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IRIS
Preliminary Safety Assessment

Westinghouse

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WESTINGHOUSE NON-PROPRIETARY CLASS 3

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IRIS Preliminary Safety Assessment
-- Volume I --

July 15, 2003

Westinghouse Electric Company LLC
Science and Technology Department
1344 Beulah Road
Pittsburgh, PA 15235-5083

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Foreword

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

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

For the analyses presented in this report the RELAP5 Mod 3.3 computer code has been adopted. This does not necessarily represent a final selection of the computer code that will be used for IRIS safety analyses during the design certification phase. The worldwide used RELAP code was selected in this phase to simplify the collaboration among member organizations of the IRIS consortium. Analyses will be updated during the development phase of the IRIS Evaluation Models, and a final selection of computer programs to be used in the analyses shall be completed. Also, activities are in progress to define "IRIS Evaluation Models for Small Break LOCA Safety Analyses", where particular emphasis is given to the development of an appropriate procedure and code selection to analyze the coupled IRIS reactor vessel and containment.

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

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

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ACRONYMS AND ABBREVIATIONS USED IN THIS REPORT

Abbreviation	Description
AC	Alternating Current
ADS	Automatic Depressurization System
ATWS	Anticipated Transients Without Scram
CFR	Code of Federal Regulations
CPSS	Containment Pressure Suppression System
CRDM	Control Rod Drive Mechanism
CV	Containment Vessel
DCD	Design Control Document
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DP	Pressure Drop
DVI	Direct Vessel Injection
EBS	Emergency Boration System
EBT	Emergency Boration Tank
EHRS	Passive Emergency Heat Removal System
ELI	Excessive increase in secondary steam flow
EMDAP	Evaluation Model Development And Assessment Procedure
ESF	Engineered Safety Features
$F_{\Delta H}$	Nuclear Enthalpy Rise Hot Channel Factor
FLB	Feedwater system pipe break
Fq	Total Peaking Factor
FWM	Feedwater system malfunction
IRIS	International Reactor Innovative and Secure
IVR	In-Vessel Retention
LOCA	Loss of Coolant Accident
LOFA	Loss of flow accident
LOL	Loss of external electrical load
LONF	Loss of normal feedwater
LOOP	Loss of a/c power to the station auxiliaries
LRSS	Reactor coolant pump locked rotor/shaft seizure
MDNBR	Minimum Departure From Nucleate Boiling
MFIV	Main Feedwater Isolation Valve
MSIV	Main Steam Isolation Valve
NSSS	Nuclear Steam Supply System
PCCS	Passive Containment Cooling System
PIRT	Phenomena Identification and Ranking Table
PMS	Protection And Safety Monitoring System
PRA	Probabilistic Risk Assessment

Abbreviation	Description
PSS	Pressure Suppression System
PWR	Pressurized Water Reactor
RCCA	Rod Control Cluster Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RTDP	Revised Thermal Design Procedure
RV	Reactor Vessel
RWAP	Uncontrolled RCCA bank withdrawal at power

Abbreviation	Description
RWFS	Uncontrolled RCCA bank withdrawal from a subcritical
RWST	Refueling Water Storage Tank
SAFDL	Specified Acceptable Fuel Design Limits
SAR	Safety Analysis Report
SG	Steam Generator
SGTR	Steam generator tube rupture
SLB	Steam system piping failure
SRP	Standard Review Plan
TT	Turbine trip

1 IRIS SAFETY FEATURES

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1.0 OVERVIEW OF THE IRIS SAFETY PHILOSOPHY

The IRIS approach to safety has been primarily focused on achieving a design with innovative safety characteristics and multiple levels of defense for accident mitigation (defense-in-depth), resulting in extremely low core damage probabilities while minimizing the occurrences of containment flooding, pressurization, and heat-up situations.

The first line of defense in the IRIS defense-in-depth approach is to eliminate initiators that could eventually lead to core damage. This concept is implemented through the "safety by design" approach, which can be simply described as designing the plant in such a way to prevent the accidents from occurring, rather than coping with their consequences. If it is not possible to eliminate the accidents altogether, then the design should be such to inherently limit their consequences and/or their probability of occurring. The key difference from previous practice is that the integral reactor design is intrinsically conducive to eliminating accidents to a degree impossible in conventional loop-type reactors. The elimination of the large LOCAs, since no large primary penetrations of the reactor vessel or large loop piping exist, is the most easily visible of the safety potential characteristics of integral reactors. Many others are possible, but they must be carefully exploited through an appropriate design that is kept focused on selecting design characteristics that are most amenable to eliminate initiating events. IRIS has strived to achieve that and the main results are summarized in Table 1.0-1 which illustrates the implications of the safety by design approach, and in Table 1.0-2, that describes the effect of safety by design on some typical design basis events for LWRs. The features summarized in Tables 1.0-1 and 1.0-2 are discussed in Section 1.1.1. A substantial effort is being exerted to qualify and perform safety analyses and quantitatively substantiate the behavior summarized in Tables 1.0-1 and 1.0-2. Section 2 of this document reports the results to date of this effort.

The IRIS defense-in-depth capability next includes multiple levels of defense for a very wide range of plant events, similar to AP600/AP1000. Defense-in-depth is built into the IRIS design, where the design goal is to always maintain the core covered with water and avoid fuel damage, with a multitude of individual plant features capable of providing some degree of defense of plant safety. In addition to the safety by design approach, other aspects of the IRIS design that contribute to defense-in-depth are discussed in the following. Some are practices common to other reactors, others are similar to the AP600/AP1000 advanced designs, and finally some are exclusive features of IRIS:

<u>Stable Operation</u>. In normal operation, the most fundamental level of defense-in-depth is to ensure that the plant can be operated stably and reliably. As in most reactors, this is achieved by designing a reactor core that promotes stable, self-correcting reactivity feedback coefficients that eliminate the potential for rapid, uncontrolled power excursions; and by an advanced control system and plant design that provide substantial margins for plant operation before approaching safety limits. A high level of safety will be also achieved by the selection of materials, by quality assurance during design and construction, and by well-trained operators during operation.

<u>Physical Plant Boundaries</u>. One of the most recognizable aspects of defense-in-depth is the protection of public safety through the physical plant boundaries. Releases of

Table 1.0-1 Implications of Safety By Design Approach

IRIS Design Characteristic	Safety Implication	Accidents Affected
Integral Layout	No large primary piping	- LOCAs
Large, Tall Vessel	Increased water inventory Increased natural circulation Can accommodate internal CRDMs	- LOCAs - Decrease in heat removal - Various events - RCCA ejection, eliminate head penetrations
Heat Removal from inside the vessel	Depressurizes primary system by condensation and not by loss of mass Effective heat removal by SG/EHRS	- LOCAs - LOCAs - All events for which effective cooldown is required - ATWS
Reduced size, higher design pressure containment	Reduced driving force through primary opening	- LOCAs
Multiple coolant Pumps	Decreased importance of single pump failure	- Locked rotor, shaft seizure/break
High design pressure steam generator system	No SG safety valves Primary system cannot over-pressure secondary system Feed/Steam System Piping designed for full RCS pressure reduces piping failure probability	 - Steam generator tube rupture - Steam line break - Feed line break
Once Through steam generator	Limited water inventory	- Steam line break - {Feed line break}
Integral Pressurizer	Large pressurizer volume/reactor power	- Overheating events, including feed line break - ATWS

Table 1.0-2 IRIS response to PWR Condition IV Events

Condition IV Design Basis Events		IRIS Design Characteristic	Results of IRIS Safety-by-Design
1	Large Break LOCA	Integral RV Layout – No loop piping	Eliminated by design
2	Steam Generator Tube Rupture	High design pressure once-through SGs, EHRS, piping, and isolation valves	Reduced consequences, simplified mitigation
3	Steam System Piping Failure	High design pressure SGs, piping, and isolation valves. SGs have small water inventory.	Reduced probability, reduced (limited containment effect, limited cooldown) or eliminated (no potential for return to power) consequences
4	Feedwater System Pipe Break	High design pressure SGs, piping, and isolation valves. Integral RV has large primary water heat capacity.	Reduced probability, reduced consequences
5	Reactor Coolant Pump Shaft Break	Spool pumps have no shaft	Eliminated by design
6	Reactor Coolant Pump Seizure	No DNB for failure of 1 out of 8 RCPs, even without Reactor Trip.	Reduced consequences
7	Spectrum of RCCA ejection accidents	With internal CRDMs there is no ejection driving force	Eliminated by design
8	Design Basis Fuel Handling Accidents	No IRIS specific design feature	No impact

radioactive fission products are directly prevented by the fuel cladding, the reactor pressure boundary, and the containment pressure boundary. For the fuel cladding boundary, the reactor protection system is designed to actuate a reactor trip to prevent exceeding the fuel design limits. The core design, together with defense-in-depth process and decay heat removal systems, provides this capability under expected conditions of normal operation, with appropriate margin for uncertainties and anticipated transient situations. The reactor coolant pressure boundary is designed with complete overpressure protection and high quality materials to provide and maintain the boundary during all modes of plant operation. The IRIS containment vessel, in conjunction with the defense-in-depth heat removal systems, is designed to: (1) not exceed its design pressure following postulated design basis accidents; (2) maintain a large margin to the design basis pressure during postulated design basis accidents to minimize leakage probability; and (3), prevent containment failure even under severe accident conditions.

Safety By Design. The key feature of the IRIS defense in depth is the previously mentioned safety by design summarized in Tables 1.0-1 and 1.0-2. An extensive discussion of the IRIS safety by design features is provided in Section 1.1.1.

Passive Safety-Related Systems. The next level of the defense-in-depth design strategy includes the IRIS safety-related passive systems and equipment. The safety-related passive systems are sufficient to automatically establish and maintain core cooling and ensure containment integrity following all postulated design basis events, assuming that the most limiting single failure occurs. These systems maintain core cooling and containment integrity after an event, without operator action and without onsite or offsite AC power sources, for at least seven days. The safety-related passive systems use only natural forces, such as gravity and natural circulation for their

continued operation. No pumps, fans, diesels, chillers, or other rotating machinery are required. A few simple valves align the passive safety systems when they are automatically actuated by the safety-related protection and safety monitoring system (PMS). To provide high reliability, these valves are designed to actuate to their safeguards positions upon loss of power, as well as, upon receipt of a safeguards actuation signal. However, the valves are also supported by multiple, reliable power sources to avoid unnecessary actuations. The PMS provides the safety-related functions of reactor trip, engineered safeguards features actuation, and post-accident monitoring. The IRIS design basis for the PMS is to provide an automatic response to any postulated accident, without requiring any operator action for extended periods of time.

Non-safety Systems. The next design level of defense-in-depth is the availability of certain non-safety systems for reducing the potential for events leading to core damage. For more probable events, these defense-in-depth, non-safety systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related systems. These non-safety-related systems establish and maintain safe shutdown conditions for the plant following design basis events, provided that at least one of the non-safety-related AC power sources is available.

Additionally, to minimize core damage probability, diverse, non-safety systems are provided to back up the main functions of the passive safety related systems. These systems are being defined on the basis of PRA considerations so to minimize the core damage and the fission product release probabilities. An example of this diversity is given in the residual heat removal function. The emergency heat removal system (EHRS) is the passive safety-related feature for removing decay heat during a transient. In case of multiple failures in the EHRS, defense-in-depth is provided by a simple, non-safety, passive containment cooling system (PCCS) and by the gravity driven injection from the pressure suppression system tanks and automatic depressurization (passive feed and bleed) functions. The introduction of these diverse features in the design is made possible in IRIS by the intrinsic characteristics of the integral reactor vessel and its coupling to the containment.

Containing Core Damage. IRIS is designed so that the reactor cavity floods following any severe accident event that may have the potential for core uncovery and melting. The objective of this cavity flooding action is to prevent reactor vessel failure and subsequent relocation of molten core debris into the containment. Retention of the debris in the vessel significantly reduces the uncertainty in the assessment of containment failure and fission products release to the environment due to ex-vessel severe accident phenomena.

1.1 Safety systems and features (inherent, passive and active)

The following sections present in some detail the IRIS features that provide protection against design basis events. First, the inherent design features that either eliminate, reduce the probability, or mitigate by design different design basis events are presented. These features are part of the IRIS safety-by-design approach introduced in Section 1.0, and are discussed in Section 1.1.1.

IRIS adopts passive safety systems to complete the protection against design basis events. The IRIS safety systems are discussed in Section 1.1.2.

The plant response to normal and abnormal operating conditions is normally performed by active systems. However, these systems are not required to perform any safety related function. For example, decay heat removal is normally performed by the startup feedwater system and by the steam dump system, while the decay heat removal safety grade function is provided by the passive emergency heat removal system.

The normally available active systems, coupled with the safety grade passive safety systems provide a diversity that lead to low release and core damage frequencies.

1.1.1 Safety by Design

The IRIS design has been primarily focused on establishing a design with innovative safety characteristics, and to achieve this goal a "safety by design" approach has been developed to eliminate or reduce the frequency and/or consequences of most serious accident sequences.

Several features of the design form the basis of the safety by design approach. These features were summarized in Table 1.0-1 and are discussed here in some more detail.

The adoption of an integral reactor coolant system eliminates the large loop piping required in other designs, and thus the potential for postulated large loss of coolant accidents is eliminated by design. The elimination of large break LOCAs is the most evident design feature of IRIS that provides an inherent elimination or mitigation of accident events, and other design features are presented here as they are a fundamental part of the IRIS defense in depth.

The adoption of an integral layout requires the design of a large vessel compared to other PWRs, with a long riser above the core to allow sufficient space for the placement of the steam generators and reactor coolant pumps in the pressure vessel. This provides a large coolant inventory in the reactor coolant system, that is a contributor to the IRIS response to small and medium break LOCAs, i.e., to rely on "maintaining water inventory" rather than "providing coolant injection". Also, the large coolant inventory provides a large heat sink that acts to effectively mitigate cooldown and heatup events.

The long riser, and the reduced pressure losses in the reactor coolant system, yield an effective natural circulation flow of coolant in the reactor coolant system to remove decay heat from the core. Finally, the tall riser provides sufficient space to accommodate internal control rod drive mechanisms (CRDMs). Not only will this allow to eliminate the potential for an RCCA (Rod Control Cluster Assembly) ejection, but will also allow to eliminate the CRDMs penetrations in the vessel upper head. Thus, the operational concerns associated with boron induced corrosion of the vessel head nozzles are eliminated by design. Internal CRDMs have been investigated worldwide and have been featured in a small district heating reactor in Beijing. However, they have never been adopted in a power production reactor. Because of concerns with time needed for their development, IRIS had initially opted to employ conventional CRDMs as reference and to implement the internal CRDMs when available. However, following recent operational issues due to boron induced corrosion of the CRDM nozzles in the upper head (e.g.,

Davis-Besse), coupled with the matured development of internal electromagnetic drives in Japan, internal CRDMs have been adopted for the reference IRIS design. The electromagnetic drives seem the preferred solution, but evaluation of all alternative designs is in progress.

Another IRIS specific feature that has been used to inherently mitigate the consequences of postulated events is the location of the steam generators inside the pressure vessel. Coupled with the large primary inventory, this is a fundamental feature to shape the IRIS response to postulated small and medium break LOCAs. The large heat transfer surface available on the steam generators inside the vessel is used to remove the heat produced in the core during the event, and provides a mean for depressurizing the reactor coolant system by condensing inside the vessel the steam produced, as opposed to loop PWRs which features a depressurization system that relies on venting reactor coolant mass to reduce pressure. Thus, coolant inventory is maintained. Also, the effective heat removal through the steam generators and the emergency heat removal systems (see Section 1.1.2) provides effective mitigation for all the events that require safety grade decay heat removal.

The adoption of an integral layout provides an overall reduction in the dimensions of the reactor coolant system, and thus allows to design a compact, higher-design-pressure containment system. During the initial phases of a loss of coolant accident, the pressure in the IRIS containment increases early in the accident, and reaches a higher allowable pressure. This higher back-pressure provides an inherent limitation to the inventory loss from the reactor coolant system. This feature is the third factor in the IRIS strategy of maintaining coolant inventory following a postulated loss of coolant accident. It should be noted that a large margin (almost 30%) to the containment design pressure is provided for all design basis accidents, and that the effective reactor coolant system and containment cooling provided by the EHRS rapidly reduces the pressure in the containment to minimize containment leakage following a postulated LOCA.

The IRIS once-through steam generators, with the primary coolant on the shell side provide a reduced volume of the secondary side, and this allows to design the IRIS steam system up to the isolation valves for full reactor coolant system design pressure. This in turn allows the elimination of the steam generator safety valves, since the steam system is protected by the reactor coolant system safety valves, prevents the reactor coolant system from overpressurizing the steam system, and reduces the probability for piping failures since the steam and feed lines are designed for full pressure. These features play an important role in the mitigation of both the probability and the consequences of postulated steam generator tubes ruptures. Not only the potential for failures is reduced since the tubes are mostly in compression (primary coolant on the shell side), but also failure propagation is highly improbable since the tube failure mode is a collapse. Additionally, an effective mitigation is provided simply by isolating the faulted steam generator.

Another feature of IRIS steam generators is the limited available water inventory: while this limits the consequences of cooldown events, this feature also limits the available inventory in the steam generators to mitigate heatup events. However, other IRIS design features, and in particular the large primary coolant inventory, more than compensate for this feature. Also, the rapid loss of mass from the steam generators provides a means for rapid detection of the fault and thus for a rapid actuation of the safety features.

An effective means for mitigating the consequences of heatup events is provided by another design characteristic of the integral layout. A large volume is available in the reactor vessel head for the pressurizer, which is thus designed with a large steam volume, to provide an inherent mitigation to events causing a pressurization of the reactor coolant system. This not only allows simplification of the design (IRIS does not feature a spray system nor automatic power-operated relief valves), but it also provides an inherent protection against reactor coolant system overpressurization.

The safety-by-design features of the reactor, coupled with the safety system concept presented in Section 1.1.2, provide an effective means of satisfying regulatory requirements for design basis events. The IRIS behavior during design basis events, coupling the inherent features of the design with the safety systems response, is discussed in more detail in Section 1.2.

1.1.2 Passive Safety Systems

The use of passive safety systems provides improvements in plant simplification, safety, reliability, and investment protection over conventional plant designs. IRIS follows the AP600/AP1000 approach and uses passive safety systems to improve the safety of the plant and to satisfy the regulatory safety criteria. The passive safety systems require no operator actions to mitigate design basis accidents.

