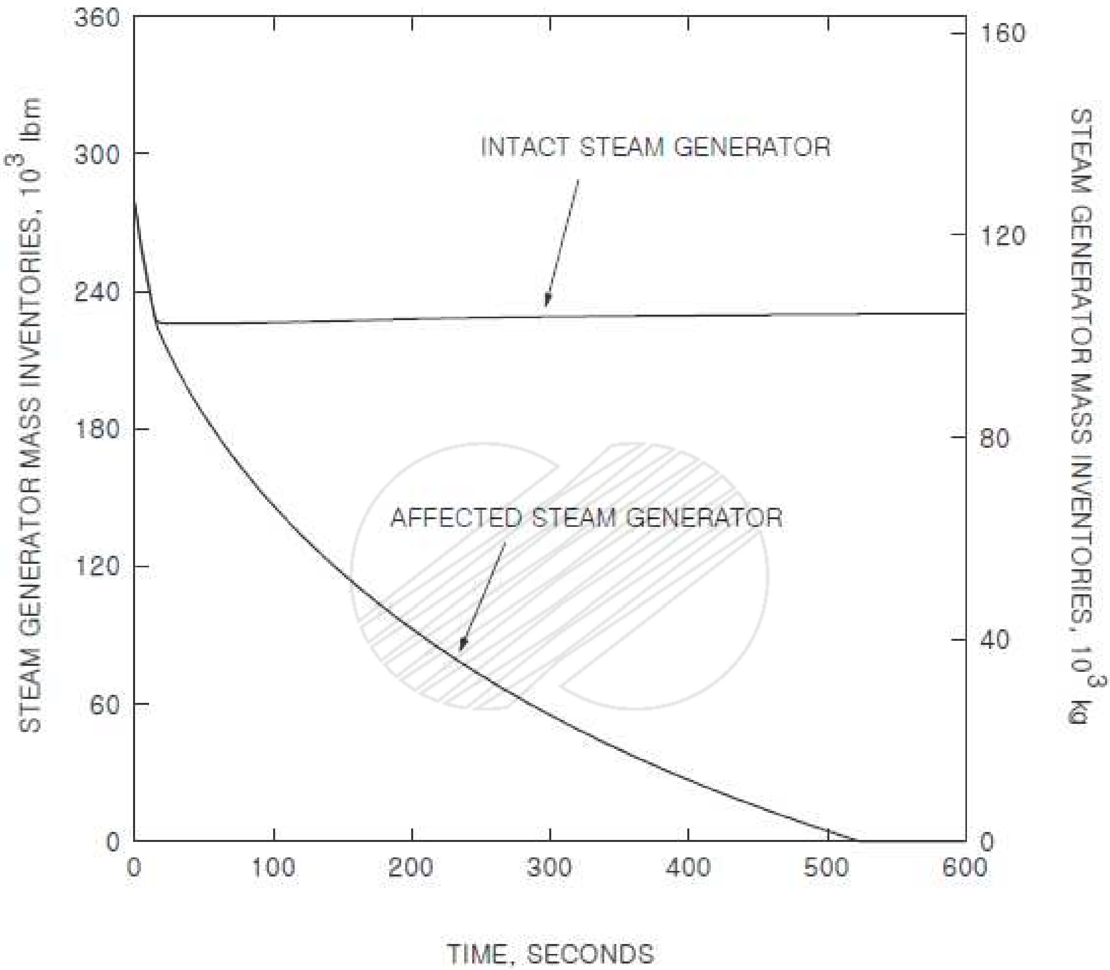
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
FEEDWATER ENTHALPY VS. TIME

Figure 15.1.5-60



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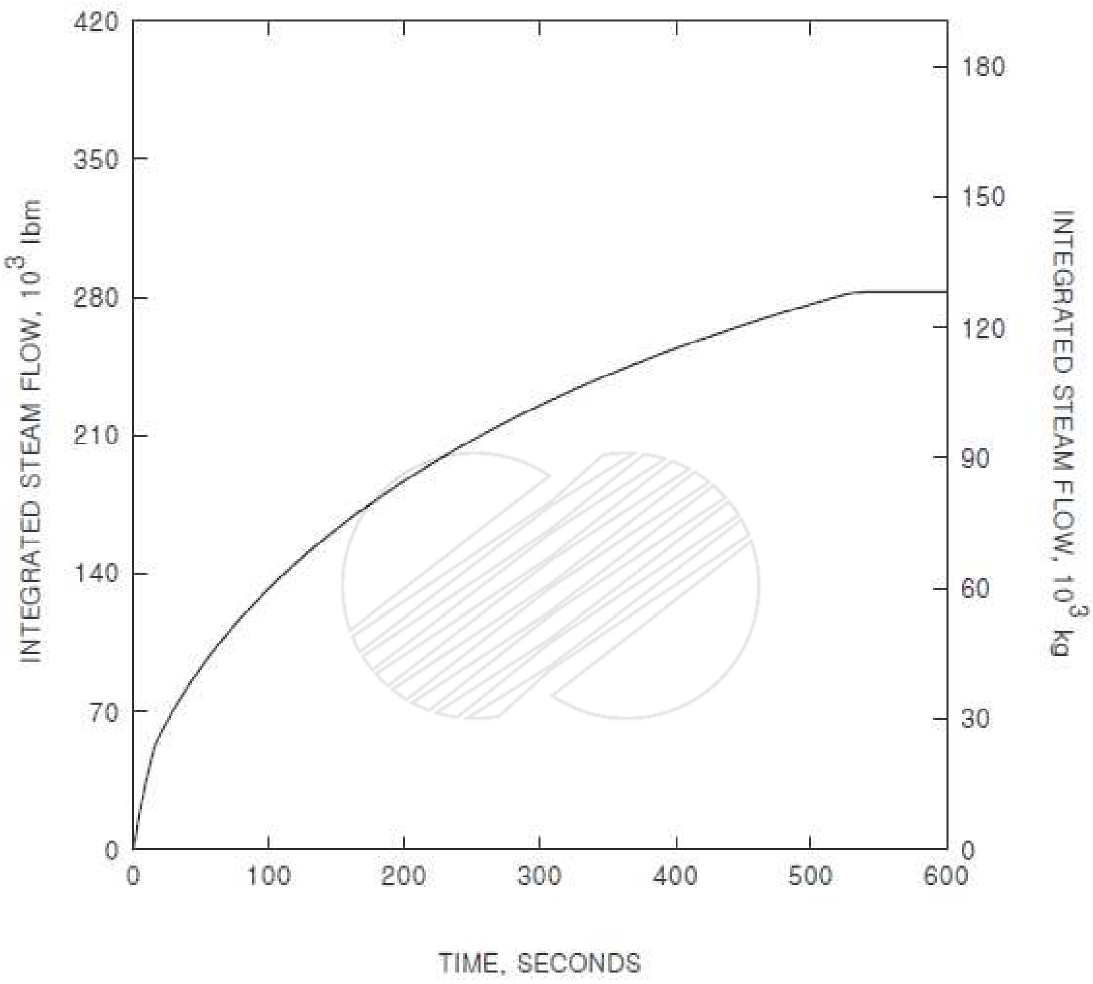
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
STEAM-GENERATOR LIQUID MASS VS. TIME