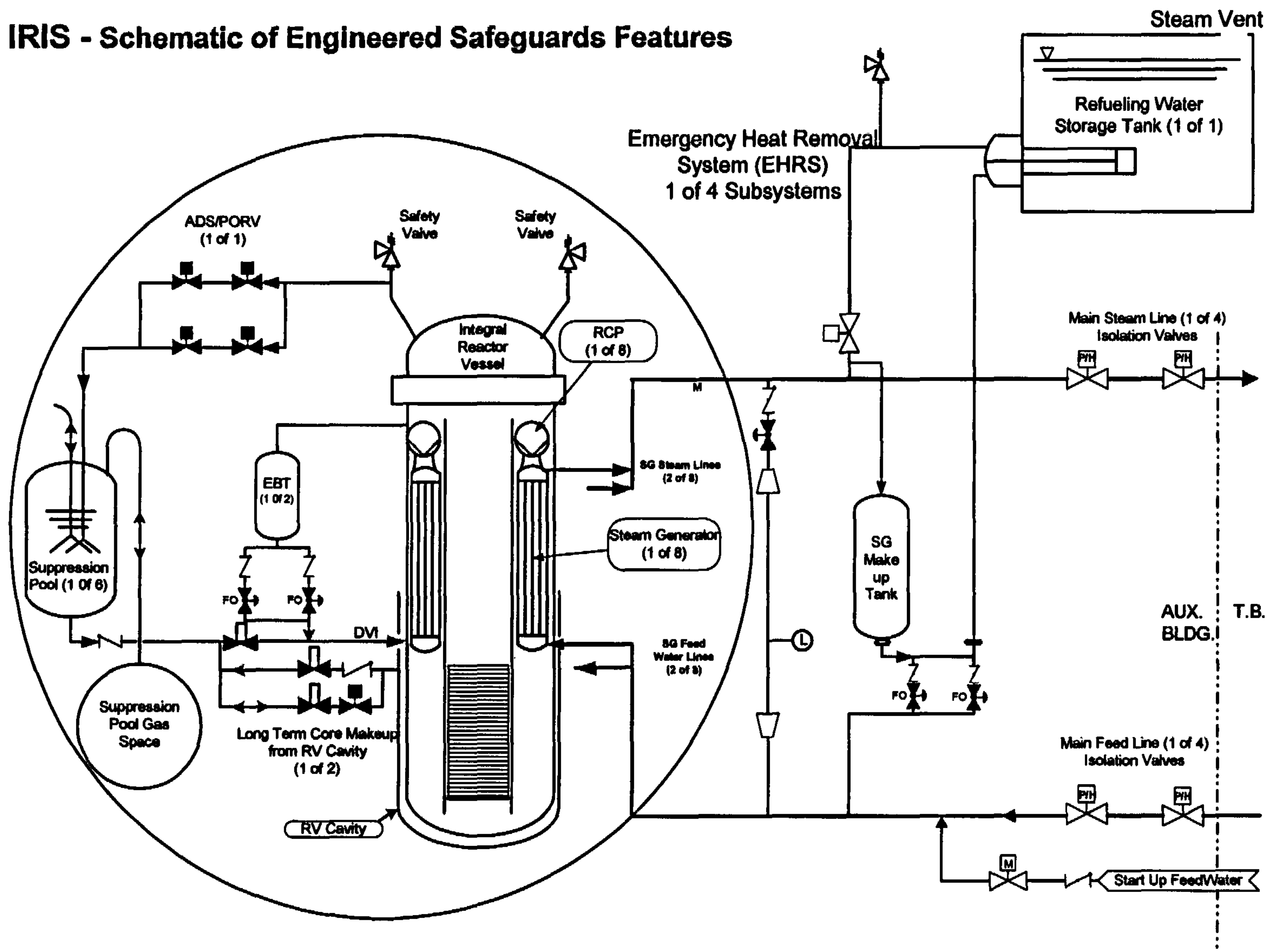
The IRIS passive systems design takes full advantage of the safety by design approach and the consequent elimination of some postulated design basis events (large LOCAs) and the inherent mitigation of several others (e.g. steam generator tube rupture, steam and feed line breaks, RCP locked rotor) through the definition of a safety strategy that is specifically tailored to respond to those remaining accident initiators, that are the more important contributors to core damage frequencies. This design approach allows the licensing safety criteria to be satisfied with a simplified plant design.

The passive systems are designed to meet the single-failure criteria, and probabilistic risk assessments (PRAs) are used to verify their reliability. The IRIS passive safety systems are even simpler than previous passive safety designs since they contain significantly fewer components, reducing the required tests, inspections, and maintenance, require no active support systems, and their readiness is easily monitored.

Section 4 of Reference 1 provides a detailed description of each one of the IRIS engineered safety features, while in the following sections a brief overview of how the different features interact in response to various events is provided. The IRIS passive systems conceptual configuration is presented in Figure 1.1.1-1, and includes:

- A passive emergency heat removal system (EHRS, Section 4.2 of Reference 1) is made of four independent trains, each including a horizontal, U-tube heat exchanger located in the refueling water storage tank (RWST) located outside the containment structure that is connected to one of the four separate SG feed/steam lines. The RWST provides the heat sink for the EHRS heat exchangers. The EHRS is sized so that a single train can provide decay heat removal in the case of a loss of secondary system heat removal capability. The EHRS operates by natural circulation removing heat from the primary system through the steam generators heat transfer surface.

Figure 1.1-1 Engineered Safety Features of IRIS



The steam produced in the steam generators is condensed in the EHRS heat exchanger, transferring the heat to the RWST water, and returning the condensate back to the SG. Following a LOCA where the loss of mass uncovers the SG tubes, the EHRS depressurizes the RV (depressurization without loss of mass) by condensing steam on the SG tubes. Thus the EHRS contributes to restoring the coolant inventory in IRIS because it condenses the steam produced by the core directly inside the reactor vessel, while transferring the decay heat to the environment. Also, in limiting the break flow by depressurizing the RV, the EHRS also limits containment pressurization and causes the containment pressure to decrease. Thus, the EHRS performs the functions of both core cooling and containment depressurization;

- A small automatic depressurization system (ADS, Section 4.3 of Reference 1) from the pressurizer steam space, assists the EHRS in depressurizing the reactor vessel when/if the reactor vessel coolant inventory drops below a specific setpoint. This ADS has one stage and consist of two parallel 4 inch lines, each with two normally closed valves. The single ADS line downstream of the closed valves discharges into the pressure suppression system pool tanks through a sparger. This ADS function ensures that the reactor vessel and containment pressures are equalized in a timely manner limiting the loss of coolant and thus preventing core uncovery following postulated LOCAs;

- Two compact (450 ft^3) full-system pressure emergency boration tanks (EBTs, Section 4.4 of Reference 1) which can deliver borated water to the RV through the direct vessel injection (DVI) lines for transient events. By their operation these tanks also provide a limited source of gravity-feed makeup water to the primary system;

- A containment pressure suppression system (CPSS, Section 4.6 of Reference 1) which consists of 6 water tanks and a common tank for non-condensable gas storage. Each suppression water tank is connected to the containment atmosphere through a vent pipe connected to a submerged sparger to condense steam released in the containment following a loss of coolant or steam/feed line break accident. The suppression system limits the peak containment pressure following a blowdown event to less than the containment design pressure. The suppression system water tanks also provide an elevated source of water that is available for gravity injection into the reactor vessel through the DVI lines in the event of a LOCA;

- A specially constructed lower containment volume that collects the liquid break flow, as well as any condensate from the containment, in a cavity where the reactor vessel is located. During a LOCA, the cavity floods above the core level, creating a gravity head of water sufficient to provide gravity driven coolant makeup to the reactor vessel through the DVI lines. The IRIS Long Term Gravity Makeup System (LGMS, Section 4.5 of Reference 1) also provides a path for gravity injection to the coolant system from the CPSS.

As in the AP600/AP1000, the IRIS safety system designs use natural gravitational forces instead of active components such as pumps, fan coolers or sprays and their supporting systems.

The safety strategy of IRIS provides a diverse means of core shutdown in the event that normally available active systems are not available by makeup of borated water from the EBT(s) and by core cooling and heat removal to the environment through the EHRS.

As discussed in Section 1.1.1, in the event of a significant loss of primary-side water inventory, the defense for IRIS is provided by the large coolant inventory in the reactor vessel and the fact that in IRIS the RV depressurization is attained with a limited loss of mass, thus maintaining a sufficient inventory in the primary system so that the core will remain covered for all postulated LOCAs. The EBT is capable of providing some water makeup to the primary system, but the IRIS strategy relies on "maintaining" coolant inventory, rather than "injecting" makeup water. This strategy is sufficient to ensure that the core remains covered with water for an extended period of time (days) even if no makeup is provided. Of course, when the reactor vessel is depressurized to near containment pressure, gravity flow from the pressure suppression system water tanks and from the containment will maintain the coolant inventory for an unlimited period of time. However, this function would not be strictly necessary for any reasonable recovery period since the core decay heat is removed directly by condensing steam inside the pressure vessel, thus minimizing the amount of primary water leaving the pressure vessel.

1.1.3 Active Safety Systems

No active system is required to perform safety-related functions for IRIS.

1.2 Anticipated Transients and Design Basis Accidents

A qualitative assessment of IRIS response to different events is discussed in the following paragraphs. This provides an overview of how the different safety features will interact in the response to various categories of events, and highlights the main phenomenological similarities and differences from other passive loop PWRs. A detailed discussion of various transients and postulated design basis accidents is provided in Section 2 of this report, and the purpose of this section is only to provide a qualitative overview of IRIS.

Note that not all the events discussed in Section 2 are considered in the following sections, since the focus is mostly on the qualitative and phenomenological differences between IRIS and other PWRs.

1.2.1 Loss of Coolant Accidents

The integral RV eliminates, by design, the possibility of large break LOCAs, since no large primary system piping is present in the reactor coolant system. Also, the probability and consequences of small break LOCA are lessened because of the drastic reduction in number of penetrations and in overall piping length, limiting the largest primary piping to a diameter of less than 4 inches and not allowing any vessel penetration lower than 2 meters above the top of the core. The innovative strategy developed to fully exploit the IRIS design characteristics in coping with a postulated small break LOCA is illustrated in Figure 1.2.2-1.

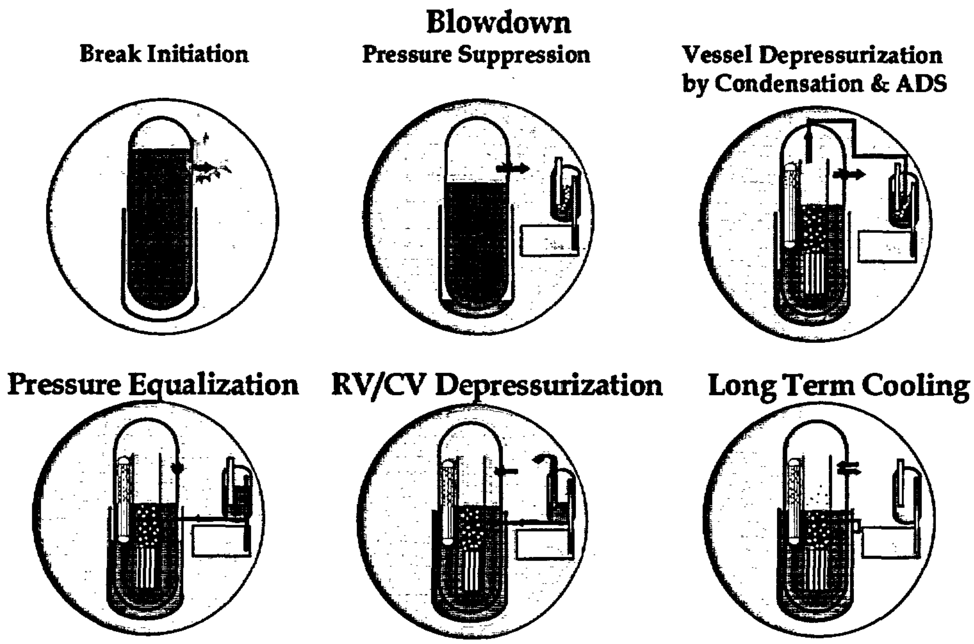


Figure 1.2-1 Overview of IRIS response to loss of coolant accident sequence

IRIS is designed to limit the loss of coolant from the vessel rather than relying on systems to inject water into the RV. This is accomplished by taking advantage of the following three safety-by-design features (Section 1.1.1):

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

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

At the end of the blowdown phase the RV and CV pressure become equal (Pressure Equalization) with a CV pressure peak of <0.9 MPa_g. The break flow stops and the gravity makeup of borated water from the suppression pool becomes available.

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

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

1.2.2 Steam Generator Tube Rupture

In IRIS, the steam generator tubes are in compression (the higher pressure primary fluid is outside the tubes) and the steam generators headers and tubes are designed for full external reactor pressure. With these safety by design features, tube rupture is much less probable and if it does occur, there is limited chance of tube failure propagation. Beside reducing the probability of the event occurrence, IRIS also provides, by design, a very effective means of mitigation for this event.

Since the steam generators, the EHRS, the feed and steam piping, and their isolation valves are designed for full reactor coolant system pressure, a tube rupture event is rapidly terminated by closure of the faulted SG main steam and feed water isolation valves upon detection of the failure. Once the isolation valves are closed, the primary water will simply fill and pressurize the faulted steam generator terminating the leak. Given the limited volume of the steam generators and piping, no makeup to the RV is even required; and since isolation of the faulted SG occur immediately upon detection, the release of radioactivity (primary fluid) to the environment will be minimized.

Compared to loop PWRs, the IRIS response to a tube rupture is such that steam generator overfill-overpressure-water relief/safety valve failure, resulting in an unisolable containment bypass scenario, is not possible. Also, the number of tubes assumed to fail has a limited effect on the system response (because the primary coolant can only fill the faulted steam generator system up to the isolation valves independent of the number of assumed tube failures) and does not impact the final plant state.

1.2.3 Increase in Heat Removal from the Primary Side

The limited water inventory in the once through steam generator has an important effect on the events in this category (Section 15.1 of the SRP). Increases in heat removal accidents due to increased steam flow are eliminated by design since the steam flow from the once-through steam generators can only temporarily exceed the feed water flow rate.

Also, the consequences of a design basis steam line break event are significantly lessened. Not only is the impact on the containment limited by the reduced discharge of mass/energy, but also no return to power due to the cooldown of the primary system is possible.

On the other hand, accidents resulting from feedwater malfunctions (increase in feedwater flow, decrease in feedwater temperature) tend to be relatively more important for IRIS. Due to the reduced water inventory in the Steam Generator System, these events have a more rapid and direct impact on the reactor coolant system. However, the large primary coolant inventory provides a large reactor coolant system thermal inertia, which acts to slow down the transient and limit the consequences of these events.

1.2.4 Decrease in Heat Removal from the Secondary Side

Events in this category (section 15.2 of the SRP, and including loss of offsite power, loss of normal feedwater, turbine trip, feed system piping failure) could potentially have larger consequences in IRIS than in loop type PWRs because of the limited water inventory in the once through steam generators. However, the IRIS design amply compensates for the limited heat sink provided by the steam generators with the large thermal inertia in the primary system (IRIS water inventory on a coolant-per-MWt basis is more than 5 times larger than advanced passive PWRs), and by the large steam volume in the IRIS pressurizer (steam volume-to-power ratio is also more than 5 times that of the AP1000, which in turn is significantly higher than other PWRs). The reactor trip setpoint is rapidly reached on a low feedwater signal, and the EHRS connected to the steam generators effectively removes sufficient heat to prevent any pressurizer overfill or high pressure relief from the reactor vessel to the containment.

1.2.5 Decrease in Reactor Coolant Flow Rate

The IRIS response to a complete loss of coolant flow is comparable to that of the AP600/AP1000, where the coastdown of the reactor coolant pumps is sufficient to maintain core cooling until the control rods are inserted and power is decreased.

On the other hand, for the design basis Locked Rotor event, the IRIS response is significantly improved over other PWRs by the increased number of reactor coolant pumps, which reduces the relative importance of the flow loss of a single pump. This design choice allows IRIS to prevent fuel damage (i.e. no departure from nucleate boiling) following a postulated locked rotor event even without a reactor trip. Finally, the design of the IRIS spool pumps eliminate the potential for a shaft seizure or shaft break.

1.2.6 Spectrum of Postulated Rod Ejection Accidents.

The integral reactor vessel has a large volume above the core that can be utilized to locate the control rod drive mechanisms (CRDMs) inside the vessel. This in-vessel CRDM location eliminates the rod ejection accident by design. Additionally, the operational failures associated with the large CRDM drive-line vessel head penetrations are also eliminated.

1.2.7 *Increase in reactor coolant inventory*

This category of events is all but eliminated in IRIS since IRIS does not utilize high pressure coolant injection following a LOCA. The inadvertent actuation of the small emergency boration tanks can be accommodated by the large pressurizer volume with no overpressure or overfill of the RV.

1.2.8 *Severe accidents (Beyond design basis accidents)*

IRIS is designed to provide in-vessel retention (IVR) of core debris by depressurizing and cooling the outside of the reactor vessel following severe accidents. With the reactor vessel intact and debris retained in the lower head, phenomena that may occur as a result of core debris being relocated to the reactor cavity are prevented. The IRIS containment design places the reactor vessel in a cavity that can always be flooded, and the reactor vessel has insulation that forms a natural circulation flow path for water to cool the outside vessel surface and prevent core melt through.

The design features of the containment ensure flooding of the vessel cavity region during accidents and submerging the lower portion of the reactor vessel in water. The liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The IRIS design also includes a provision for draining part of the pressure suppression system (PSS) water tanks water into the reactor cavity.

The IRIS design also includes a second means of containment cooling should cooling via the EHRS be postulated to fail. In this event, direct cooling of the containment outer surface is provided by the passive containment cooling system (PCCS) and containment pressurization is limited to less than its design pressure. These systems plus multiple means of providing gravity driven makeup to the core provides diverse means of preventing core uncovery and damage and ensuring containment integrity and heat removal to the environment.

The IRIS containment is inerted using nitrogen gas to prevent the possibility of hydrogen, generated by a damaged core, from igniting. This inerted containment atmosphere eliminates the need for hydrogen ignitors and recombiners to prevent a deflagration or explosion.

A discussion of IRIS response to Severe Accidents is not provided in this report and will be addressed in future reports.

1.3 REFERENCES

[1] Conway, L.E., et al., "IRIS Plant Description Document," WCAP-16062-P (Proprietary) and WCAP-16062-NP (Non Proprietary), March 21, 2003

2 ACCIDENT ANALYSES FOR IRIS

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2.0 Accident Analyses

This section provides an overview of plant conditions, ground rules and assumptions that are the basis for the analyses developed in sections 2.1 through 2.8.

2.0.1 Classification of Plant Conditions

The ANS/ANSI 18.2[1] classification divides plant conditions into four categories according to frequency of occurrence and radiological consequences. The four categories are as follows:

- Condition I: Normal operation and operational transients
- Condition II: Faults of moderate frequency
- Condition III: Infrequent faults
- Condition IV: Limiting faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk and those extreme situations having the potential for the greatest risk should be those least likely to occur.

This safety analysis study will be structured on the basis of the AP1000 Accident Analyses[2] and on the Standard Review Plan[3]. Acceptance Criteria for each Plant Condition are defined on the basis of NRC Standard Review Plan and of 10CFR50, Appendix A, General Design Criteria.

2.0.1.1 Condition I: normal operation and operational transients

Condition I occurrences are those events that are expected to occur frequently or regularly in the course of normal power operation, refueling maintenance or maneuvering of the plant. It is therefore required that Condition I occurrences are accommodated with margin between a plant parameter and the value of that parameter requiring either automatic or manual protective action.

Since Condition I events occur frequently, they must be considered from the point of view of their effect on the consequences of fault conditions (II, III, IV). For this reason, analysis of each fault condition should be based on a conservative set of initial conditions corresponding to the most adverse conditions that can occur during Condition I operation.

A typical list of Condition I events is:

(a) Steady State and Shutdown Operation

(b) Operation with Permissible Deviations:
(b.1) Operation with components or systems out of service
(b.2) Leakage from fuel with limited cladding defects
(b.3) Excessive radioactivity in the reactor coolant (fission products, corrosion products, tritium)
(b.4) Operation with steam generator tube leaks

(b.5) Testing

(c) Operational Transients
(c.1) Plant heatup and cooldown
(c.2) Step load changes (up to +/- 10%)
(c.3) Ramp load changes (up to +/- 5% per minute)
(c.4) Load rejection up to full load rejection

Analysis of Condition I events will be performed to evaluate the IRIS control system and operation programs: based on these evaluations, the list of typical Condition I events may be eventually expanded to allow efficient operation of the plant. Key IRIS design features that will have a positive effect on operational transients are the large pressurizer volume and the large thermal inertia of the primary system. Other IRIS features that will have an impact on the operational transients are the once through steam generators and the long residence time (defined as the time for the coolant to complete a full passage of the reactor coolant system) of IRIS which is over 40 seconds compared to about 10 for a typical PWR. Analysis of Condition I events will be addressed in a separate topical.

2.0.1.2 Condition II: faults of moderate frequency

These faults are not expected during normal plant operation but they can reasonably be expected to happen during plant life. At worst, they will result in a reactor trip with the plant capable of resuming operation. By definition, these faults or events do not propagate to a more serious condition and are not expected to result in a consequential loss of function of any barrier to the escape of radioactive products. The reactor core shall maintain the capability of reactivity control and to remove the generated heat, and the statistical failures of the fuel-cladding barrier shall be within the capability of the plant cleanup system.

Those events for which a significant difference exists in IRIS versus passive PWRs are boldfaced in Tables 2.0-1 to 2.0-3. All the events typically included in a PWR SAR are discussed here, including the ones eliminated by design in IRIS. For each event, the section in this report where they are addressed is indicated. The Condition II faults addressed in this study are listed in Table 2.0-1.

Table 2.0-1 : Condition II events analyzed in this study

	Acronym	Accident	Section
Increase in Heat Removal from the Primary system	FWM	**Feedwater system malfunction resulting in a decrease of feedwater temperature**	2.1.1
	FWM	**Feedwater system malfunction resulting in an increase of feedwater flow**	2.1.1
	ELI	**Excessive increase in secondary steam flow**	2.1.2
		Inadvertent operation of a steam generator relief valve	2.1.2
		Inadvertent operation of the passive emergency heat removal system	2.1.4

	Acronym	Accident	Section
Decrease in Heat Removal by the Secondary System	LOL	Loss of external electrical load	2.2.1
	TT	Turbine trip	2.2.1
		Inadvertent closure of main steam isolation valves	2.2.1
		Loss of condenser vacuum and other events that result in a turbine trip	2.2.1
	LOOP	Loss of a/c power to the station auxiliaries	2.2.2
	LONF	Loss of normal feedwater	2.2.2
Decrease in Reactor Coolant System Flow Rate	LOFA	Loss of flow accident – Partial loss of flow	2.3.1
Reactivity and Power Distribution Anomalies	RWFS	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	2.4.1
	RWAP	Uncontrolled RCCA bank withdrawal at power	2.4.2
		RCCA misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly	2.4.3
		Startup of an inactive reactor coolant pump at an incorrect temperature	2.4.4
		Chemical and Volume Control System malfunction that results in a decrease of the boron concentration in the reactor coolant	2.4.6
Increase in Reactor Coolant Inventory		Inadvertent operation of the passive core cooling system during power operation	2.5.1
		Chemical and volume control system malfunction that increases reactor coolant inventory	2.5.2
Decrease in Reactor Coolant Inventory		Inadvertent opening of a pressurizer relief valve	2.6.1
		Break in instrument line or other lines from the reactor coolant pressure boundary that penetrates containment	2.6.2

The design requirements that will be supported by IRIS safety analyses for Condition II events are as follows:

- Occurrences are accommodated with, at most, a reactor trip with the plant capable of returning to operation.
- Release of radioactive materials in effluent to unrestricted areas shall be in conformance with the Code of Federal Regulations (CFR) Section 10 CFR 20.
- These incidents shall not generate a more serious incident without other incidents occurring independently.
- There shall be no consequential loss of function of any barrier to the escape of radioactive products (no fuel rod failure or overpressurization):
  - Pressure in the reactor coolant and main steam systems should be maintained within 110% of the design value (GDC 15). IRIS Design pressure for both the RCS and the MSS is 2500 psia.
  - Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling (MDNBR) remains above the 95/95 DNBR limit for PWRs. (GDC 10, 17).
  - All Fuel rod damage and failure design criteria shall be satisfied
- (Single failure criterion) An accident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of potential fuel failures shall be determined.

2.0.1.3 Condition III: infrequent faults

Condition III events are faults that may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of reactivity is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with 10CFR100. By definition, a Condition III event alone does not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers.

Faults include in this category are listed in Table 2.0.2.

2.0.1.4 Condition IV: limiting faults

Condition IV events are faults that are not expected to take place, but are postulated because their consequences include the potential for release of significant amounts of radioactive material. These are the faults the plant must be designed against and represent limiting design cases. Condition IV events do not cause a fission product release resulting in doses in excess of the guideline values of 10 CFR 100. A single condition IV event does not cause a consequential loss of required functions of systems needed to cope with the fault.

Faults included in this category are listed in Table 2.0.3.

The effect of the IRIS innovative safety-by-design approach is evident especially in condition IV events: 7 out of 8 events are boldfaced, indicating significant differences with AP600/AP1000 and other PWRs. Of these 7 events three are completely eliminated (Large Break LOCA, RCCA Ejection and Shaft Seizure), while for the other four the consequences are eliminated or greatly mitigated by appropriate design choices (safety-by-design).

Table 2.0-2 : Condition III events analyzed in this study

		Accident	Section
Increase in Heat Removal from the Primary system	SLB	Steam system piping failure (minor)	2.1.3
Decrease in Reactor Coolant System Flow Rate	LOFA	Loss of flow accident – Complete loss of flow	2.3.2
Reactivity and Power Distribution Anomalies		RCCA misalignment (single RCCA withdrawal at full power)	2.4.3
		Inadvertent loading and operation of a fuel assembly in an improper position	2.4.7
Decrease in Reactor Coolant Inventory		Inadvertent operation of Automatic Depressurization System	2.6.1
	LOCA	Loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break)	2.6.5
Radioactive Release from a subsystem or component		Gas waste management system leak or failure	2.7.1
		Liquid waste management system leak or failure	2.7.2
		Release of radioactivity to the environment due to a liquid tank failure	2.7.3
		Spent fuel cask drop accidents	2.7.5

Table 2.0-3 : Condition IV events analyzed in this study

		Accident	Section
Increase in Heat Removal from the Primary system	SLB	Steam system piping failure (major)	2.1.3
Decrease in Heat Removal from the Primary system	FLB	Feedwater system pipe break	2.2.3
Decrease in Reactor Coolant System Flow Rate	LRSS	Reactor coolant pump locked rotor	2.3.3
		Reactor coolant pump shaft seizure	2.3.3
Reactivity and Power Distribution Anomalies		Spectrum of RCCA ejection accidents	2.4.8
Decrease in Reactor Coolant Inventory	SGTR	Steam generator tube rupture	2.6.3
	LOCA	Loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break)	2.6.5
Radioactive Release from a subsystem or component		Design basis fuel handling accident	2.7.4

2.0.2 Optimization of Control System

A control system setpoint study will be performed prior to plant operation to simulate performance of the primary plant control systems and overall plant performance. In this study, emphasis is placed on the development of the overall plant control system that automatically maintains conditions in the plant within the allowed operating window and with optimum control system response and stability over the entire range of anticipated plant operating conditions. The control system setpoints are developed using the nominal protection and safety monitoring systems implemented in the plant. Where appropriate (such as in margin to reactor trip analysis), instrumentation errors are considered and are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and plant control system setpoint study show that the plant can be operated and can meet both safety and operability requirements throughout the core life and for various levels of power operation.

The focus of this study is on the "safety requirements", while parallel activities are in progress to define an appropriate control system to satisfy "operability requirements". The plant control system setpoint study is comprised of analyses of the following control systems: rod control, turbine control, axial offset control, rapid power reduction, steam dump (turbine bypass), steam generator flow, pressurizer pressure, and pressurizer level. This study will be presented in a separate topical.

2.0.3 Plant Characteristics and Initial Conditions Assumed in Accident Analyses

2.0.3.1 Design Plant Conditions

Table 2.0.4 lists the principal plant parameters assumed in the analyses performed.

Given the still evolving nature of the project, data used for these analyses represent the most updated information available, but may be subject to changes as the design proceeds.

A conceptual layout of IRIS safety systems used in the analyses was shown in Figure 1.1.2-1 and for convenience is repeated in Figure 2.0.3-1.

2.0.3.2 Initial Conditions

For most accidents that are departure from nucleate boiling (DNB) limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the departure from nucleate boiling ratio (DNBR) design limit value as described in WCAP-11398-P-A[4]. This procedure is known as the Revised Thermal Design Procedure (RTDP).

Table 2.0-4 Nominal Values of Pertinent Plant Parameter Used in Accident Analysis

Thermal power output (MWt)	1002
Effective thermal power generated by reactor coolant pumps (MWt)	1.8
Core thermal power (MWt)	1000
Core inlet temperature (°F)	557.6
Vessel average temperature (°F)	590.4
Reactor coolant system pressure (psia)	2250
Reactor coolant flow per loop (gpm)	14.2 E+03
Steam flow from NSSS (lbm/hr)	3.99 E+06
Steam pressure at steam generator outlet (psia)	841
Steam temperature at steam generator outlet (°F)	602.6
Assumed feedwater temperature at steam generator inlet (°F)	440
Average core heat flux (Btu/hr-ft^2)	1.06 E+05

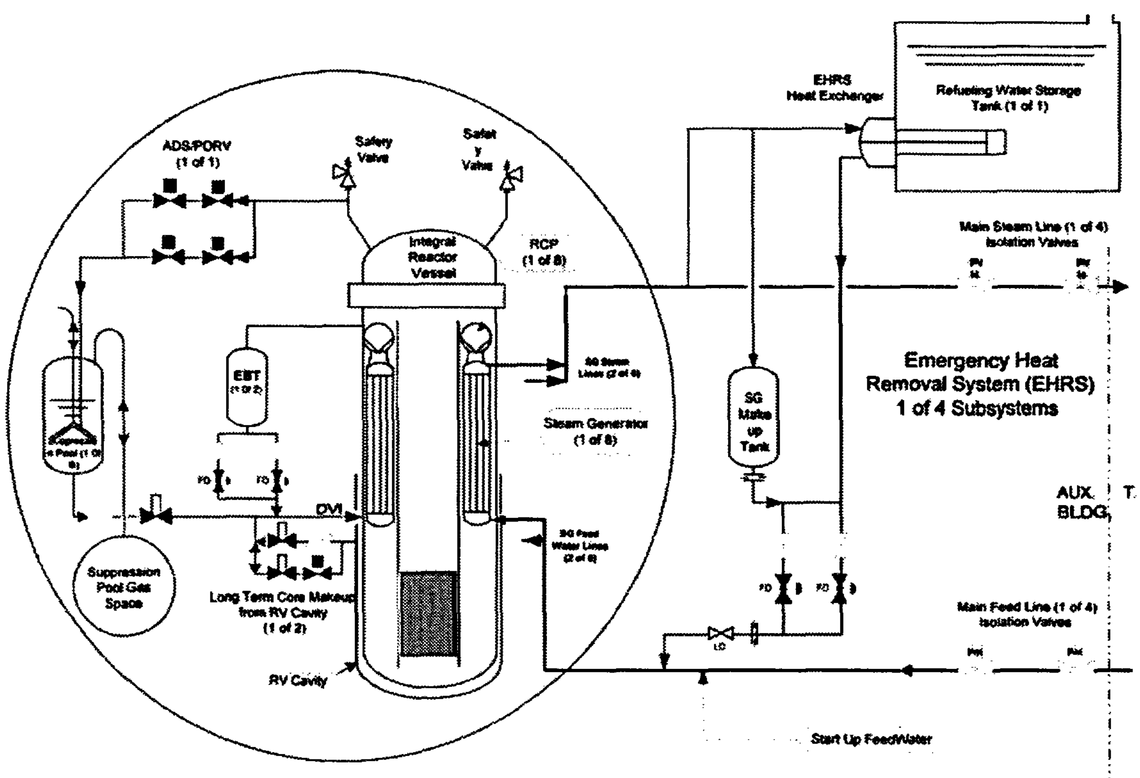

Figure 2.0-1 Schematic overview of IRIS Engineered Safety Features

For analyses for which the RTDP is not used, the initial conditions are obtained by adding the maximum steady state errors to rated values. The following conservative steady state errors are assumed in the analysis. Note that high uncertainties are assumed at this stage of the project to provide design margin: it is however expected that smaller errors will be supported and used in the final safety analyses.

- Core power — ± 2 percent allowance for calorimetric error
- Average reactor coolant system temperature — +6.5 to –7.0 °F allowance for controller deadband and measurement errors
- Pressurizer pressure — ±50 psi allowance for steady state fluctuations and measurement errors

Initial values for core power, average reactor coolant system temperature, and pressurizer pressure are selected to minimize the initial DNBR or to maximize the pressure peak during the transient, as discussed in the sections describing the specific accidents.

2.0.3.3 Power Distributions

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and the total peaking factor (F_q). Core design of the IRIS reactor is still in progress, and design limits for the power distributions have been defined only preliminarily.

For transients that are DNB limited, the radial peaking factor is important. The radial peaking factor increases with decreasing power level due to the allowed control rod insertion. This effect on $F_{\Delta H}$ is included in the core limits given in Figure 2.0.3-2. Transients that are DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications. A full power design limit $F_{\Delta H}$ of 1.65 has been assumed for these analyses.

The axial power shape used in DNB calculations is: (a) the 1.55 chopped cosine for transients analyzed at full power; or (b) the most limiting power shape calculated or allowed for accidents initiated at less than rated power or asymmetric RCCA conditions.

TBD

Figure 2.0-2 Core Limits for IRIS

2.0.4 Reactivity Coefficients Assumed In Accident Analysis

The transient response of the reactor system is dependent on reactivity feedback effects, in particular, the moderator temperature and the Doppler power coefficients. These coefficients are currently being evaluated.

In the analysis of certain events, conservatism requires the use of bounding reactivity coefficient values. The justification for use of conservative reactivity coefficients is given on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life, even though such combinations might not represent realistic situations.

Coefficients used in these analyses have been defined on the basis of preliminary IRIS analysis and by comparison to other Westinghouse designs, and are summarized in Table 2.0-5.

Table 2.0-5 Best Estimate and Safety Analysis coefficients for IRIS

Reactivity Coefficients	IRIS - Best Estimate	
	Least Negative	Most Negative
Doppler only Power Coefficient (pcm/% power)		
Upper Curve	*TBD*	*TBD*
Lower Curve	*TBD*	*TBD*
Doppler Temperature Coefficient (pcm/F)	*TBD*	*TBD*
Moderator Temperature Coefficient (pcm/F)	*TBD*	*TBD*
Moderator Density Coefficient (pcm / kg/m^3)	*TBD*	*TBD*
Boron Coefficient (pcm/ppm)	*TBD*	*TBD*
Reactivity Coefficients	**Safety Analyses – Design Limits**	
	Least Negative	**Most Negative**
Doppler only Power Coefficient (pcm/% power)		
Upper Curve	*-6.7*	*-10.2*
Lower Curve	*-12.6*	*-19.4*
Doppler Temperature Coefficient (pcm/F)	*-1.0*	*-3.5*
Moderator Temperature Coefficient (pcm/F)	*0.0*	*-40.0*
Moderator Density Coefficient (pcm / kg/m^3)	*0.0*	*0.43*
Boron Coefficient (pcm/ppm)	*-5.0*	*-13.5*

2.0.5 Rod Cluster Control Assembly Insertion Characteristic

Rod insertion characteristics for IRIS have not yet been defined. Based on fuel assembly design similarities (same active fuel length) the same insertion characteristic as provided for AP1000 has been assumed for now.

The negative reactivity insertion following a reactor trip depends on the acceleration of the RCCAs as a function of time and the variation of rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel. In analyses where all of the reactor coolant pumps are coasting down prior to, or simultaneous with, RCCA insertion, a time of 2.1 seconds is used for insertion to dashpot entry.

In analyses where some or all the reactor coolant pumps are running, the RCCA insertion time to dashpot is conservatively taken as 2.5 seconds.

The RCCA insertion characteristic (Figure 2.0.5-1) and the fraction of total reactivity insertion versus normalized rod position (Figure 2.0.5-2) are combined in Figure 2.0.5-3 to give a fraction of total negative reactivity insertion as a function of time.

The reactivity insertion characteristic is calculated for a core where the axial distribution is skewed to the lower region of the core. An axially skewed distribution to the bottom of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a rector trip, which is used as input to the point kinetics core models used in transient analyses. The bottom-skewed power distribution is not an input into the point kinetics core model.

There is inherent conservatism in the use of Figure 2.0.5-2 in that it is based on a skewed flux distribution, which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significantly more negative reactivity is inserted than shown in the curve, due to more favorable axial distribution existing prior to trip.

A conservatively low 4.0 percent Δk total reactivity insertion following a trip is assumed in transient analyses, except where specifically noted otherwise.

The normalized RCCA negative reactivity insertion versus time curve given in Figure 2.0.5-3 is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three-dimensional or axial one-dimensional core models, the reactivity insertion resulting from the reactor trip is calculated directly by the kinetics code and is not separable from the other feedback effects. In this case the RCCA position versus time in Figure 2.0.5-1 is used as code input.