Figure 15.1.5-61



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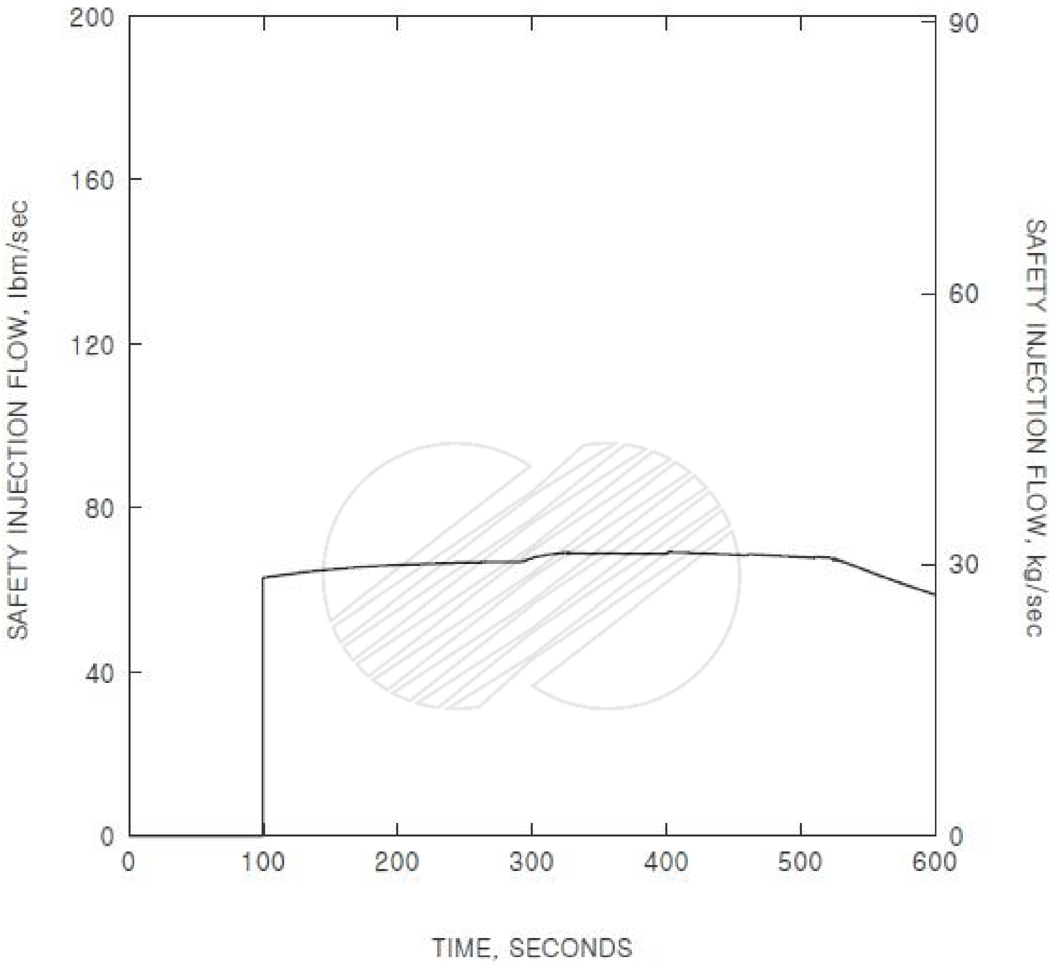
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
INTEGRATED STEAM RELEASE VS. TIME

Figure 15.1.5-62



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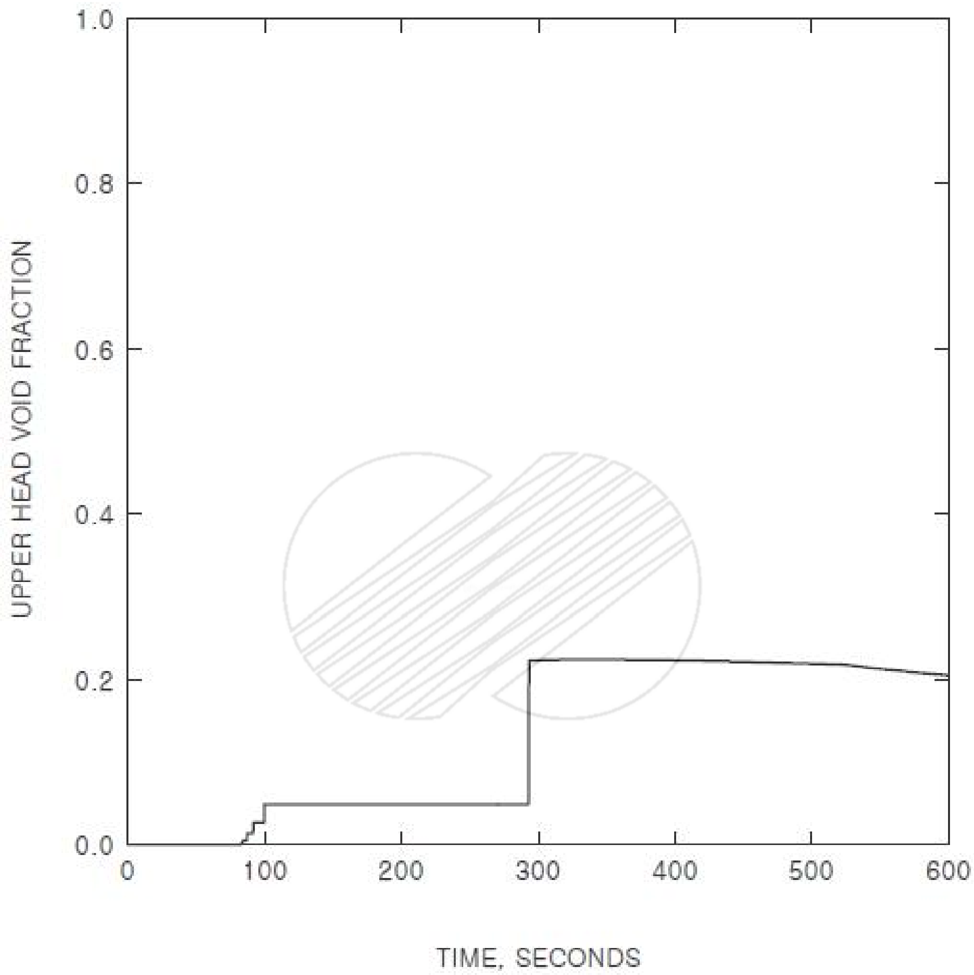
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
SAFETY INJECTION FLOW VS. TIME