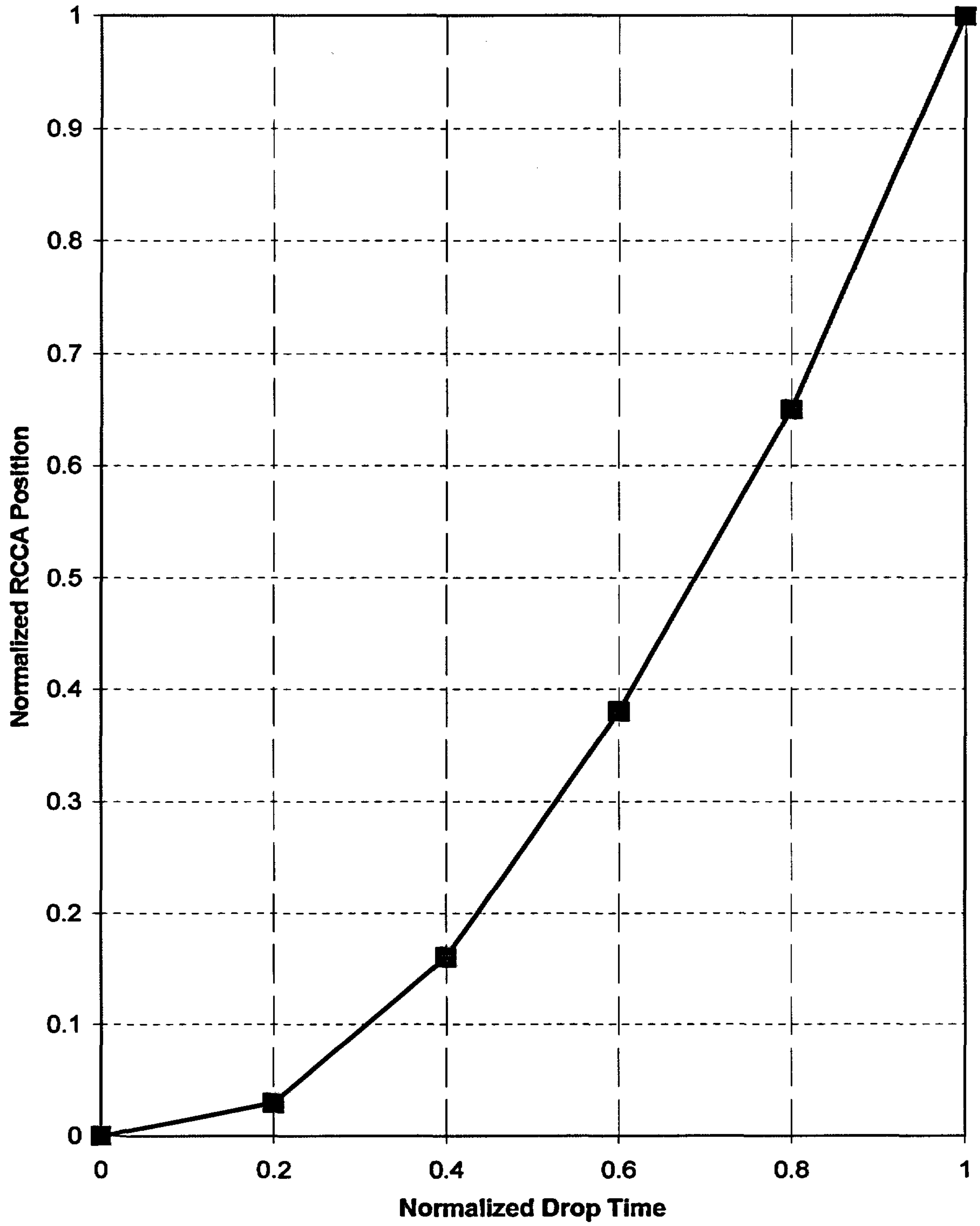

Figure 2.0-3 Normalized RCCA position versus time (normalized to 2.5 sec for the case with all or some pumps operating during the insertion period)

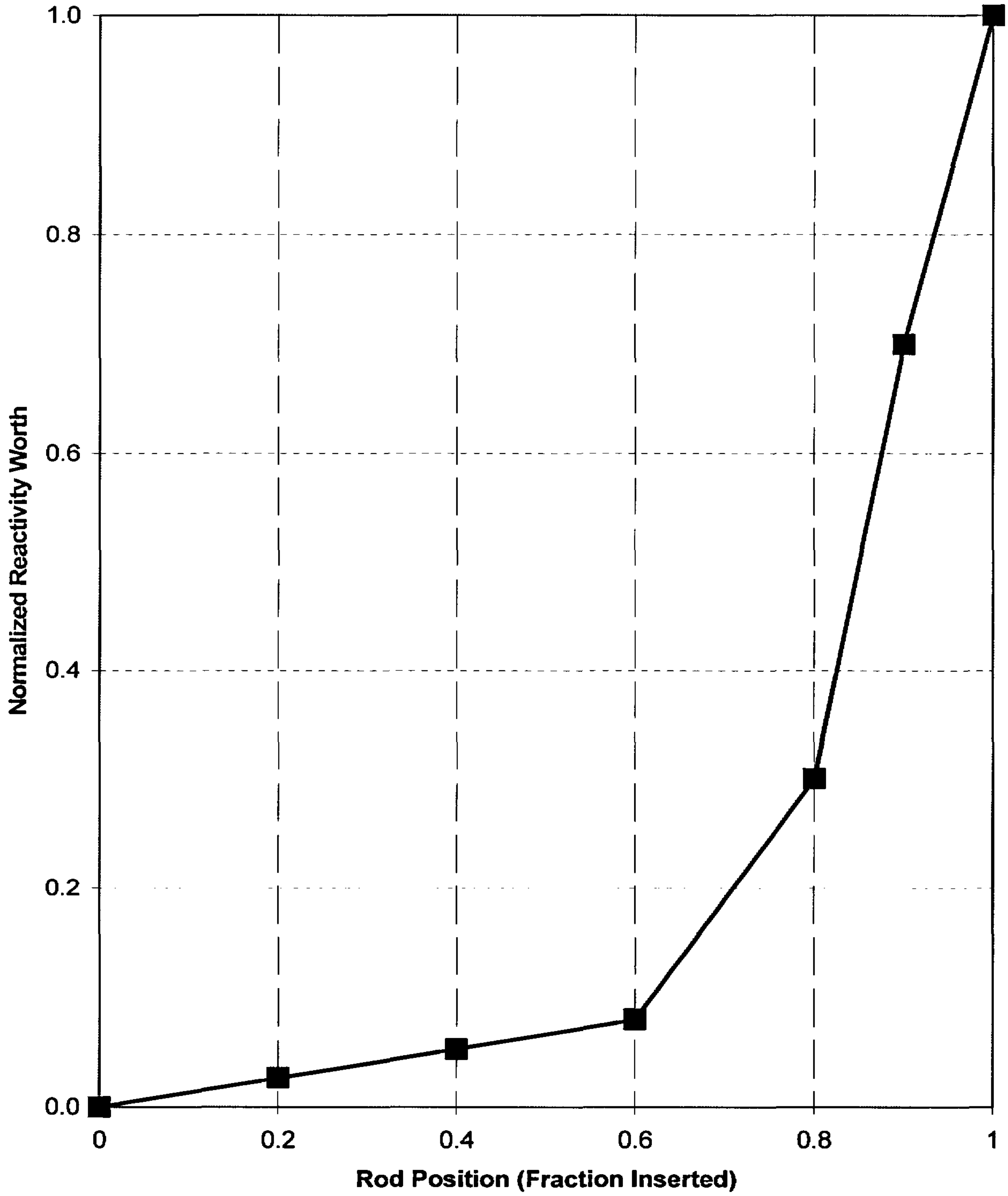

Figure 2.0-4 Normalized Negative Reactivity Insertion as a Function of RCCA position. Rod inserted to the dashpot.

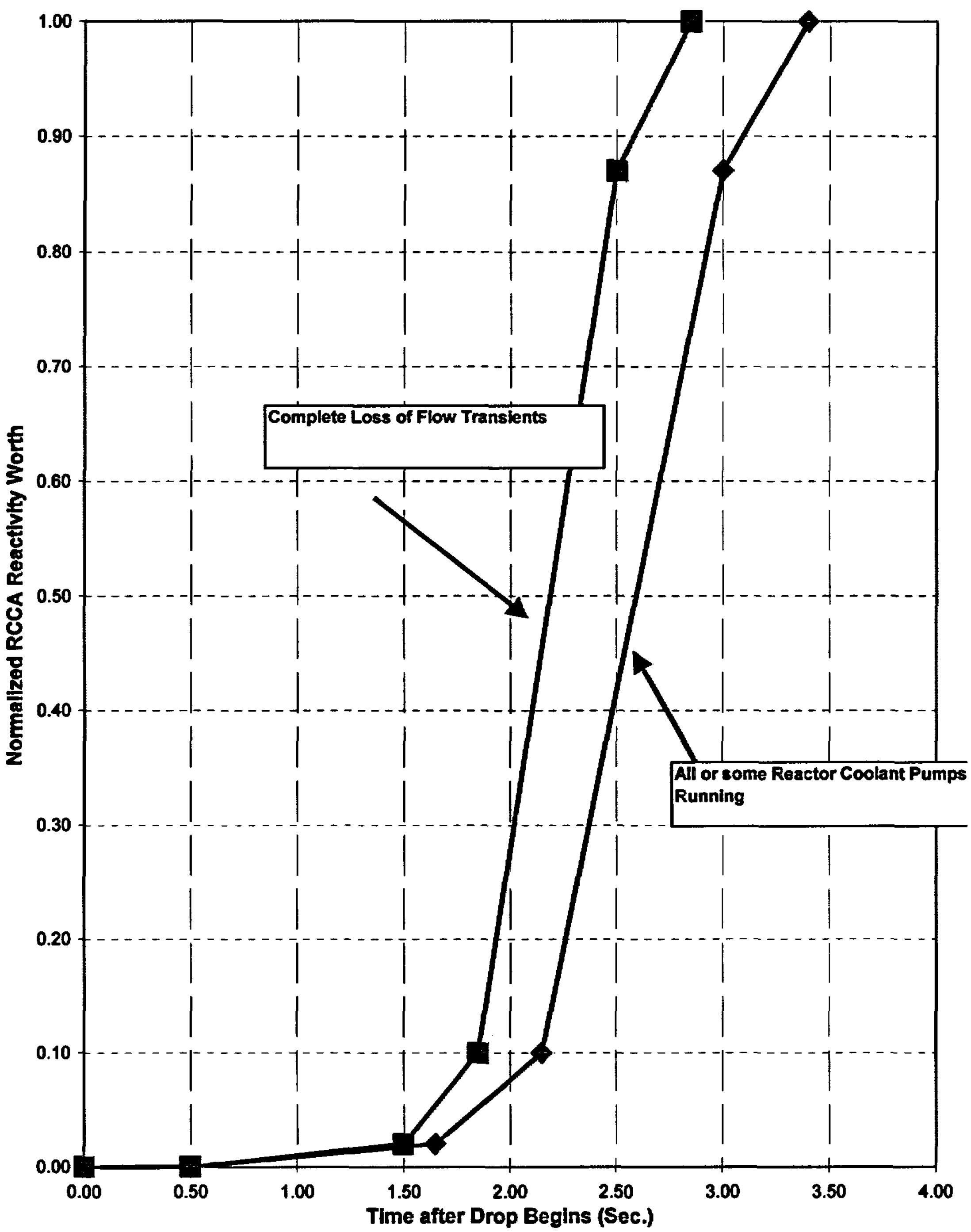

Figure 2.0-5 Normalized Reactivity Worth (as a fraction of total reactivity insertion, assumed as 4.0% ΔK) versus Drop Time.

C01

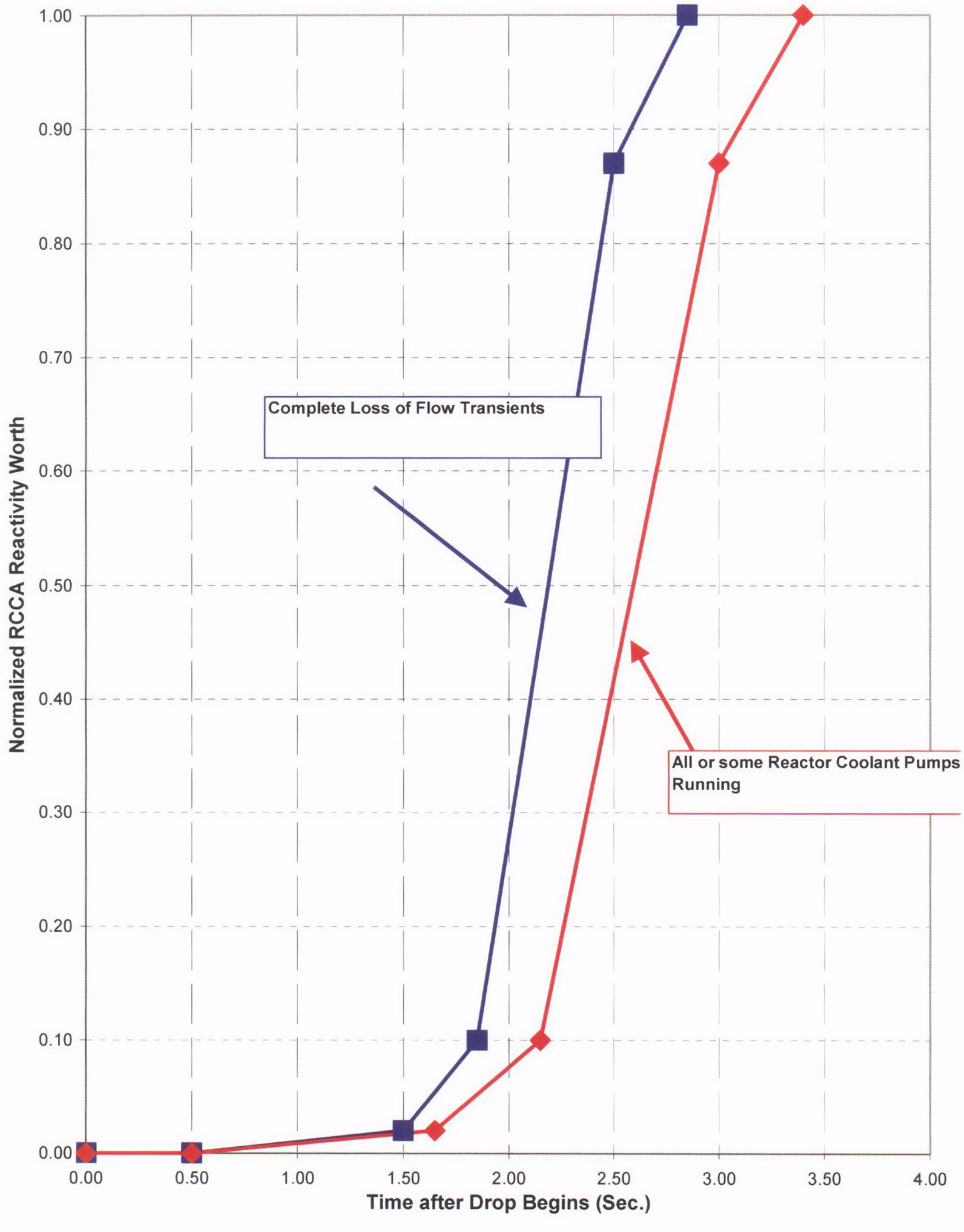

Figure 2.0-5 Normalized Reactivity Worth (as a fraction of total reactivity insertion, assumed as 4.0% ΔK) versus Drop Time.

C01

2.0.6 Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses

The safety analyses performed to date assume electromagnetic CRDMs. A reactor trip signal acts to open two trip breaker sets connected in series, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers and in the release of the rod by the mechanisms. The total delay to trip is defined as the time delay from the time the trip conditions are reached to the time the rods are free and begin to fall. Since the IRIS protection and monitoring system is still being finalized, a limited number of trip and actuation signals have been credited in this study, and conservative setpoints have been assumed in accident analyses. Setpoints and the time delay assumed for each trip function are summarized in Table 2.0-6. Reference is made in this table to the overtemperature and overpower ΔT trip; however, overtemperature and overpower ΔT for IRIS have not been completely defined. Moreover, the implementation of a digital protection system provides additional flexibility, and an alternative protection to the overtemperature and overpower ΔT may be implemented in IRIS. For safety analyses, simply the full power setpoints have been determined and will be used. Note that the provided list is not inclusive: only signals used in the analyses of this report are listed. Therefore, this list does not provide a complete overview of the IRIS protection and safety monitoring system.

Table 2.0-6 also summarizes the setpoints and the instrumentation delay for engineered safety features (ESF) functions used in accident analysis. Time delay associated with equipment actuated (such as valve stroke times) by ESF functions are summarized in Table 2.0-7 and 2.0-8.

The difference between the limiting setpoint assumed in these analyses and the nominal setpoint represents an allowance for instrumentation channel error and setpoint error. Also, conservative setpoint values have been used in these analyses to provide design flexibility and to compensate for design uncertainties: the final setpoint values are expected to be significantly less conservative. Nominal plant setpoints are specified in the Technical Specifications. During plant startup tests, it will be demonstrated that actual instrument time delays are equal or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times are determined periodically in accordance with Technical Specifications.

2.0.7 Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux

Instrumentation uncertainties and calorimetric uncertainties used in establishing the power range high neutron flux setpoint for IRIS are still being finalized, but is not expected that any significant difference from current PWRs will occur, except that digital, rather than analog, protection system will be used for IRIS. For now, typical PWRs setpoints for the neutron flux are assumed, but the digital reactor protection system should allow to realize significant improvements.

Table 2.0-6 Reactor Trip and Safeguards Actuation (S) Signals and Delays Used in Accident Analyses in this Report.

Description	Signal	Setpoint for Safety Analyses calculations	Trip Time Delays
Reactor Trip	Power Range High Neutron Flux, High Setting	Neutron Flux > 118% Nominal Full Power	0.9 s
	Power Range High Neutron Flux, Low Setting	Neutron Flux > 35% Nominal Full Power	0.9 s
	Overtemperature ΔT Trip	TBD (Section 2.0.6)	2.0 s
	Overpower ΔT Trip	TBD (Section 2.0.6)	2.0 s
	High Pressurizer Pressure	2414 psia (166.42 bar)	2.0 s
	Low Pressurizer Pressure	2000 psia (137.89 bar)	2.0 s
	High Pressurizer Level	water level > 11.25 ft (3.43m) height	2.0 s
	Low Coolant Flow	Reactor Flow < 87% Nominal	1.45 s
	Reactor Coolant Pump undervoltage	(delay from loss of voltage includes time to reach setpoint)	1.50 s
	Low Feedwater Flow	(< 75% of Reference Flow. Variable setpoint TBD)	4.0 s
S-Signals	S-Signal and Steamline Isolation on low Tcold	500 F (260 C)	2.0 s
	S-Signal and Steamline Isolation on Low Steamline Pressure	400 psia (27.57 bar)	2.0 s
	S-Signal on Low-Low Pressurizer Pressure	1800 psia (124.1 bar)	2.0 s
	S-Signal on High Containment Pressure	5 psig (0.34 barg)	2.0 s

EHRS Actuation, Partial	High Steam Line Pressure Low Steam Line Pressure Low Steam Generator Level	1305 psia (90 bar) 400 psia (27.57 bar) Not Assumed in the Analyses	2.0 s 2.0 s 2.0 s
EHRS Actuation, Full	High Containment Pressure, and Low-Low Pressurizer Pressure High-High Pressurizer water level	5 psig (0.34 barg) 1800 psia (124.1 bar) water level >11.25 ft (3.43m) height	2.0 s 2.0 s
RCP trip	Reactor Coolant Pump Trip Following an S-Signal Low Pressurizer Level Various Trips not credited in the analyses	**N.A.** Water level < 4.262 ft (1.299 m) height	(15.0 s) 2.0 s (15.0 s)
EBT (Any EBT actuation signal trips the reactor coolant pumps)	Low Reactor Coolant Inlet Temperature Low Steamline Pressure High Containment Pressure, and Low-Low Pressurizer Pressure High Pressurizer water level	500 F (260 C) 400 psia (27.57 bar) 5 psig (0.34 barg) 1800 psia (124.1 bar) water level >11.25 ft (3.43m) height	2.0 s 2.0s 2.0s 2.0 s
ADS	High Containment Pressure, and Low-Low Prz Pressure	5 psig (0.34 barg) 1800 psia (124.1 bar)	2.0 s
Other assumptions specific to some events are discussed in detail in the event analysis			

Table 2.0-7 Delays for the actuation of active components in safety analyses

Component	Delay
Steamline isolation valve closure	5 sec (maximum)
Feedwater isolation valve closure, feedwater control valve closure, or feedwater pump trip	5 sec (maximum)
EBT discharge valve opening time	15 sec (maximum)
Chemical and volume control system isolation	10 sec
Air Operated Valves opening time (EHRS)	2 sec
Pressurizer Safety and Relief Valves	2 sec
ADS valve opening time	≤ 30 sec

Table 2.0-8 EHRS automatic actuation sequence assumed in safety analyses to minimize inventory in EHRS.

Event	Time (seconds)
Signal Acquisition. MFIV assumed close.	0
5 seconds delay to allow for MFIV closure before signal to open MSIV. MSIV begins closing	5
MSIV assumed closed.	10
Air Operated Valves Open (EHRS), system operation initiated.	12

The calorimetric uncertainty is the uncertainty assumed in the determination of core thermal power as obtained from secondary plant measurements. The neutron flux instrument output is calibrated (set equal) to this measured power on a daily basis. The thermal power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators, steam pressure and temperature. Installed plant instrumentation is used for these measurements.

2.0.8 Plant systems and Components Available for Mitigation of Accident Effects

The plant is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features that minimize the probability and effects of fire and explosions.

An appropriate quality assurance program is implemented to provide confidence that the plant systems satisfactorily perform their assigned safety functions. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANS/ANSI N18.2-1973[1] is used. The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-1994 and Regulatory Guide 1.53 in the application of the single-failure criterion.

2.0.9 Fission Product Inventories

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the reactor core if the accident involves fuel damage. The radiological consequence analyses will use the conservative design basis fuel source terms that will be identified in Appendix 15A of the SAR or DCD.