Figure 15.1.5-63



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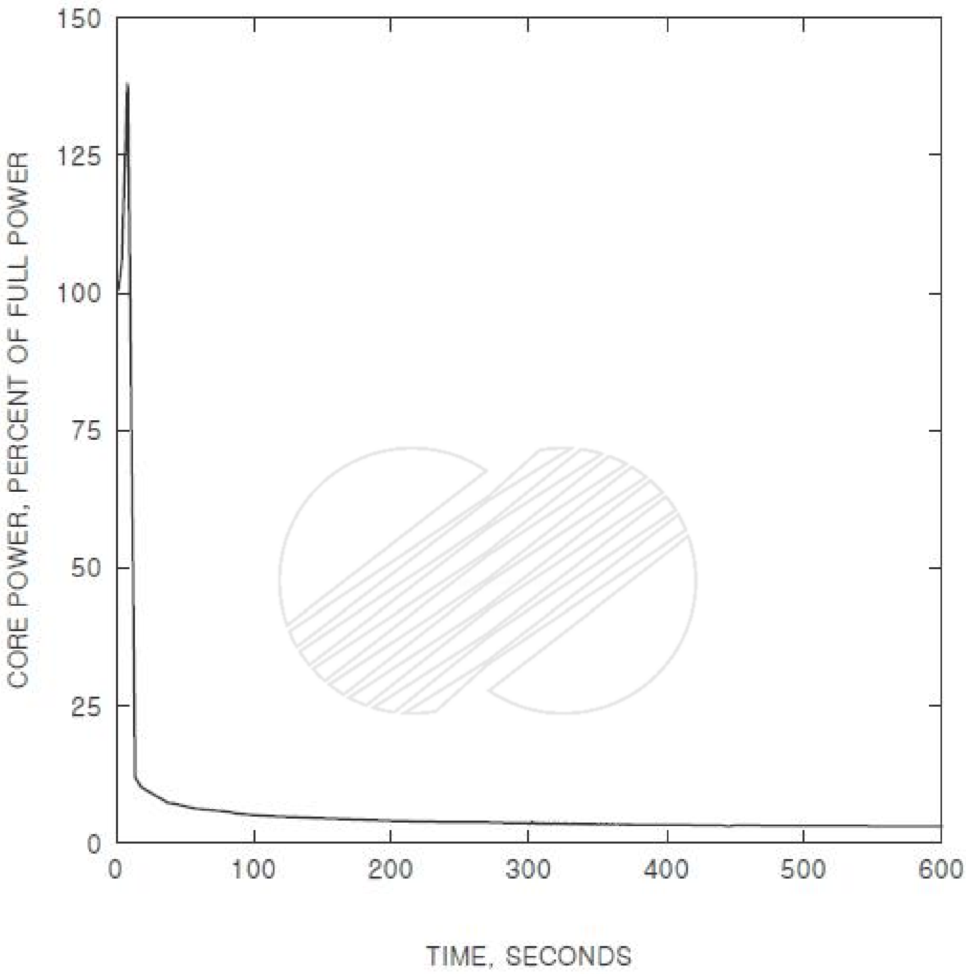
ZERO POWER LARGE STEAM LINE BREAK
WITH OFFSITE POWER AVAILABLE:
UPPER HEAD VOID FRACTION VS. TIME

Figure 15.1.5-64



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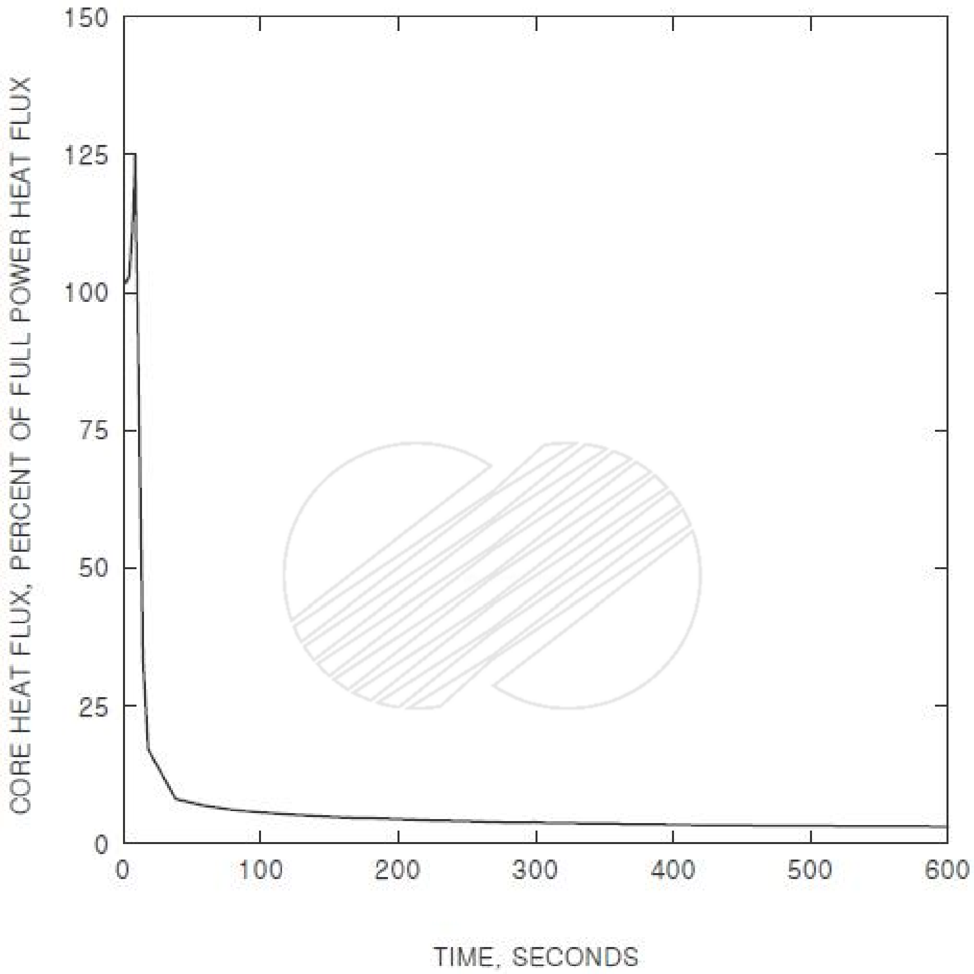
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: CORE POWER VS. TIME

Figure 15.1.5-65



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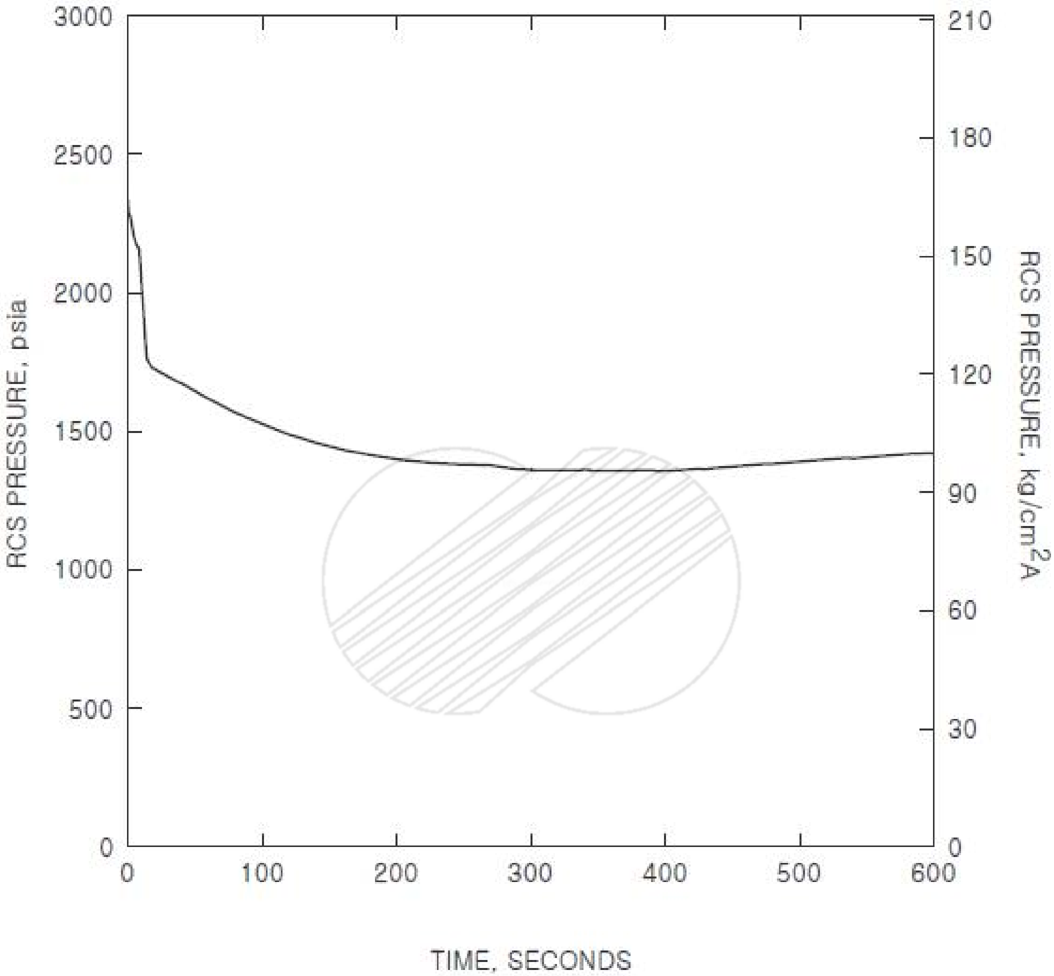
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: CORE HEAT FLUX VS. TIME

Figure 15.1.5-66



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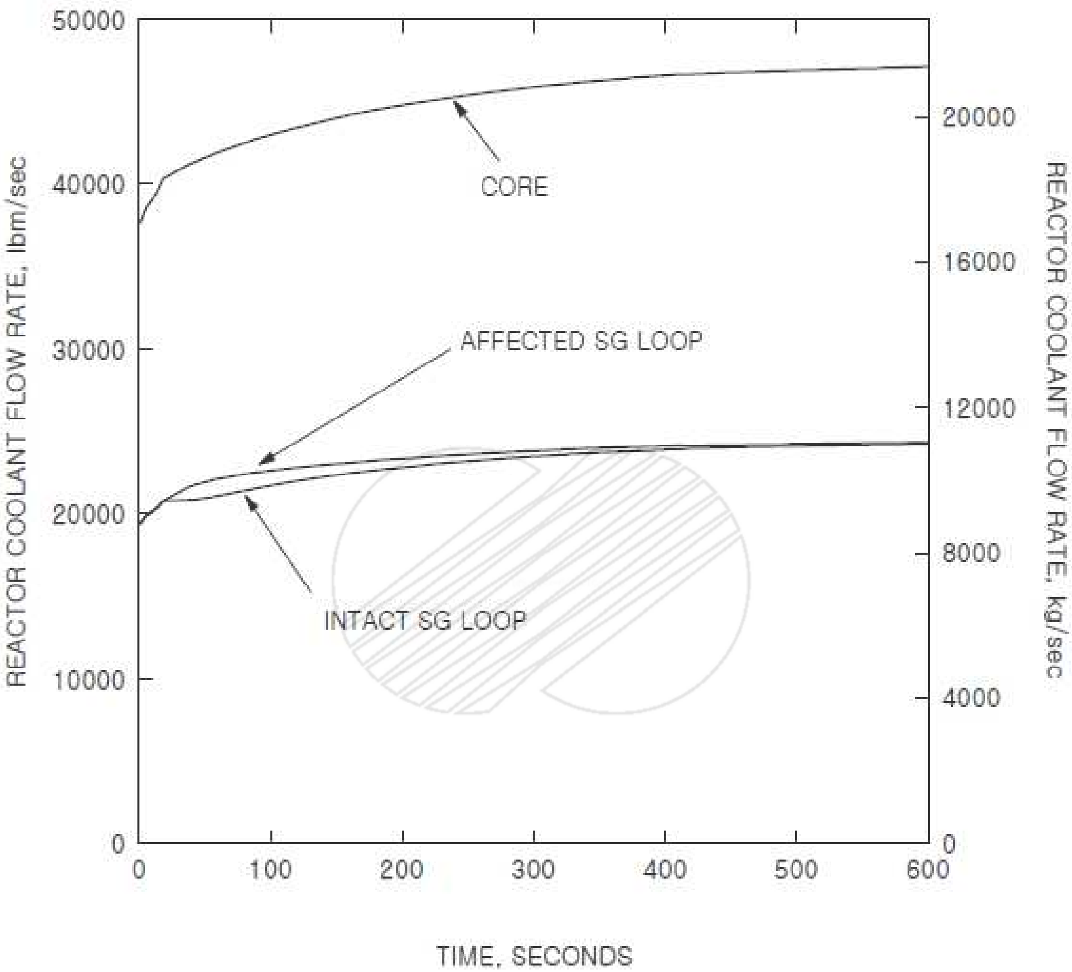
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: RCS PRESSURE VS. TIME