2.0.10 Residual Decay Heat

A decay heat model similar to that used by Westinghouse for PWR safety analyses will be used in IRIS analyses. The Decay Heat curve assumed for the analyses is based on the ANS 1979 standard + 2σ for infinite operating time.

2.0.11 Computer Codes Used

The development of appropriate evaluation models and code selection for the system will be based on the evaluation model development and assessment procedure (EMDAP) outlined in reference [5], and starts with an identification and ranking of the physical phenomena that each evaluation model has to address. Phenomena are then ranked according to their importance for each accident sequence, and this information is collected in a "Phenomena Identification and Ranking Table (PIRT) and Scaling Analysis Report".

The PIRT and its assessment are then used to define appropriate assessment base and evaluation models for each event, and assist in the selection of the appropriate codes and modeling assumptions.

However, to assist in the development of a PIRT for IRIS, the safety assessment developed in this report is required, and thus a preliminary selection of codes has been necessary to perform these analyses. This selection however does not provide a final choice or a proposal of calculation tools for IRIS evaluation models.

Summaries of the two principal computer codes used in these analyses are provided below. Other codes — in particular, specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (see Section 2.6.5) — are summarized in their respective accident analyses sections.

2.0.11.1 RELAP 5 Computer Code

The RELAP 5 Mod3.3[6] code has been used to develop a model of the IRIS primary and secondary system to study the overall thermal-hydraulic plant behavior. The program simulates the neutron kinetics, reactor coolant system, steam generators and safety systems. The program computes pertinent plant variables including temperatures, pressures, and power level.

2.0.11.2 VIPRE-W Computer Code

VIPRE-W [7] is the Westinghouse version of the VIPRE-01[8], a three-dimensional subchannel code that has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core and hot channels. VIPRE-W modeling of a PWR core is based on a one-pass modeling approach [7]. In one-pass modeling, hot channels and their adjacent channels are modeled in detail, while the rest of the core is modeled on a relatively coarse mesh. The behavior of the hot assembly is determined by superimposing the power distribution upon the inlet flow distribution while allowing for flow mixing and flow distribution between flow channels. Local variations in fuel rod power, fuel rod and pellet fabrication, and turbulent mixing are also considered in determining conditions in the hot channels. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow and pressure drop.

The VIPRE-W core model as approved by the NRC [7] is used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core under all expected operating conditions and during transients. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the departure from nucleate boiling ratio (DNBR) design limit value as described in WCAP-11398-P-A[4]. This procedure is known as the Revised Thermal Design Procedure (RTDP).

2.0.12 Component Failures

2.0.12.1 Active Failures

SECY-77-439[9] provides a definition of active failures. An active failure results in the inability of a component to perform its intended function.

An active failure is defined differently for different components. For valves, an active failure is the failure of a component to mechanically complete the movement required to perform its function. This includes the failure of a manually or remotely operated valve to change position on demand. The spurious, unintended movement of the valve is also considered as an active failure.

Spring-loaded safety or relief valves that are designed for and operate under single-phase fluid conditions are not considered for active failures to close when pressure is reduced below the valve set point. However, when valves designed for single-phase flow are challenged with two-phase flow, such as a pressurizer safety valve, the failure to reseat is considered as an active failure.

For other active equipment - such as pumps, fans, and rotating mechanical components - an active failure is the failure of the component to start or to remain operating.

For electrical equipment, the loss of power, such as the loss of offsite power or the loss of a diesel generator, is considered as a single failure. In addition, the failure to generate an actuation signal, either for a single component actuation or for a system-level actuation, is also considered as an active failure.

Spurious actuation of an active component in safety-related passive systems is considered as an active failure. An exception is made for active components if specific design features or operating restrictions are provided that can preclude such failures (such as power lockout, confirmatory open signals, or continuous position alarms).

A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure. The error is limited to manipulation of safety-related equipment and does not include thought-process errors or similar errors that could potentially lead to common cause or multiple errors.

2.0.12.2 Passive Failures

SECY-77-439 also provides a description of passive failures. A passive failure is the structural failure of a static component that limits the effectiveness of the component in carrying out its design function. A passive failure is applied to fluid systems and consists of a breach in the fluid system boundary. Examples include cracking of pipes, sprung flanges, or valve packing leaks.

Passive failures are not assumed to occur until 24 hours after the start of the event. Consequential effects of a pipe leak—such as flooding, jet impingement, and failure of a valve with a packing leak — must be considered.

Where piping is significantly overdesigned or installed in a system where the pressure and temperature conditions are relatively low, passive leakage is not considered a credible failure mechanism. Line blockage is also not considered as a passive failure mechanism.

2.0.12.3 Limiting Single Failures

The most limiting single active failure (where one exists) of safety-related equipment is identified and discussed in each analysis description. The consequences of this failure

are described therein. In some instances, because of redundancy in protection equipment, no single failure that could adversely affect the consequences of the transient will be identified.

2.0.13 Operator Actions

The safety analyses presented in this report assume that no operator action is required to respond to any transient or accident condition. IRIS response is entirely based on the monitoring and protection system.

For events where the EHRS heat exchanger is actuated, the plant automatically cools down to the safe shutdown condition. When a stabilized condition is reached automatically following a reactor trip, it is expected that the operator may, following event recognition, take manual control and proceed with orderly shutdown of the reactor in accordance with the normal, abnormal, or emergency operating procedures. The exact actions taken, and the time at which these actions occur, depend on what systems are available and the plans for further plant operation.

However, for these events, operator actions are not required to maintain the plant in a safe and stable condition. Operator actions typical of normal operation are assumed for the inadvertent actuations of equipment to initiate a Condition II event.

2.0.14 Loss of Offsite AC Power

As required in GDC 17 of 10 CFR Part 50, Appendix A, anticipated operational occurrences and postulated accidents are analyzed assuming a loss of offsite AC power. The loss of offsite power is not considered as a single failure, and the analysis is performed without changing the event category. In the analyses, the loss of offsite AC power is considered to be a potential consequence of the event.

A loss of offsite AC power will be considered a consequence of an event due to disruption of the grid following a turbine trip during the event. Event analyses that do not result in a possible consequential disruption of offsite AC power do not assume offsite power is lost.

For those events where offsite AC power is lost, an appropriate time delay between turbine trip and the postulated loss of offsite AC power is assumed in the analyses. A time delay of 3 seconds is used. This time delay is based on the inherent stability of the offsite power grid as discussed in AP1000 DCD [2]. Following the time delay, the effect of the loss of offsite AC power on plant auxiliary equipment—such as reactor coolant pumps, main feedwater pumps, condenser, startup feedwater pumps, and RCCAs — is considered in the analyses. Turbine Trip (TT) occurs following a predefined delay after a reactor trip condition is reached. This delay is part of the IRIS reactor trip system.

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(a, c)

Figure 2.0-6 Timing of Loss of Offsite Power

The IRIS protection and safety monitoring system and passive safeguards systems are not dependent on offsite power or on any backup diesel generators. Following a loss of AC power, the protection and safety monitoring system and passive safeguards are able to perform the safety functions and there are no additional time delays for these functions to be completed.

2.0.15 References

[1] American National Standard Institute N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1973

[2] Westinghouse Electric Co., "AP1000 Design Control Document," APP-GW-GL-700, Revision 2, April 29, 2002

[3] NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981

[4] Friedland, A.J., and S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A (proprietary) and WCAP-11398-A (Non proprietary), April 1989

[5] Draft DG-1120, "Transient and Accident Analysis Methods", June, 2002 (available on NRC PERR at Accession Number ML003770849)

[6] NUREG/CR-5535-VI, "*RELAP5/MOD3 Code Manual*", Volumes I to V, Idaho National Engineering Laboratory, June 1995.

[7] Sung, Y.X., et al., "VIPRE-W Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A and WCAP-15306-NP-A (non proprietary), October 1999

[8] Stewart, C.W., et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Core," Volume 1-3 (Revision 3, August 1989), Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute.

[9] Case, E. G., "Single Failure Criterion," SECY-77-439, August 17, 1977.

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2.1 Increase in Heat Removal from the Primary System

A number of events that could results in an increase in heat removal from the reactor coolant system are postulated. These events are discussed in this section. Detailed analyses are presented for the more limiting events. A complete set of analyses have not been completed at this time.

Analyses will be presented in Section 15.1 of the IRIS Design Control Document for the following events that are identified as more limiting:

A. Feedwater system malfunctions that result in a decrease in feedwater temperature (Section 15.1.1)

B. Feedwater system malfunctions that result in an increase in feedwater flow (Section 15.1.1)

C. Excessive increase in secondary steam flow (15.1.3)

D. Inadvertent opening of a steam generator relief valve (15.1.4)

E. Steam system piping failure (15.1.5)

F. Inadvertent actuation of the EHRS heat exchanger (15.1.6)

In this document, specific consideration is provided for the following events:

(1) Excessive heat removal due to feedwater system malfunctions, Section 2.1.1

(2) Increase in secondary steam flow, Section 2.1.2

(3) Steam system piping failure, Section 2.1.3

(4) Inadvertent EHRS actuation, Section 2.1.4

The above items are considered to be ANS/ANSI Condition II events, with the exception of small steam system piping failures, which are considered to be Condition III, and large steam system piping failures, which are Condition IV events. Section 2.0.1 contains a discussion of ANS/ANSI classifications.

The events in this category present the potential for a reduction in the reactor coolant system temperature. In the presence of a negative moderator reactivity feedback condition, a decrease in the moderator temperature results in an increase in core power (nuclear flux). These transients are attenuated by the thermal capacity of the reactor coolant system. The overpower/overtemperature protection (neutron overpower, overtemperature and overpower ΔT) prevents a power increase that could lead to a departure from nucleate boiling ratio (DNBR) that is less than the design limit values. In addition, compared to loop-type PWRs, IRIS presents several features that have an impact on the system response to these events and that act to minimize the potential for an excessive reduction in reactor coolant system temperature. They are:

1. IRIS once-through steam generators and main steam system contain a limited water inventory. In loop-type PWRs with recirculation steam generators a large coolant inventory is available on the secondary side: this provides a large heat sink to

mitigate heatup events (see Section 2.2), but also creates the potential for events that increase the heat removal from the primary side. In IRIS, the limited inventory in the steam generators leads to a system for which the secondary system heat removal capability is strictly controlled by the feedwater flow: to increase the heat removal rate, the feedwater flow must be increased, and there is only a very limited potential for steam flow increase not accompanied by a feedwater flow increase. Therefore, following an isolation of the feedwater system, the mass contained in the steam generator does not provide a sufficient heat sink for a significant heat removal over an extended period of time. For example, following a major rupture in a steamline (discussed in detail in Section 2.1.3), the limited mismatch between feed and steam flow will limit the blowdown to the containment to the feedwater flow delivered to the faulted steam generator. Following feedwater isolation, the heat removal capability of the faulted steam generator will rapidly decrease, preventing any large decrease in reactor coolant system temperatures and limiting the mass and energy release to the containment.

2. Another design feature that affects the system response to these events is the large reactor coolant system volume: the IRIS heat sink is in fact located in the reactor coolant system rather than in the steam generator. The large inventory in the reactor coolant system mitigates the potential for reduction in the system temperature. Also, it provides a long grace period before any perturbation is transmitted in the reactor coolant system from the steam generators to the core, so that the system response (reactor trip) will actually occur before the potential for core power increase is realized.

3. The EHRS is designed such that, in case of a spurious actuation, there is no possibility of an increase in heat removed from the primary system, and thus no potential for a cooldown at power. While the EHRS is capable of removing almost 15% of full power at nominal RCS conditions, the design is such that the system can be actuated only following a reactor trip and feed and steam line isolation. If the valves of the EHRS fail open, the check valves in the EHRS line connecting each train to a separate feedline will prevent flow through the system.

The specific impact of these design features on different events is discussed in the following Sections.

2.1.1 Excessive Heat Removal due to Feedwater System Malfunctions

A change in steam generator feedwater conditions that result in an increase in feedwater flow or a decrease in feedwater temperature could result in excessive heat removal from the plant primary coolant system. In the presence of a negative moderator reactivity feedback condition, a decrease in the moderator temperature results in an increase in core power (nuclear flux). In automatic rod control, a primary system temperature decrease beyond the temperature deadband of the control system will cause the RCCAs to withdraw in an attempt to maintain the program Tavg. The resulting RCCA withdrawal also results in an increase in core power. These transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system. The overpower/overtemperature protection (neutron overpower, overtemperature and

overpower ΔT) prevents a power increase that could lead to a departure from nucleate boiling ratio (DNBR) that is less than the design limit values.

Two different events are analyzed as limiting feedwater system malfunctions: a reduction in feedwater temperature and an increase in feedwater flow. A reduction in feedwater temperature may be caused by a low-pressure heater train or a high-pressure heater train out of service or bypassed. An increase in feedwater flow may be caused by a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error.

These events are not analyzed in detail in this assessment since they are judged not to be significantly different or more severe than for current Westinghouse PWRs. As part of the IRIS design certification, it will be demonstrated that the overpower/overtemperature protection is adequate to preclude the occurrence of DNB and excessive moisture carryover. For failures that result in an increase in feedwater flow, there is also the possibility of steam generator overfill and resulting damage to the steam turbine due to excessive moisture carryover. An excessive addition of feedwater flow is prevented by the feedline high flow signal trip or by the low steam temperature signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps, and reactor.

While both from a phenomenological and protection point of view this event will not present significant differences from loop-type PWR experience, some IRIS specific consideration will apply to the analysis:

1. The reduced inventory in the IRIS steam generator system and the adoption of once-through steam generators, will lead to different system response: lacking the mitigation provided by the large inventory in the steam generators, the reactor coolant system will be rapidly affected by the feedwater malfunction (either a reduction in feedwater temperature or a increase in feedwater flow) and a decrease in reactor coolant system temperature will follow shortly after the initiation of the event. Mitigation will be provided by the large thermal inertia of the IRIS reactor coolant system (that will delay any significant reduction in reactor coolant system temperature) and by the large transmission time between the steam generators and the core. Due to the large coolant inventory, any perturbation of the coolant at the exit of the steam generator requires more than 20 seconds before it reaches the core inlet. This provides a large grace period that will simplify the design and requirements of the overpower/overtemperature protection system.

2. The once-through steam generators are not provided with a level monitoring at full power conditions. The high water level signal trip typical of loop-PWR is replaced in IRIS by a high feedwater flow trip signal or by a low steam temperature signal trip.

Given the similar transient evolution, this event will rely on methods of analysis that will not present significant differences from common PWR experience.

2.1.2 Increase in Secondary Steam Flow

In conventional loop PWR, an excessive increase in secondary steam flow results in a power mismatch between the reactor core power and the steam generator load demand.

The plant control system is typically designed to accommodate a 10-percent step load increase or a 5-percent-per-minute ramp load increase in the range of 25- to 100-percent full power. Any loading rate in excess of these values may cause a trip actuated by the protection and safety monitoring system.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the turbine bypass system or turbine speed control.

Due to the adoption of once through steam generators and the low inventory in the IRIS steam generators, the severity of this event is bounded by the feedwater system malfunction. This is due to the fact that only limited mismatch between steam and feed flow are possible: the only way to increase steam flow in IRIS is to increase the feedwater flow. Even following an equipment failure in the turbine bypass control, steam flow will not increase beyond the feedwater flow. The feedwater malfunction with an increase in feedwater flow for IRIS has a similar evolution to the excessive increase in secondary steam flow for conventional PWRs, as it leads to a sudden increase in heat removal from the secondary system. The quantification of this event requires however that a more detailed design of the IRIS control system be completed. No specific analysis for this event are therefore performed at this time for IRIS.

Similar considerations apply to the inadvertent opening of a single steam dump, relief or safety valve.

2.1.3 Steam System Piping Failure

The steam release arising from a rupture of a main steam line results in an initial increase in steam flow, which decreases during the accident as the pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core becomes critical and returns to power. A return to power following a steam line rupture with the most reactive RCCA assumed stuck in its fully withdrawn position is a potential problem mainly because of the resulting high-power peaking factors. The core is ultimately shut down by the boric acid solution delivered by the safety related systems.

The analysis of a main steam line rupture is performed to demonstrate that the following Standard Review Plant Section 15.1.5 criterion is satisfied: assuming the most reactive stuck RCCA with or without offsite power and assuming a single failure in the safety systems, the core cooling capability is maintained. DNB and possible cladding perforation following a Condition IV steamline break may be unacceptable. However, the steam line break analysis is typically performed to show that the DNB design basis is not exceeded for any steam line rupture.

While the physical phenomena that occur during the event evolution in IRIS are similar to those that occur during a PWR event, several design features of IRIS have an important effect on the system response, and provide an inherent mitigation of the event.

The main IRIS feature that impacts the steamline break response is the limited water inventory in the once-through steam generator system. In IRIS once-through steam generators the limited inventory limits the potential mismatch between steam and feed flow. Following a steam line break event at power, the cooldown would be limited by the amount of feedwater flow delivered to the steam generators. Only a limited cooldown of the reactor coolant system would occur, and thus the reactivity insertion is small and the core power increase is limited. The mass and energy release to the containment would lead to an increase in the containment pressure. The event would be terminated upon reaching either the high containment pressure or low steamline pressure setpoints, which would automatically initiate the steam line break mitigation sequence. Upon reaching either one of these two signals, steamline and feedline isolation would be initiated, terminating the delivery of feedwater to the steam generator system, by isolating main and startup feedwater, and the blowdown from the remaining steam generators, by isolating the steam lines. Given the limited inventory in the steam generators, the mass and energy release to the containment would end shortly after feedwater is isolated and therefore the high containment pressure signal provides adequate protection to IRIS containment.

In case of a steamline break event at zero power, the same considerations made above for the full power case remain valid for the containment: adequate protection will be provided by the high containment pressure signal that, by isolating feed and steam lines, will terminate the mass and energy release to the containment. For the core response and RCS transient analysis, no significant transient would occur following the release of mass from the faulted steam generator couple, unless a concurrent failure of the main feedwater control system is assumed. The total water inventory in a IRIS steam generators pair is less than 7,000 lbs for all operating conditions, which, on a per-MWt basis is between 1/6th and 1/4th of a loop type PWR. On the other hand, the amount of coolant in the RCS that provides the thermal inertia that reduces the cooldown rate is 4 to 5 times that of a loop type PWR on a per-MWt basis. Thus, a potential for a return to power only exists if the feedwater flow is assumed not to be isolated during the transient and continues to provide feedwater to the faulted steam generator pair.

Another feature of IRIS, common with the AP600/AP1000 design, is that the steam generators and feedwater system (main and startup feedwater) do not have a safety function in the removal of decay heat: this function is demanded to passive systems and can be performed effectively with a complete isolation of the main and startup feedwater system. This feature allows the protection system to provide a rapid steamline and feedline isolation following detection of a steam line break event, either on a high containment pressure, on a low steam generator pressure or on a low reactor coolant system cold temperature, thus terminating the feedwater flow to the faulted steam generator and consequently the cooldown of the RCS.

Analysis will be provided to demonstrate the IRIS containment and core response to steam line break events both at zero and hot full power, and an appropriate evaluation model for IRIS that accounts for these differences will be developed. For these preliminary assessment however, only these qualitative considerations are provided as appropriate evaluation models are still being developed. This is not only due to the confidence that, based on these considerations, IRIS will meet the criteria for steamline break analyses, but also to the fact that some design details regarding the control

system that might have an influence on the results are still being finalized, in particular the main and startup feedwater control/isolation system.

2.1.4 Inadvertent Operation of the EHRS

The inadvertent actuation of the EHRS could be caused by operator action or a false actuation signal. Either a single train or multiple train actuation might occur, up to and including a full actuation of all the four subsystems that compose this system.