Figure 15.1.5-67



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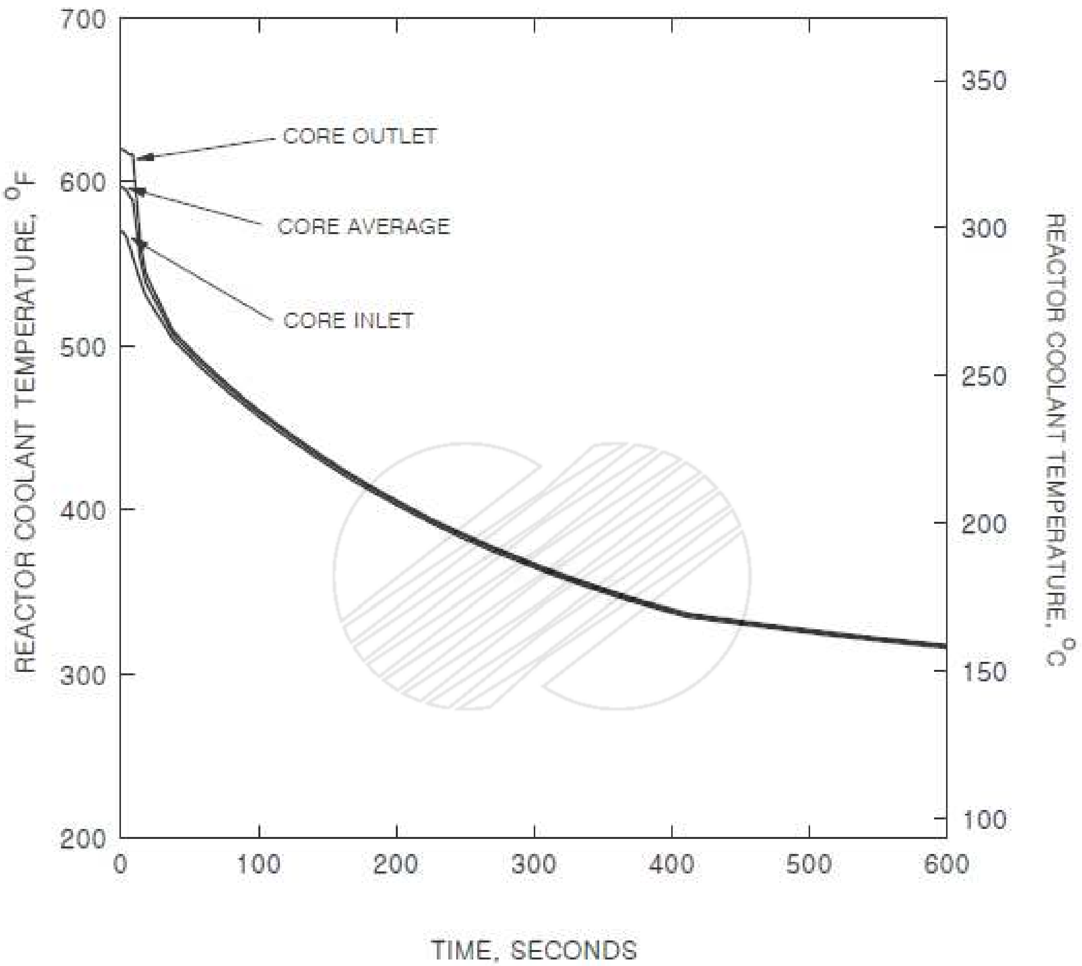
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: REACTOR COOLANT FLOW RATE VS. TIME

Figure 15.1.5-68



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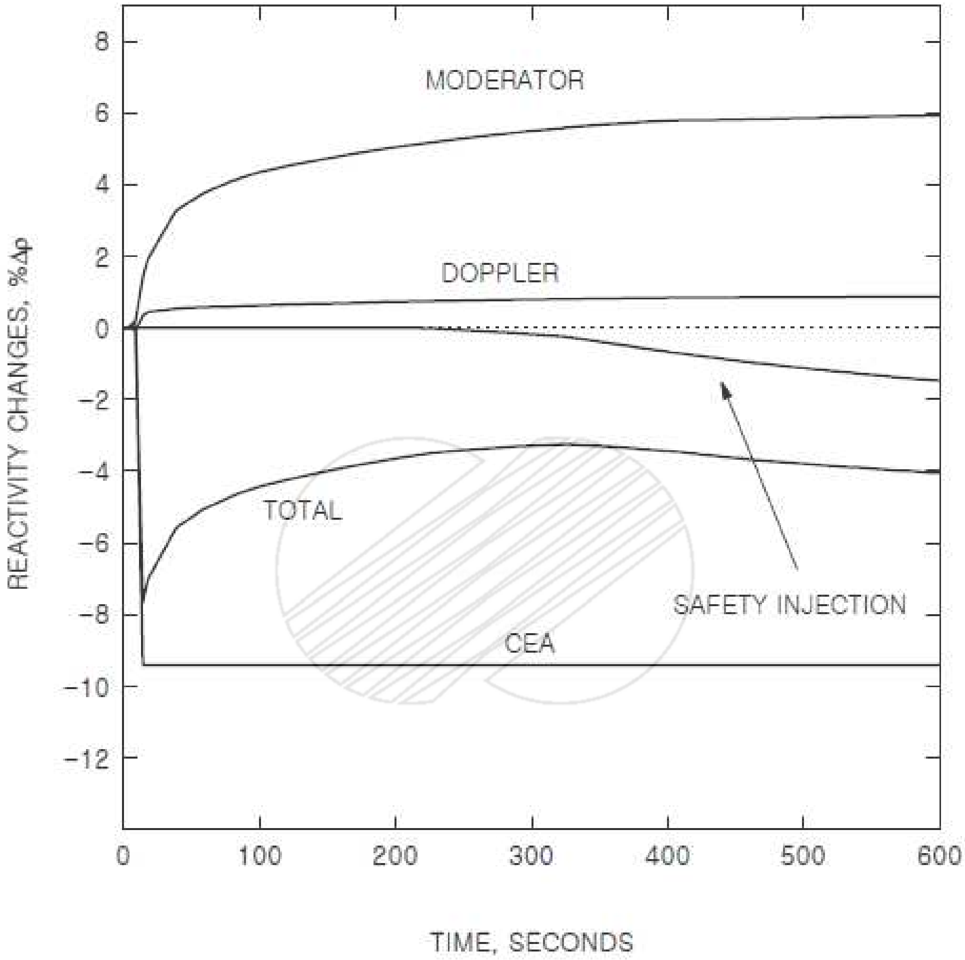
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: REACTOR COOLANT TEMPERATURES VS. TIME