The EHRS is designed such that no spurious actuation has the potential for an increase in the heat removal rate from the reactor coolant system, and thus for a potential insertion of positive reactivity that follows the reactor coolant system cooldown.

Each of the four trains of the EHRS consists of a heat exchanger, connected through a steam and feed-line to a pair of steam generators. Actuation of an EHRS involves opening of one of two parallel isolation valves, following an isolation of the main feed and steam lines. The EHRS heat exchangers are located in the refueling water storage tank (RWST), which provides the heat sink for the EHRS.

The response of the plant to an inadvertent EHRS actuation with the plant at no-load or at power conditions, leads to a complete isolation of the feed and steam lines, followed by actuation of one or more trains of the EHRS depending on the assumed failure. The EHRS actuation will trip the reactor, if not already tripped, and proceed to a plant cooldown to the safe shutdown conditions. Since the EHRS actuation follows a feed and steam line isolation, no potential for an increase in heat removal exists during power operation. If the failure assumes a spurious opening of the EHRS isolation valves without steam and feed isolation, check valves in the EHRS lines will prevent any flow through the EHRS heat exchangers and thus prevent any increase in heat removal through the steam generator. If the check valves are assumed to fail in the open position, the only consequence of the event would be that feedwater flow is partially diverted from the steam generators directly to the steam line. This will lead to a decrease in heat removal, not an increase. The check valves in the EHRS connecting lines to the feedline minimize the potential for this occurrence, and thus protect the turbine from excessive moisture carryover.

Only spurious actuation of all four trains of EHRS at no load conditions has the potential for a plant cooldown that may in principle lead to an excessive cooldown and a re-criticality , if the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip. This signal however also generates an actuation signal for the emergency boration system, that will deliver borated water to the reactor core to maintain the core subcritical. An analysis will be performed to verify that the sizing basis of the EHRS and of the EBS are adequate, and that any fault that leads to a EHRS actuation at power or at no-load conditions is inconsequential for IRIS.

2.2 Decrease in Heat Removal by the Secondary System

A number of transients and accidents have been postulated which could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system (RCS). These events are discussed in this section. Detailed analyses are presented for several such events that have been identified as bounding. Thus, a complete set of analyses has not been completed at this time.

Analyses will be presented in Section 15.2 of the IRIS Design Control Document for the following events that are identified as more limiting:

A. Steam pressure regulator malfunction or failure that results in decreasing steam flow (Section 15.2.1)

B. Loss of external load (15.2.2)

C. Turbine trip (15.2.3)

D. Inadvertent closure of main steam isolation valves (15.2.4)

E. Loss of condenser vacuum and other events resulting in turbine trip (15.2.5)

F. Loss of non-emergency AC power to the station auxiliaries (15.2.6)

G. Loss of normal feedwater flow (15.2.7)

H. Feedwater system pipe break (15.2.8)

In this document, specific consideration is provided for the following events, which cover five of the above eight events and bound the remaining:

(5) Loss of External Load and Turbine Trip, Section 2.2.1

(6) Loss of Normal Feedwater and Loss of non-emergency AC power, Section 2.2.2

(7) Feedwater system pipe break, Section 2.2.3

The above items are considered to be ANS/ANSI Condition II events, with the exception of a feedwater system pipe break, which is considered to be an ANS/ANSI Condition IV event. Section 2.0.1 contains a discussion of ANS/ANSI classifications.

The events in this category present the potential for a sudden reduction in the heat transfer rate in the steam generator, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge and RCS pressure rise. The pressurizer safety valves may open to prevent overpressurization of the reactor coolant system. These valves are sized to protect the Reactor Coolant System (RCS) against overpressurization. Also, assuming the loss of the normal heat sink, the EHRS is actuated to remove decay heat and bring the plant to a safe shutdown condition.

These events are collectively analyzed for the following reasons:

(1) To confirm that the pressurizer safety valves are adequately sized to prevent overpressurization of the RCS;

(2) To form the basis of the required ASME overpressure protection report;

(3) To ensure that the increase in RCS temperature does not result in Departure from Nucleate Boiling (DNB) in the core (on a 95% probability / 95% confidence limit basis). The Reactor Protection System is designed to automatically terminate any such transient before the DNBR falls below the applicable limit value;

(4) To verify the capability of the EHRS to remove decay heat.

Compared to loop-type PWRs, IRIS presents several features that have an impact on the system response to these events, and in particular:

(1) In conventional PWRs, the SG guarantees a large available water inventory and heat sink to remove decay heat before the actuation of engineered safety features become necessary. The IRIS once through SGs have only a limited secondary water inventory in the tubes and thus a very limited `intrinsic` capability of removing heat from the primary system when/if feedwater flow is not delivered to the SGs. Also, since steam flow is rapidly reduced following a loss of normal feedwater to the SGs, the turbine is rapidly tripped following any loss of feed flow events by closing the Fast Closure Turbine Stop Valves. This feature (i.e. rapid turbine trip) tends to further reduce the heat removal capability of the steam generators. Following a Turbine Trip, if the steam dump system is not available, the pressure in the steam system will start to increase, and the heat removed at the SGs will drop rapidly. Once the SG pressure reaches the setpoint for the EHRS actuation, the EHRS is actuated to remove decay heat.

(2) On the other hand, while the heat sink provided by the SGs is small for IRIS, the integral reactor coolant system provides a large heat sink. Heat-up events resulting from a loss of heat sink (i.e. loss of feed or steam flow) tend therefore to be mitigated by the large coolant inventory available on the primary side, that will reduce the rate of heat-up in the RCS, thus providing ample time for the actuation of the start-up feedwater system or, if it is unavailable, for the actuation of the EHRS.

(3) Another important feature of IRIS is the large pressurizer volume available in the upper head. The IRIS pressurizer is significantly larger than in other PWRs, with an overall volume of over 2,500 ft^3. The critical parameter is the steam volume to thermal power ratio, for which the difference with current PWRs is even more evident. Since the events in this category are typically analyzed to verify that the RCS pressure remains within the acceptable limits and to verify that no water relief occurs at the pressurizer safety valves, the increased size of the IRIS pressurizer guarantees additional margin and acts to mitigate the response to these events.

(4) The automatic steam dump system of IRIS, together with the Reactor Control System, is capable of accommodating a full load rejection without reactor trip. In safety analyses the steam dump valves and the start-up feedwater system are assumed to fail to actuate following a large loss of load, unless their actuation leads to a more severe transient evolution.

2.2.1 *Loss of External Load / Turbine Trip*

2.2.1.1 Identification of Causes and Accident Description

A loss of external electrical load may result from any of the following:

- Abnormal variation in the electrical network frequency or other adverse network operating conditions;
- Trip of the generator or opening of the main breaker from the generator with failure of turbine trip; in this case, the action of the turbine control system causes a large Nuclear Steam Supply System load reduction;
- Rapid closure of the turbine stop valves on loss of the hydraulic control fluid pressure actuated by one of a number of possible turbine trip signals;
- Spurious closure of the turbine stop or control valves or steamline isolation valves.

This anticipated transient is analyzed as a turbine trip from full power as this event is more severe than the total loss of external electrical load, loss of condenser vacuum or other events that result in a turbine trip since it results in a more rapid reduction in steam flow.

Upon initiation of the stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate turbine bypass. The automatic turbine bypass system accommodates up to 40% of rated steam flow. Reactor coolant temperature and pressure do not increase significantly if the turbine bypass system and pressurizer pressure control systems are functioning properly. If the condenser is not available, the excess steam generation is typically relieved to the atmosphere. Additionally, main feedwater flow is lost if the condenser is not available; feedwater flow is maintained by the startup feedwater system to provide adequate residual and decay heat removal capability.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant would be expected to trip from the protection and monitoring system if a safety limit is approached. A continued steam load of approximately 5 percent would exist after a total loss of external electrical load because of the steam demand of plant auxiliaries.

If a safety limit is approached, protection is provided by several different trips: low steam/feed flow, high pressurizer pressure, high pressurizer water level, and overtemperature ΔT trips would all be available to mitigate the consequences of the event.

If the steam dump valves fail to open following a large loss of load, the steam generator pressure and reactor coolant temperatures will increase rapidly. However, the pressurizer safety valves are sized to protect the reactor coolant system and steam generators against overpressure for all load losses without assuming the operation of the turbine bypass system, pressurizer spray, or automatic rod cluster control assembly control. The pressurizer safety valves can relieve sufficient steam to maintain the reactor coolant system pressure within 110 percent of the reactor coolant system design

pressure. The pressure in the steam generator system will rise until the setpoint for the actuation of the EHRS is reached. The actuation of the EHRS will provide adequate heat removal capability and rapidly reduce the pressure and temperatures in the reactor coolant system and in the steam generator system. Depending on the initial conditions, the EHRS may actuate before the safety valves setpoint is reached, preventing any release from the reactor coolant system to the containment.

The pressurizer safety valves capacity is sized to accommodate a complete loss of heat sink, with the plant initially operating at the maximum turbine load. Due to the mild transient evolution in IRIS compared to other PWRs given the large steam volume in the pressurizer, they are effectively capable of preventing any significant pressure increase beyond their opening setpoint for a complete loss of heat sink event.

2.2.1.2 Method of Analysis

To show the adequacy of the pressure relieving devices and also to demonstrate core protection margins, the behavior of the plant is evaluated in this analysis for a complete loss of steam load from full power, without credit for the rapid power reduction system. This assumption delays reactor trip until conditions in the reactor coolant system result in a trip due to other signals. Thus, the analysis assumes a bounding transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for startup feedwater to mitigate the consequences of the transient. The EHRS is available for long term recovery.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, analyses are performed to evaluate the effects produced by a possible consequential loss of offsite power during a complete loss of steam flow. As discussed in Section 2.0.4, the loss of offsite power is considered as a direct consequence of a turbine trip occurring while the plant is operating at power. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

The RELAP5 code (discussed in Section 2.0.11) has been used in this analysis. The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, and steam generators. The program computes pertinent plant variables including temperatures, pressures, and power level.

Typically, four turbine trip cases are analyzed:

A. Minimum reactivity feedback, with no RCS pressure control

B. Minimum reactivity feedback, with no RCS pressure control, with loss of offsite power

C. Minimum reactivity feedback, with RCS pressure control

D. Minimum reactivity feedback, with RCS pressure control, with loss of offsite power

Cases A and B are analyzed to calculate a conservative maximum RCS pressure. Cases C and D are performed to calculate a conservative minimum DNBR, and are analyzed using the revised thermal design procedure as discussed in Section 2.0.3.

Only cases A and B are analyzed in this report. The reason is that due to the very long circulation time in the IRIS reactor, especially in the downcomer region, the reactor trip is expected to occur before any temperature perturbation reaches the core. Therefore, it is expected that these events will lead to a mild DNB transient that will not be limiting. For the case with loss of offsite power, the event shall be bounded by the complete loss of flow event since in the former case the loss of forced flow will follow the reactor trip, while in the complete loss of flow the loss of forced flow precedes the reactor trip. Naturally, the DNB cases will be analyzed in a complete chapter 15 study of IRIS.

The major assumptions used in the analysis are summarized below.

- **Initial Operating Conditions.** For cases A and B above, initial conditions of core power, reactor coolant temperature, pressurizer pressure and pressurizer level are obtained by applying maximum errors to the nominal full-power values in the conservative direction.
- **Reactivity Coefficients.** Since the Loss of Load / Turbine Trip event results in a primary system heatup, it is conservative to model minimum moderator reactivity feedback conditions. Hence, minimum moderator reactivity feedback conditions are assumed for this event. A conservatively large absolute value of the Doppler-only power coefficient is used (see Table 2.0-5). This is equivalent to a total integrated Doppler Reactivity from 0- to 100-percent of 0.016 Δk.
- **Reactor Control.** From the standpoint of both the maximum pressures attained and DNBR, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.
- **Steam Relief.** No credit is taken for operation of the steam dump system. The steam generator pressure rises during the transient until the EHRS is actuated.
- **Pressurizer Spray.** The current IRIS design does not feature sprays.
- **Feedwater Flow.** Feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for start-up feedwater flow. During the transient the steam generator pressure will start increasing as the SGs are coupled to the reactor coolant system. Since the SG and EHRS are designed for full reactor coolant system design pressure, the steam generator system overpressure protection is provided by the pressurizer safety valves. The actuation of the EHRS will terminate the event and the pressure increase in the steam generator system, and proceed to cool down the plant.
- **Reactor Trip.** Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the rapid power reduction on the turbine trip. Trip signals are expected due to low steam/feed flow, high pressurizer pressure, overtemperature ΔT, RCP undervoltage and high pressurizer water level. Although a trip signal will be rapidly generated on a low steam/feed trip that will lead to a very mild evolution of the transient, the analysis does not credit the low steam/feed flow and delays the trip until another signal, typically the high pressurizer pressure, is generated. This is not required since the low steam/feed is a valid reactor protection signal for IRIS, but nonetheless this assumption is provided to demonstrate the capability of the high pressurizer pressure trip coupled with relief from the pressurizer

safety valves in preventing any reactor coolant system overpressurization until the EHRS is actuated.

- **Safety Systems and Single Failure.** The protection and safety monitoring system may be required to function following a turbine trip. Pressurizer safety valves may be required to open to maintain system pressures below acceptable limits. No single failure prevents operation of systems required to function. The EHRS actuation may be required to terminate the steam relief at the pressurizer safety valves and to cool-down the reactor coolant system and the steam generator system.
- **Availability of Offsite Power.** This case is analyzed with and without offsite power available. As discussed in Section 2.0.4, the loss of offsite power is considered to be a consequence of an event due to disruption of the electrical grid following a turbine trip during the event. The grid is assumed to remain stable for 3 seconds after the turbine trip. In the analysis of this event, the turbine trip is the initiating event and therefore offsite power is assumed to be lost 3 seconds into the transient. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down. The case without offsite power available may result in an early trip on low pump speed, which protects the core thermal margin, and leads to a milder pressurization than in the case with offsite power available.

2.2.1.3 Results

As mentioned in Section 2.2.1, no DNB case analysis is presented here.

The calculated sequence of events for the accident is shown in Table 2.2-1.

Figures 2.2.1-1 through 2.2.1-7 show the transient responses for the turbine trip with offsite power available. No credit was taken for the rapid power reduction system, for the turbine bypass system and for the low feed/steam flow trip. The reactor is tripped on the high pressurizer pressure signal. The neutron flux remains essentially constant at full power until the reactor is tripped (Figure 2.2.1-1). The RCS pressure is shown in Figure 2.2.1-2, the pressure transient is mitigated by the large steam volume in the pressurizer, and only a modest pressurization rate is experienced in the transient. The safety valves actuation occurs later than in conventional PWRs, and is capable of rapidly stabilizing the pressure at the safety valves setpoint. As the pressure in the steam generator system rises, as shown in Figure 2.2.1-3, the EHRS setpoint is reached and the passive system is actuated. The EHRS is capable of removing decay heat and provides sufficient cooling to reduce the RCS pressure and proceed to a plant cooldown.

Figures 2.2.1-8 to 2.2.1-14 show the transient responses for the turbine trip without offsite power available. In this case, a reactor trip is rapidly reached on a pump undervoltage: this reduces the energy released to the coolant system during the transient and leads to a milder pressurization rate. The net effect is that the EHRS is actuated before the setpoint for the pressurizer safety valves is reached, and provides sufficient cooldown capability to prevent any further pressurization and to prevent actuation of the safety valves.

2.2.1.4 Conclusions

Results of the analyses and qualitative considerations show that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the RCS are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is expected to be maintained by operation of the reactor protection system; i.e., the DNBR will be maintained above the limit value.

The above analysis preliminarily demonstrates the ability of the NSSS to safely withstand a full load rejection.

2.2.2 Loss of AC Power to the Plant Auxiliaries and Loss of Normal Feedwater

2.2.2.1 Identification of Causes and Accident Description

The loss of offsite power to the plant auxiliaries is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The onsite standby AC power system remains available but is not credited to mitigate the accident.

From the decay heat removal point of view, in the long term this transient is more severe than the turbine trip event analyzed in Section 2.2.1 because, for this case, the decrease in heat removal by the secondary system is accompanied by a reactor coolant flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core.

The reactor will trip upon reaching one of the setpoints in the primary or secondary systems as a result of the flow coastdown and decrease in secondary heat removal or due to the loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system (which is started automatically when low level occurs in any steam generators pair or when normal feedwater is lost) and by the turbine bypass system. If either is unavailable, emergency core decay heat removal is provided by the EHRS. The EHRS is a passive system that connects a U-tube heat exchanger located in the RWST with the feed and steam lines of each steam generator pair. The RWST provides the heat sink for the heat exchanger.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant system and in the EHRS loop.

In summary, following a loss of AC power with turbine and reactor trips, the sequence described below occurs:

- The loss of AC power leads to a loss of normal feedwater and a loss of forced reactor coolant system flow. As a consequence of a complete loss of normal

feedwater, feed flow to the steam generators is rapidly terminated. The rapid decrease in feedwater flow leads to a reactor and turbine trip on a low steam/feed flow signal. The same signal also actuates the startup feedwater system.

- Plant Vital instruments are supplied from the Class 1E and uninterruptable power supply. The onsite standby power system, if available, supplies AC power to the selected plant loads.

- As the steam system pressure rises following the turbine trip, the condenser is assumed not to be available for turbine bypass. As the no-load temperature is approached, the steam dump, if available, is used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition if startup feedwater is available to supply water to the steam generators.

- If steam dump is also not available, the pressure in the main steam system will rise until the setpoint for the EHRS actuation is reached. EHRS actuation will isolate the steam generators by closing the main steam and feed isolation valves and will provide decay heat removal in natural circulation through the heat exchanger located in the RWST.

- If startup feedwater is not available, the EHRS is actuated.

A loss of AC power to the plant auxiliaries is a condition II event, a fault of moderate frequency.

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of AC power sources) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If startup feedwater is not available, the safety-related EHRS heat exchanger is automatically aligned by the protection and safety monitoring system to remove decay heat.

The sequence of events following a loss of normal feedwater is very similar to the loss of offsite power sequence, with the main difference given by the fact that offsite power remains available throughout the event.

A loss of normal feedwater is also classified as a condition II event, a fault of moderate frequency.

Due to the similar nature of the two events, and to the fact that both are analyzed to verify the EHRS decay heat removal capability, both are discussed together in this section, and a common method of analysis is used.

2.2.2.2 Method of Analysis

The analysis of the loss of normal feedwater and loss of AC power events is performed to demonstrate the adequacy of the protection and safety monitoring system, the EHRS, and the reactor coolant system natural circulation capability in removing long-term decay heat. This analysis also demonstrates the adequacy of these systems in preventing excessive heatup of the reactor coolant system with possible overpressurization or loss of reactor coolant system water.

The RELAP5 code (discussed in Section 2.0.11) has been used in this analysis. The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, and steam generators. The program computes pertinent plant variables including temperatures, pressures, and power level.

The assumptions used in this analysis minimize the energy removal capability of the EHRS and maximize reactor coolant expansion. Note that the transient response of the plant following a loss of AC power to plant auxiliaries is similar to the loss of normal feedwater flow accident, except that power is assumed to be lost to the reactor coolant pumps at the time of the reactor trip. For the loss of normal feedwater transient, the reactor coolant volumetric flow remains at its normal value following the reactor trip. The reactor coolant pumps continue to run until automatically tripped when the low pressurizer level reactor coolant pumps trip setpoint is reached.

Other major assumptions used in the analysis are summarized below.