Figure 15.1.5-69



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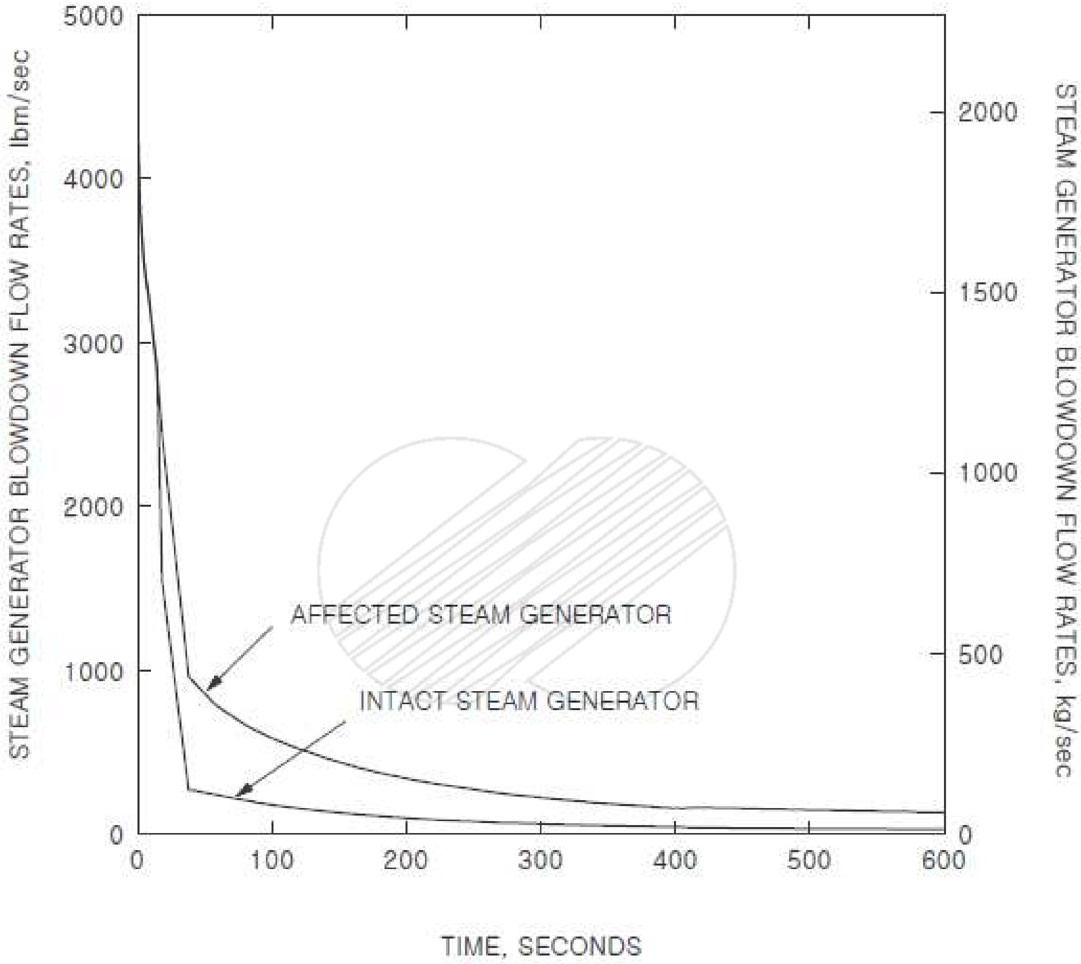
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: REACTIVITY CHANGES VS. TIME

Figure 15.1.5-70



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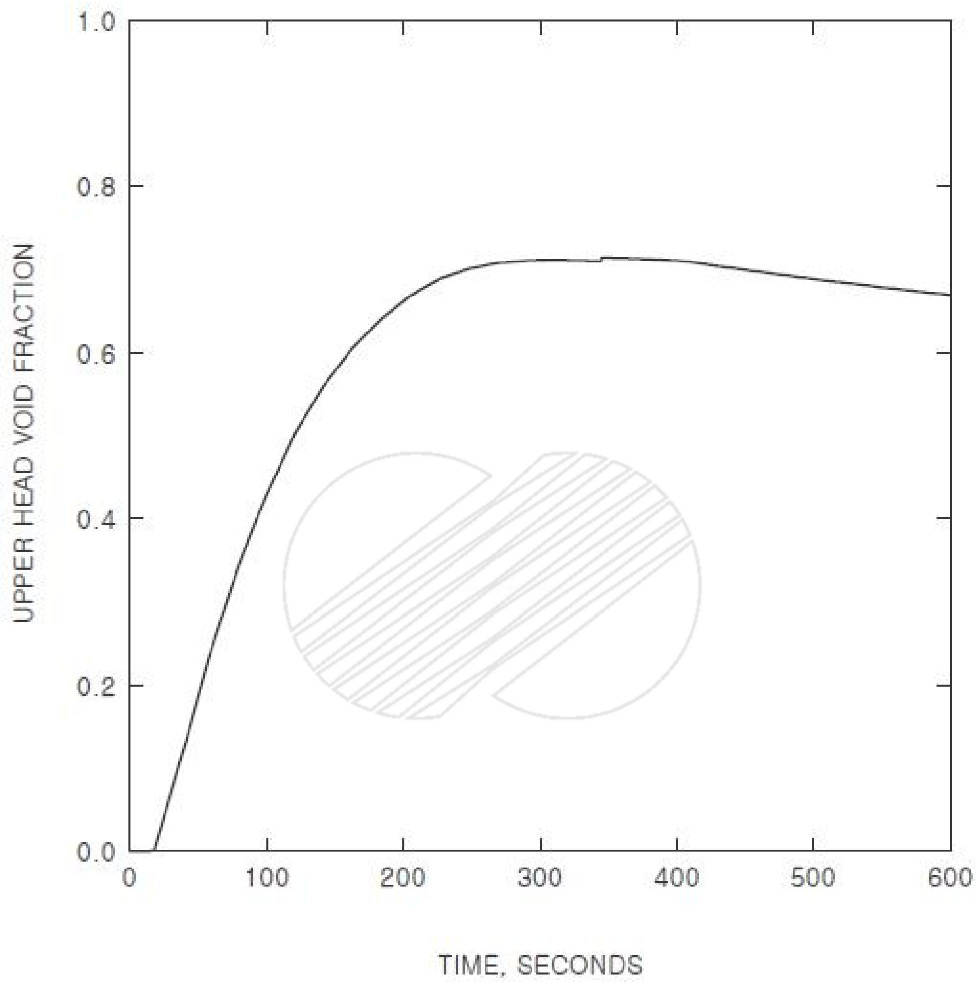
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: STEAM-GENERATOR BLOWDOWN RATES VS. TIME