- **Initial Operating Conditions.** Initial conditions of core power, reactor coolant temperature, and pressurizer pressure are obtained by applying maximum errors to the nominal full-power values in the conservative direction. For this preliminary analysis, qualitative considerations based on the comparison with other PWRs and on IRIS specific design features have been used to identify the most conservative scenario. A complete quantitative assessment shall be performed as part of the design certification analyses.
- **Reactivity Coefficients.** Since the loss of normal feedwater and loss of AC power events result in a primary system heatup, it is conservative to model minimum moderator reactivity feedback conditions. A conservatively large absolute value of the Doppler-only power coefficient is used (see Table 2.0-5). This is equivalent to a total integrated Doppler Reactivity from 0- to 100-percent of 0.016 Δk.
- **Reactor Control.** It is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.
- **Steam Relief.** No credit is taken for operation of the steam dump system. The steam generator pressure rises during the transients until the EHRS is actuated.
- **Pressurizer Spray.** The current IRIS design does not feature sprays.
- **Feedwater Flow.** Main feedwater flow is assumed to be lost at the beginning of the transient. No credit is taken for start-up feedwater flow. During the transient the steam generator pressure will start increasing as the steam generators are coupled to the reactor coolant system, with the steam generator system overpressure protection provided by the pressurizer safety valves. The actuation of the EHRS will terminate the event and the pressure increase in the steam generator system, and proceed to cooldown the plant using this passive system.
- **Reactor Trip.** Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the rapid power reduction on the turbine trip. Trip signals are expected due to low steam/feed flow, high pressurizer pressure, overtemperature ΔT and high pressurizer water level. A trip signal will be rapidly

generated on a low steam/feed trip. In loss of AC power analysis, the loss of power is assumed to occur once a reactor trip setpoint, typically the low feedwater flow, is reached. If the loss of power is assumed as the initiating event, the first results would be an immediate reactor trip and the concomitant coastdown of the reactor coolant pump. This leads to a less conservative scenario than assuming that power is lost only following reactor trip.

- **Safety Systems and Single Failure.** The only safety function required is the core decay heat removal that is carried by the EHRS. Due to the EHRS design, no single failure prevents operation of the system or any one of its subsystems: the actuation of each EHRS subsystem requires the closing of one of the two feed and steam isolation valves, and the opening of one of the two fail-open valves arranged in parallel at the EHRS discharge into the feedline. Because no single failure can be assumed that impairs the opening of both fail-open valves or of both the isolation valves on the feed or steam line, the failure of a single fail-open valve is assumed as the limiting single failure.

  Since the IRIS design is still being completed and different power operation programs are still being evaluated to identify the optimal operational strategy, this analysis assumes a very conservative actuation logic for the EHRS to provide adequate design flexibility and to demonstrate the capability of the EHRS in meeting the acceptance criteria under very severe assumptions. EHRS Actuation Signals based of low feedwater and no startup feedwater flow, low steam generator levels and primary side variables are not credited. Also, a maximum turbine trip delay of 5 seconds following reactor trip is assumed. This has the effect of reducing the pressure and level in the steam generators and results in a longer time to reach the high steam pressure setpoint in the main steam system. Since this is the only actuation signal credited in the analysis, this leads to a long time between the loss of heat removal capability that follows the loss of normal feedwater and the turbine trip and the actuation of the EHRS.

2.2.2.3 Results

Only a partial analysis of the loss of normal feedwater and loss of offsite power is provided here. Each event is analyzed for 6,000 seconds in this study.

The transient response of IRIS following a loss of AC power to the plant auxiliaries is shown in Figures 2.2.2-1 through 2.2.2-8. A calculated sequence of events for this event is listed in Table 2.2-1.

Immediately following actuation, the heat transfer capability of the EHRS is sufficient to start cooling down the plant. The actuation of the EHRS is delayed due to the low RCS initial temperatures assumed and to transient assumptions that tend to delay the actuation of the EHRS on a high steam pressure signal. Note that only the high steam pressure setpoint has been credited for EHRS actuation, and this conservative assumption has been made to demonstrate the large thermal inertia available in the IRIS RCS that, coupled with the large pressurizer steam volume, provides an extended grace period before decay heat removal becomes necessary. Following actuation of the EHRS, the cooldown progresses until a low Tcold signal setpoint is reached. This signal actuates the emergency boration tanks. During this transient, the emergency boration

tanks operate in water recirculation mode. Due to the limited water inventory in the boration tanks, the actuation of the EBTs does not significantly modify the evolution of the transient, and does not lead to a significant increase in the RCS fluid volume. Note that for the purpose of verifying the capability of the EBT and of the low Tcold S-signal to provide borated water and prevent any potential for return to power during the cooldown, a case with maximum reactivity feedback coefficients should also be analyzed. This analysis is not provided here but a preliminary assessment was performed to verify the design basis of the EBT. A complete analysis shall be provided later as part of the IRIS design certification

As the plant cools down, the density of the coolant is increased and this leads to a volume shrinkage. While the cooldown proceeds, assuming that no normal charging and boration is available, the pressurizer volume will empty and the pump suction will be uncovered. In this case a natural circulation path is maintained through the steam generator shroud checkvalves, that open following a loss of forced flow from the reactor coolant pumps. This flow path allows a cooldown of the plant to the safe shutdown condition. Note that even with all eight steam generator check valves unavailable, the reactor coolant system will reach a stable condition, but at a higher temperature and pressure than the safe shutdown condition.

The cooldown of the reactor coolant system is reduced during the transient as the heat removal capacity of the EHRS is lowered due to the reduction of the reactor coolant system temperatures.

Since only two of four of the EHRS subsystems are automatically actuated in this sequence, the steam generators connected to the two idle subsystems remain isolated. The pressure in these steam generators in dictated by the temperature in the reactor coolant system during the transient.

The safety valves of the pressurizer may open during the initial part of the transient, but are not actuated during the cooldown since the EHRS heat removal rate always exceeds decay heat.

The transient response of the IRIS following a loss of normal feedwater is shown in Figures 2.2.2-8 through 2.2.2-16. A calculated sequence of events for this event is listed in Table 2.2-1.

The sequence of events for this case is similar to the loss of offsite power case. However, reactor coolant pump flow remains available until a safety grade pump trip is generated on the low pressurizer level or loss of subcooling margin at the pump suction. No other significant difference between the two cases occur.

2.2.2.4 Conclusions

Results of the analyses and qualitative considerations show that the plant design is such that a loss of normal feedwater or loss of offsite power does not adversely affect the core, the reactor coolant system, or the steam system. The heat removal capability of the EHRS is such that reactor coolant water is not relieved from the pressurizer safety valves. The EHRS is capable of removing decay heat and proceed to a plant cooldown.

The integrity of the core is maintained by operation of the reactor protection system (i.e., the DNBR will be maintained above the limit value), and reactor coolant system and steam generator pressures remain below 110% of their design value.

2.2.3 Feedwater System Pipe Break

2.2.3.1 Identification of Causes and Accident Description

A major feedwater line rupture is a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators in order to maintain fluid inventory in the steam generator tubes. If the break is postulated in a feedwater line between the check valve and a steam generator pair, fluid from the steam generator pair may also be discharged through the break. A break upstream of the feedwater line check valve would affect the plant only as a loss of feedwater. This case is covered in previous Section 2.2.2.

Depending on the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup. Due to the limited water inventory in the IRIS steam system, the potential for a cooldown event is not significant. Moreover, potential reactor coolant system cooldown resulting from a secondary pipe rupture is discussed in detail in Section 2.1.3. Therefore, only the reactor coolant system heatup effects are evaluated here for a feedwater line rupture.

The feedwater line rupture reduces the ability to remove heat generated by the core from the reactor coolant system for the following reasons:

- Feedwater flow to the steam generators is reduced. This reduces the heat removal from the steam generators and causes reactor coolant temperatures to increase prior to reactor trip.

- The break may be large enough to prevent the addition of main feedwater or startup feedwater after reactor trip.

- Fluid in the affected steam generator pair may be discharged through the break and would not be available for decay heat removal after trip. Also, since in IRIS the safety grade decay heat removal system is provided by the EHRS and this system is connected to the feed and steam line isolation valves, the break may be large enough to prevent operation of one of the four trains of the EHRS. Thus, if the feedline rupture is assumed to occur on one of the feedlines that are connected to one of the two actuated EHRS subsystems, this event will evolve as a loss of normal feedwater where only one EHRS subsystem is available (note that each subsystem of the EHRS is designed so that no single failure will prevent its operation). It is therefore more challenging from the point of decay heat removal than both a loss of normal feedwater or a loss of offsite power, and is analyzed to demonstrate that decay heat removal is effectively performed with a single EHRS available. Note that no credit is taken for any operator actions to manually align the two subsystems of the EHRS that are not automatically initiated.

A major feedwater rupture is classified as a Condition IV event.

In IRIS, this event presents an evolution similar to the loss of normal feedwater and loss of offsite power events. The reason is that IRIS does not rely on the steam generators inventory for either short term or long term decay heat removal. To mitigate events that result in a loss of secondary side heat removal, IRIS relies on the large reactor coolant system inventory, on the large pressurizer steam volume and on effective heat removal by the EHRS.

The severity of the feedwater line rupture event depends on a number of system parameters, including the break size, initial reactor power, and the functioning of various control and safety-related systems. Based on analogy with other PWRs design, it is assumed in this study that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. This assumption will be verified prior to design certification by performing adequate sensitivity studies. At the beginning of the transient, the main feedwater is assumed to malfunction due to an adverse environment. Interactions between the break and the main feedwater control system result in no feedwater flow being injected or lost through the steam generator feedwater nozzles. To conservatively account for the worst possible interaction between the break and the feedwater control system, no credit is taken for the low steam/feed flow signal that would be rapidly generated following a loss of feedwater flow. The reactor is tripped on any of a number of protection signals on the primary (high pressurizer pressure, high pressurizer level, Overtemperature ΔT,...) or secondary (low steam pressure, high steam pressure, ...) system.

After reactor trip, a full double-ended rupture of the feedwater line is assumed such that the faulted steam generator blows down through the break and no main feedwater is delivered to the intact steam generators. These assumptions delay the occurrence of the break until the point where the primary and secondary system conditions are more severe, and should represent the most limiting feedwater rupture that can occur. Sensitivity studies must be completed to verify this assumption, but this is considered sufficient for this stage of investigation, that aims at providing an evaluation of the transient evolution and of the available margins.

An analysis is performed both with and without offsite power available. For the case where offsite power is not available, the loss of power is assumed to occur at the time of reactor trip, since this is more conservative than the case where power is lost at the initiation of the event. The only difference between the cases with and without offsite power available is the operating status of the reactor coolant pumps.

The following provides the protection for a main feedwater line rupture:

- A reactor trip on one of the following conditions:
  - High Pressurizer Pressure
  - Overtemperature ΔT
  - Low steam/feed flow
  - "S" signals from either of the following
    - Two out of four low steam line pressure in any steam line
    - Two out of four high containment pressure (high-2)

- The EHRS provides a passive method for decay heat removal. The four trains of the EHRS remove heat from the primary system and use the RWST as heat sink. Operation of a train of the EHRS is initiated by the opening of one of two parallel power-operated valves at the EHRS cold leg.

The EHRS is designed to assure that adequate decay heat removal is provided, assuming that one of the two automatically actuated trains of the EHRS is connected to the faulted feedline, such that:

A. No substantial overpressurization of the RCS shall occur; and

B. Sufficient liquid in the RCS shall be maintained in order to provide adequate decay heat removal.

Due to design features of IRIS and in particular to the sizing basis for the EHRS, this event is analyzed to demonstrate that the same acceptance criteria defined for a loss of normal feedwater are met.

2.2.3.2 Method of Analysis

The RELAP5 code (discussed in Section 2.0.11) has been used in this analysis. The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, and steam generators. The program computes pertinent plant variables including temperatures, pressures, and power level.

The cases analyzed assume a double-ended rupture of the largest feedwater pipe at full power. Major assumptions made in the analyses are as follows.

- Initial Operating Conditions Initial conditions of core power, reactor coolant temperature, and pressurizer pressure are obtained by applying maximum errors to the nominal full-power values in the conservative direction. For some parameters a complete sensitivity study is required to identify the most conservative combination of initial assumption. For this preliminary analysis, qualitative considerations based on the comparison with other PWRs have been used to identify the most conservative scenario. A complete quantitative assessment shall be performed as part of the design certification analyses.

- Reactivity Coefficients Since the feedline rupture event results in a primary system heatup, it is conservative to model minimum moderator reactivity feedback conditions. Hence, minimum moderator reactivity feedback conditions are assumed for this event. A conservatively large absolute value of the Doppler-only power coefficient is used (see Table 2.0-5). This is equivalent to a total integrated Doppler Reactivity from 0- to 100-percent of 0.016 Δk.

- Reactor Control It is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

- Steam Relief No credit is taken for operation of the steam dump system. The steam generator pressure rises during the transients until the EHRS is actuated.

- <u>Pressurizer Control</u> The current IRIS design does not feature sprays. Pressurizer Safety Valves setpoint is assumed to be at its minimum value. No credit is taken for charging and letdown.

- <u>Feedwater Flow</u> Main feedwater to all steam generators is assumed to stop at the time the break occurs (all main feedwater spills out through the break). No credit is taken for start-up feedwater flow.

- <u>Break Area</u> The worst possible break area is assumed.

- <u>Reactor Trip</u> Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the rapid power reduction on the turbine trip. The Reactor Trip during this event occurs in this analysis due to a high pressurizer pressure signal. No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

  - Low steam/feed flow

  - Overtemperature ΔT

  - High pressurizer level

  - High containment pressure

- <u>Safety Systems and Single Failure.</u> The only safety function required is the core decay heat removal that is carried by the EHRS. Due to the EHRS design, no single failure prevents operation of any subsystem: the actuation of each EHRS train requires the closing of one of the two feed and steam isolation valves, and the opening of one of the two fail-open valves arranged in parallel at the EHRS discharge into the feedline. Because no single failure can be assumed that impairs the opening of both fail-open valves or of both the isolation valves on the feed or steam line, the failure of a single fail-open valve is assumed as the limiting single failure.

- <u>Offsite power assumptions.</u> Two cases are considered in the analysis, both with and without offsite power available. For the case without offsite power available, the loss of offsite power is assumed at the time of reactor trip. This is more conservative than the case where power is lost at the initiation of the event. Note that the case with offsite power available is not analyzed here because, due to the fast actuation of the emergency boration tank actuation (on an S-Signal generated by the low steam pressure), the reactor coolant pumps are tripped by the protection system a few seconds after reactor trip. Since the only difference between the cases with and without offsite power available is the status of the pumps, the case with offsite power available does not present significant differences from the case without offsite power.

The EHRS is initiated when the pressure in any steam line drops to the low steam pressure signal. This signal also initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives an S-Signal that initiates flow of cold borated water from the emergency boration tanks to the reactor coolant system. Thus, upon reaching the low steam pressure setpoint, an S-Signal is generated, two trains of the EHRS are aligned, the EBTs are aligned, and the feed and steam isolation valves are closed.

No reactor control systems are assumed to function. The reactor protection and safety monitoring system is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system.

The engineered safety systems assumed to function are the EHRS, the Emergency Boration system and steam and feed line isolation valves. For the EHRS the worst possible configuration is assumed, with one of the two automatically aligned trains connected to the faulted feedline and therefore not available for decay heat removal. No single failure prevents operation of this system, and thus the single failure assumed is the failure of one of the two parallel discharge valves in the available EHRS train.

No credit is taken for operator action, in particular no credit is taken for any action to align the remaining two trains of the EHRS.

If the RCPs are turned off, there will be a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the RCS has been shown in Section 2.2.2, for the loss of AC power transient, to be sufficient to remove core decay heat following reactor trip. Pump coastdown characteristics are discussed in Sections 2.3.

2.2.3.3 Results

Calculated plant parameters following a major feedwater line rupture without offsite power available are presented in Figures 2.2.3-1 through 2.2.3-8. The calculated sequence of events is listed in table 2.2-1.

The system response to the feedline break event is similar to the response to a loss of offsite power event. It should be noted that this event in IRIS leads to an even less severe evolution than a loss of offsite power case. This is due to the fact that IRIS, different from conventional loop-type PWRs, does not rely on steam generator inventory for early heat removal. IRIS once through steam generators only have a very limited inventory, and the system relies instead on RCS thermal inertia to provide a sufficient grace period for the actuation of the passive decay heat removal systems. Therefore, the rapid loss of inventory during a feedline break does not have a significant impact on the system response. On the other hand, the rapid depressurization of the system leads to a faster actuation of the EHRS on a low steam line pressure in this case than in the loss of offsite power event. While only one train of the EHRS is available for this event, due to the assumption that the break is on one of the feedlines connected to the two actuated EHRS trains, the faster actuation leads to an earlier cooldown. Since the EHRS is sized to provide adequate decay heat removal with only one train available, the only effect of having a single train available is a slower cooldown compared to the loss of offsite power case.

2.2.3.4 Conclusions

The preliminary results presented here show that IRIS response to a feedline break does not present significant differences from a response to a loss of offsite power or loss of normal feedwater. Results of the analyses show that for the postulated feedwater line rupture, a single trains of the EHRS is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core.