Figure 15.1.5-71



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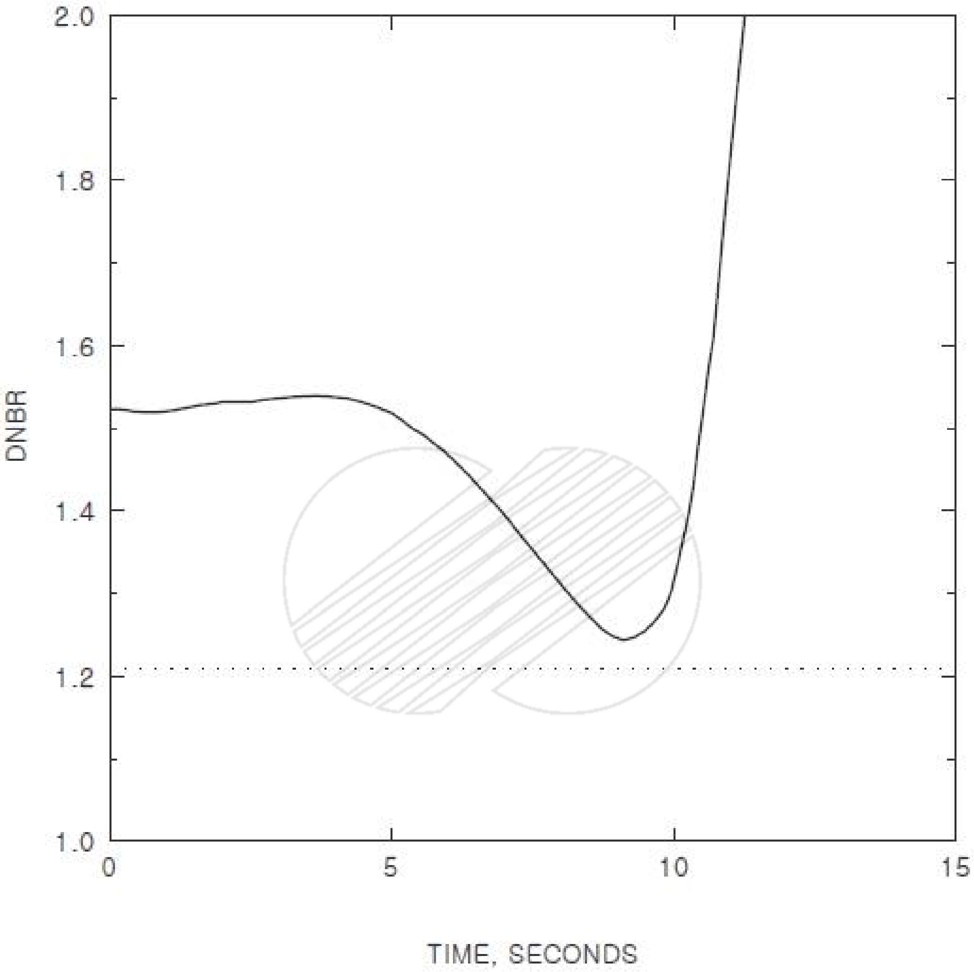
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: UPPER HEAD VOID FRACTION VS. TIME

Figure 15.1.5-72



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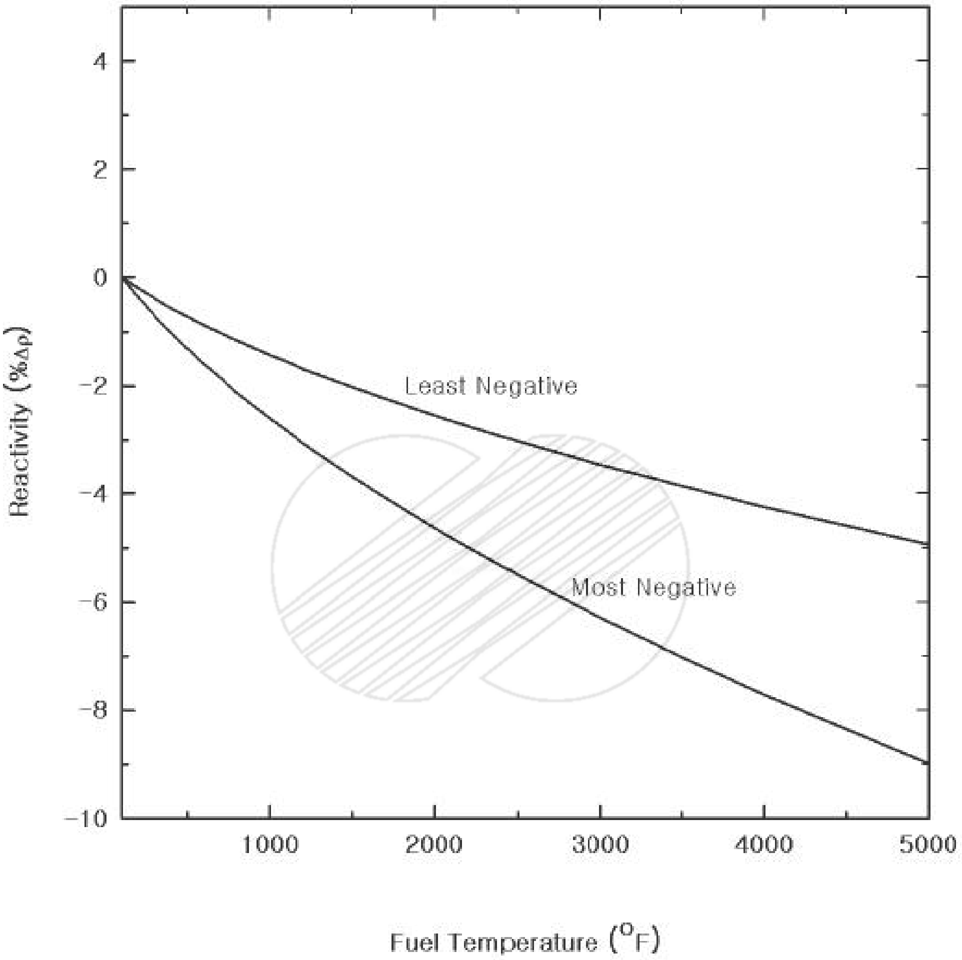
FULL POWER STEAM LINE BREAK OUTSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE: MINIMUM DNBR VS. TIME

Figure 15.1.5-73



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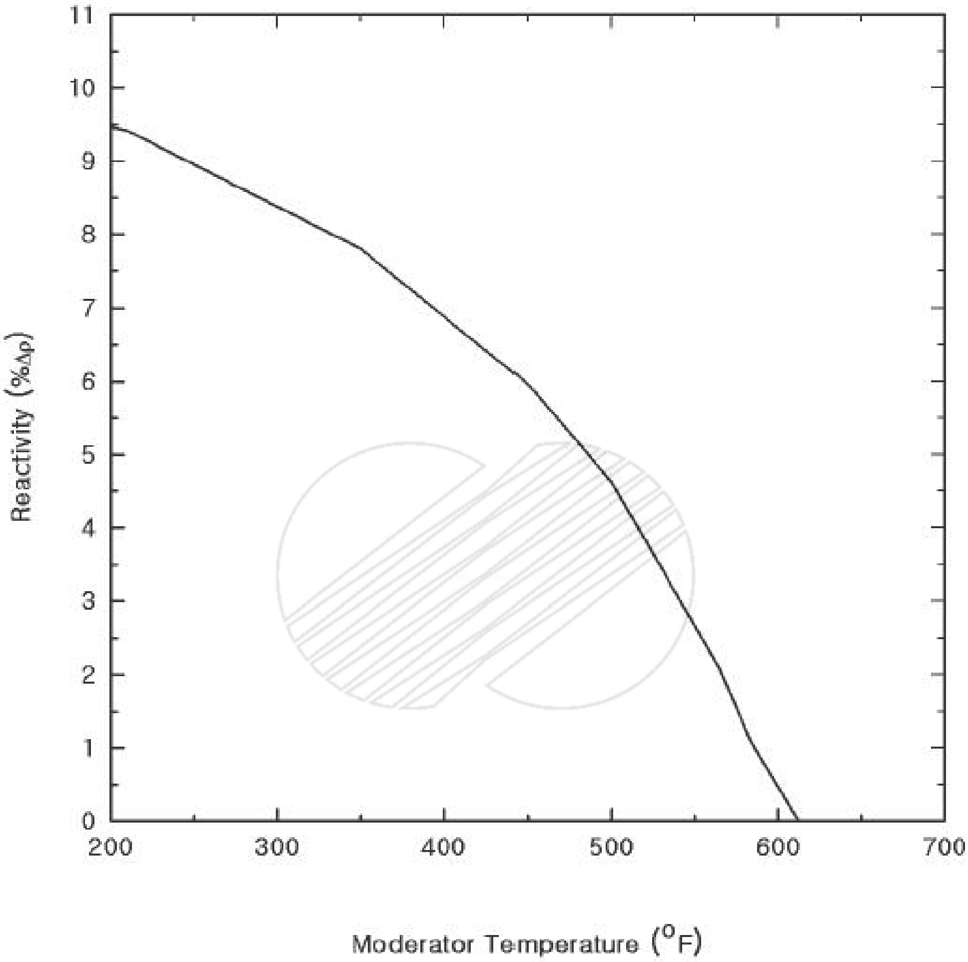
FUEL COOLDOWN CURVE FOR
YGN 3&4

Figure 15.1.5-74



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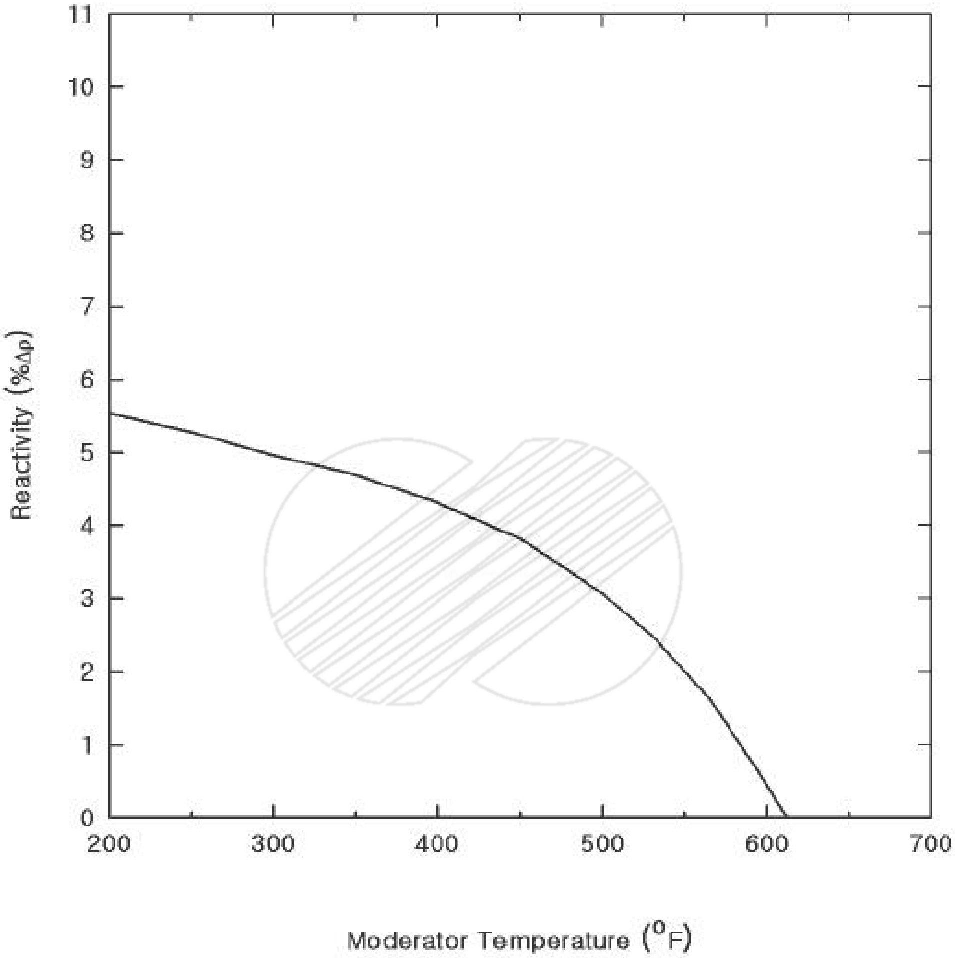
MODERATOR COOLDOWN CURVE FOR
YGN 3&4 (ALL RODS IN)

Figure 15.1.5-75



KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

MODERATOR COOLDOWN CURVE FOR
YGN 3&4 (ALL RODS OUT)

Figure 15.1.5-76