Table 2.2-1
TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM (Sheet 1 of 4)

Accident	Event	Time (seconds)
Turbine Trip (2.2.1)		
1. With offsite power available, minimum reactivity feedback, without pressurizer control	Turbine trip, loss of main feedwater flow	0.0
	High pressurizer pressure reactor trip setpoint reached	9.6
	Rods begin to fall into core	11.6
	Initiation of steam release from pressurizer safety valves	19.3
	Peak RCS pressure occurs	19.4
	EHRS actuate	23.9
2. Without offsite power, minimum reactivity feedback, without pressurizer control	Turbine trip, loss of main feedwater flow	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	Reactor coolant pumps undervoltage reactor trip setpoint reached	4.0
	Rods begin to fall into core	4.5
	Initiation of steam release from pressurizer safety valves	N.A.
	EHRS actuate	25.1
	Peak RCS pressure occurs	27.7

Table 2.2-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM (Sheet 2 of 4)

Accident	Event	Time (seconds)
Loss of Offsite Power (2.2.2)		
	Feedwater is lost	0.0
	Low Feed Flow Reactor Trip reached	0.1
	Rods begin to fall into core, AC power is lost, reactor coolant pumps start to coastdown	4.1
	Pressurizer safety valve open (First Time)	231
	EHRS actuation signal reached on high steamline pressure	1632
	Feed and Steam Line Isolation Completed	1646
	EHRS valve completely open	1648
	Maximum pressurizer water level reached	1670
	Pressurizer safety valve close (Final Closure)	1705
	"S" Signal on Low Tcold reached	4035
	Emergency boration tank valve completely open	4067

Table 2.2-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM (Sheet 3 of 4)

Accident	Event	Time (seconds)
Loss of Normal Feedwater (2.2.2)		
	Feedwater is lost	0.0
	Low Feed Flow Reactor Trip reached	0.1
	Rods begin to fall into core	4.1
	Pressurizer safety valve open (First Time)	383
	EHRS actuation signal reached on high steamline pressure	1657
	Feed and Steam Line Isolation Completed	1672
	EHRS valve completely open	1674
	Maximum pressurizer water level reached	~1700
	Pressurizer safety valve close (Final Closure)	1750
	RCP trip on low pressurizer level	2876
	"S" Signal on Low Tcold reached	4166
	Emergency boration tank valve completely open	4198

Table 2.2-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM (Sheet 4 of 4)

Accident	Event	Time (seconds)
Feedwater System Pipe Break (2.2.3)		
	Main Feedwater to all steam generators stops due to interaction between the break and the main feedwater control system	0.0
	High pressurizer pressure setpoint reached, full double-ended rupture starts	14.1
	Pressurizer safety valve open (First Time)	15.5
	Rods begin to fall into core, AC power is lost, reactor coolant pumps start to coastdown	16.1
	"S" Signal on low steam generator level (or low steamline pressure) reached in faulted SG, SG Isolation and EHRS actuation.	25.5
	Pressurizer safety valve close (Final Closure)	34
	Feed and Steam Line Isolation Completed	39.5
	EHRS valve completely open	41.5
	Emergency boration tank valve completely open	74
	Maximum pressurizer water level reached	128

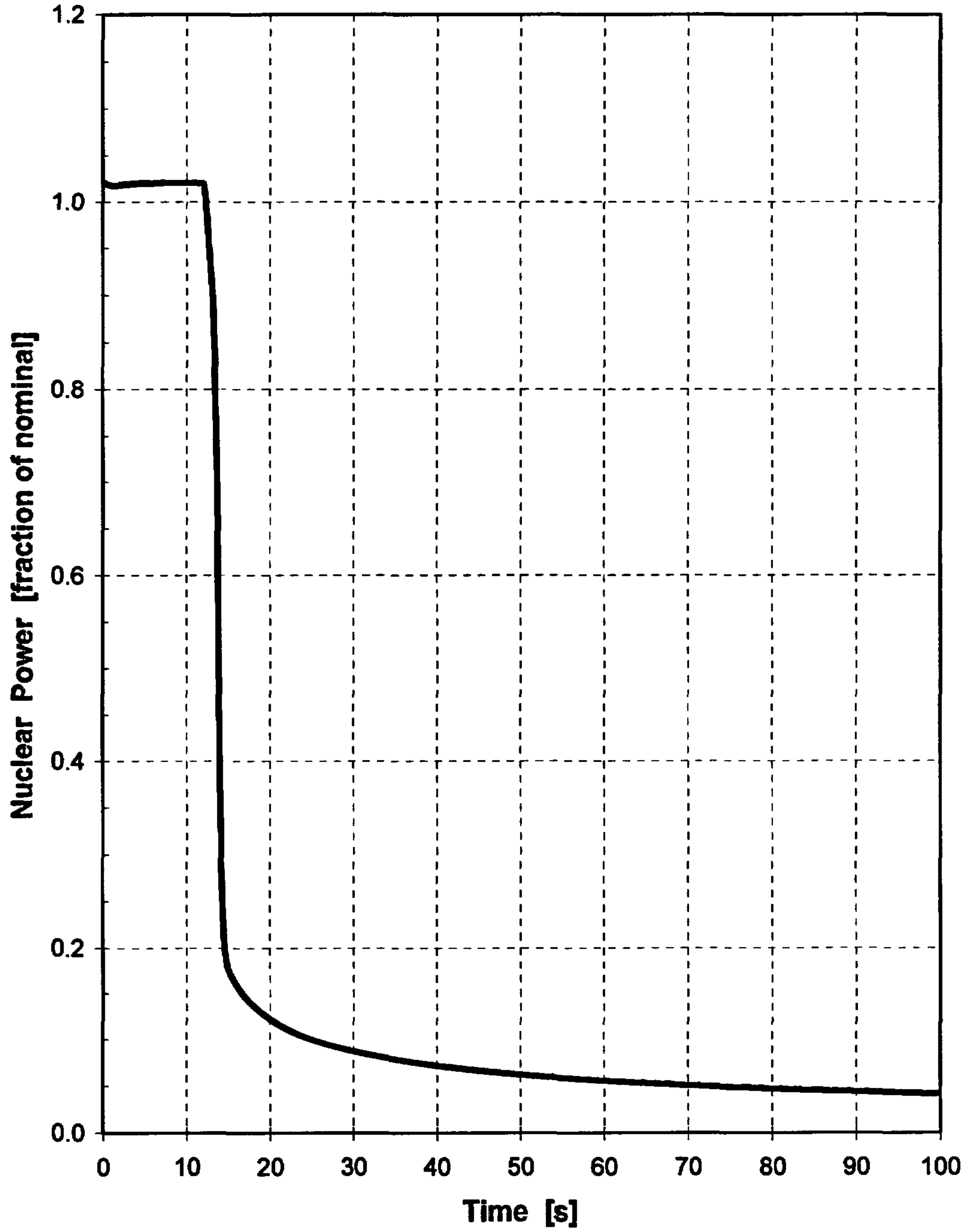


Figure 2.2.1-1
Nuclear Power Transient for Turbine Trip with Offsite Power Available

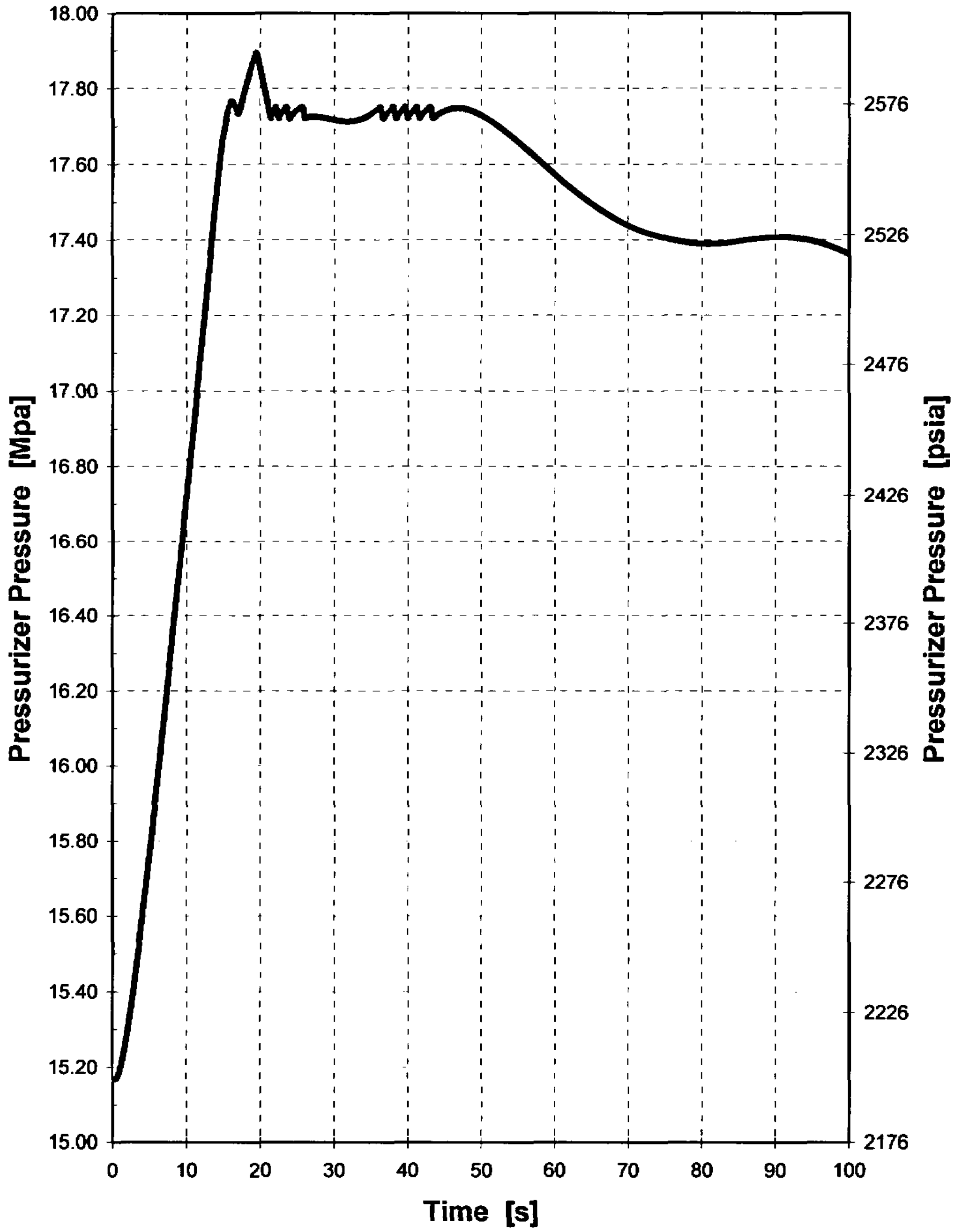


Figure 2.2.1-2
Pressurizer Pressure Transient for Turbine Trip with Offsite Power Available

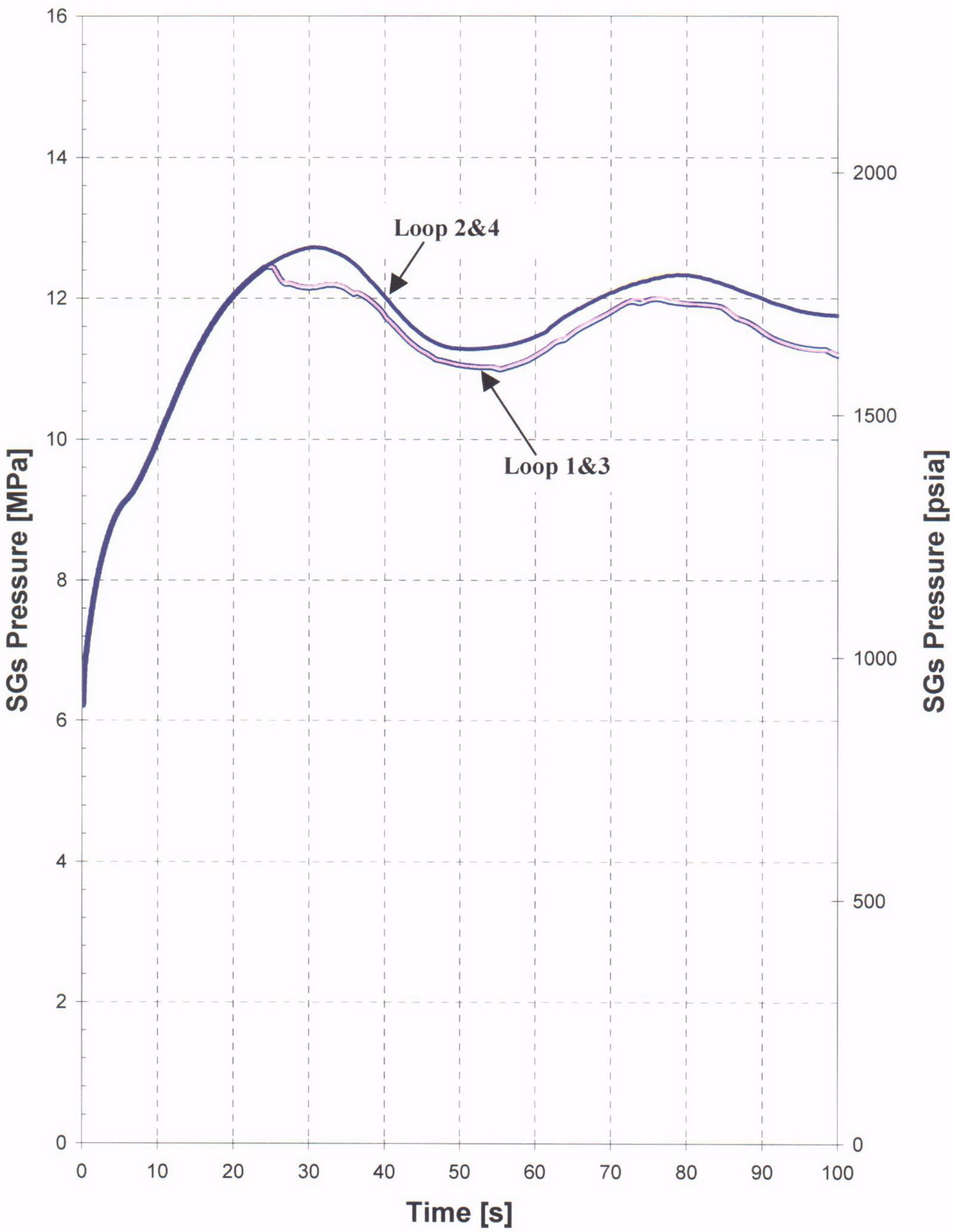


Figure 2.2.1-3
Steam Generator Pressure Transient for Turbine Trip with Offsite Power Available

C02

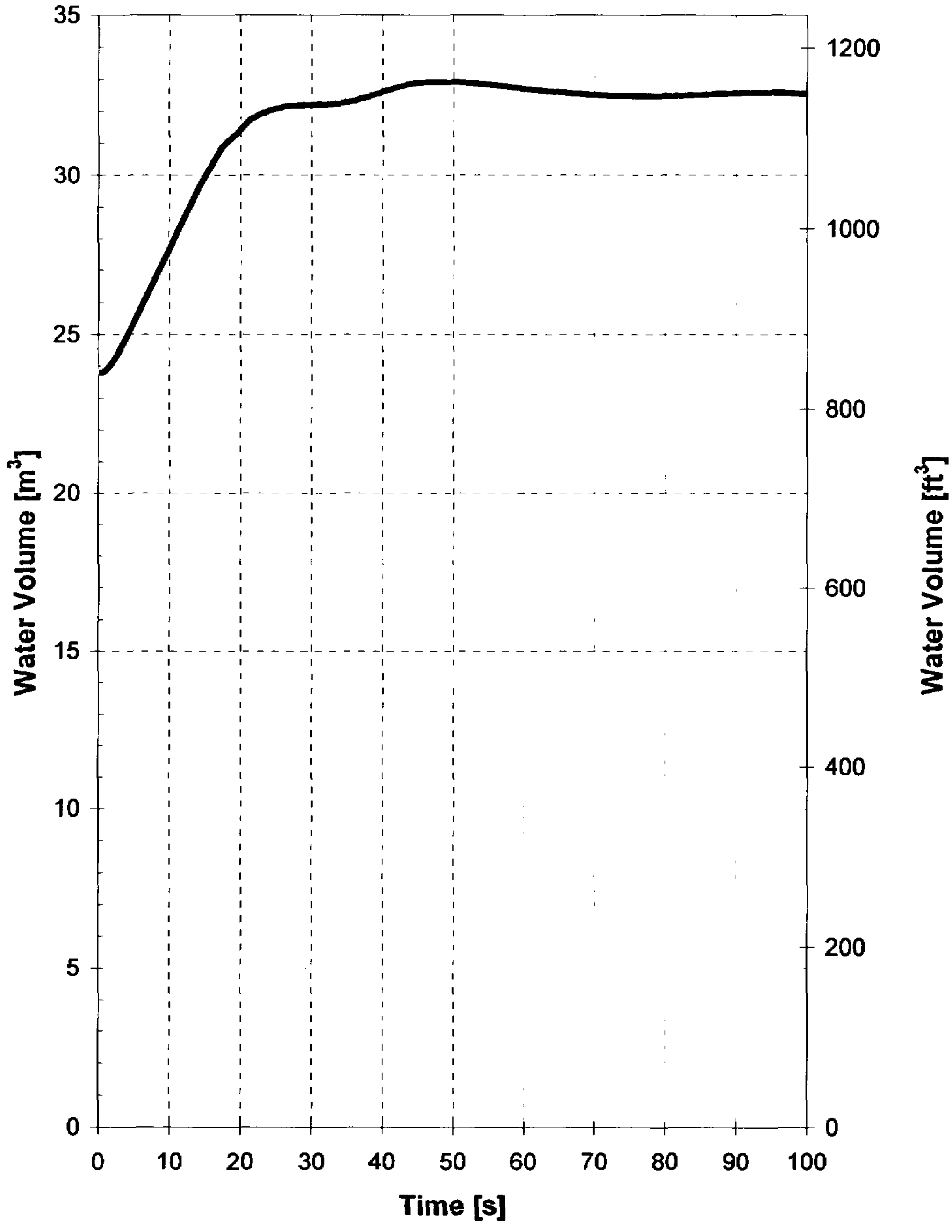


Figure 2.2.1-4
Pressurizer Water Volume Transient for Turbine Trip with Offsite Power Available

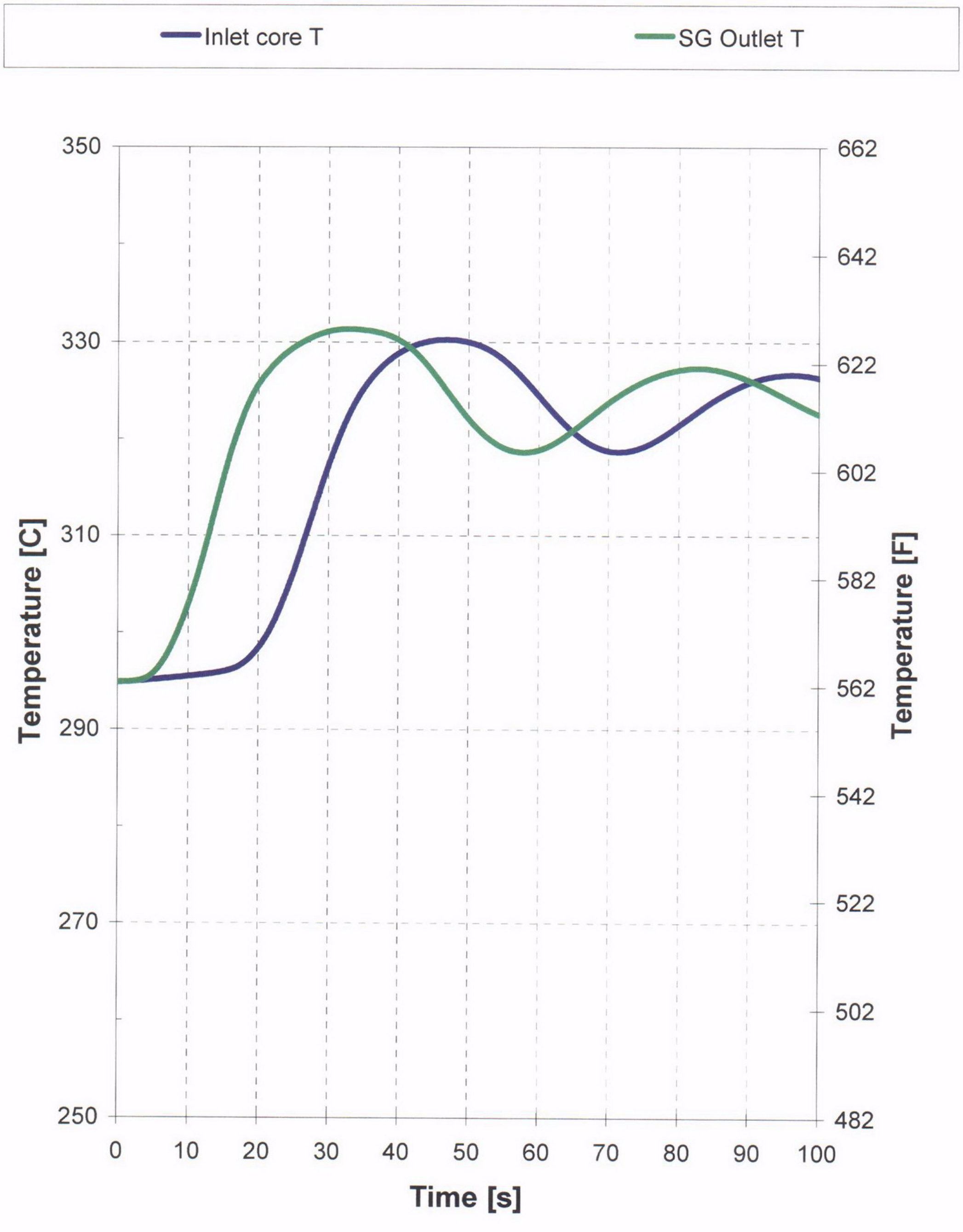


Figure 2.2.1-5
RCS Temperatures Transient for Turbine Trip with Offsite Power Available [1]

C03

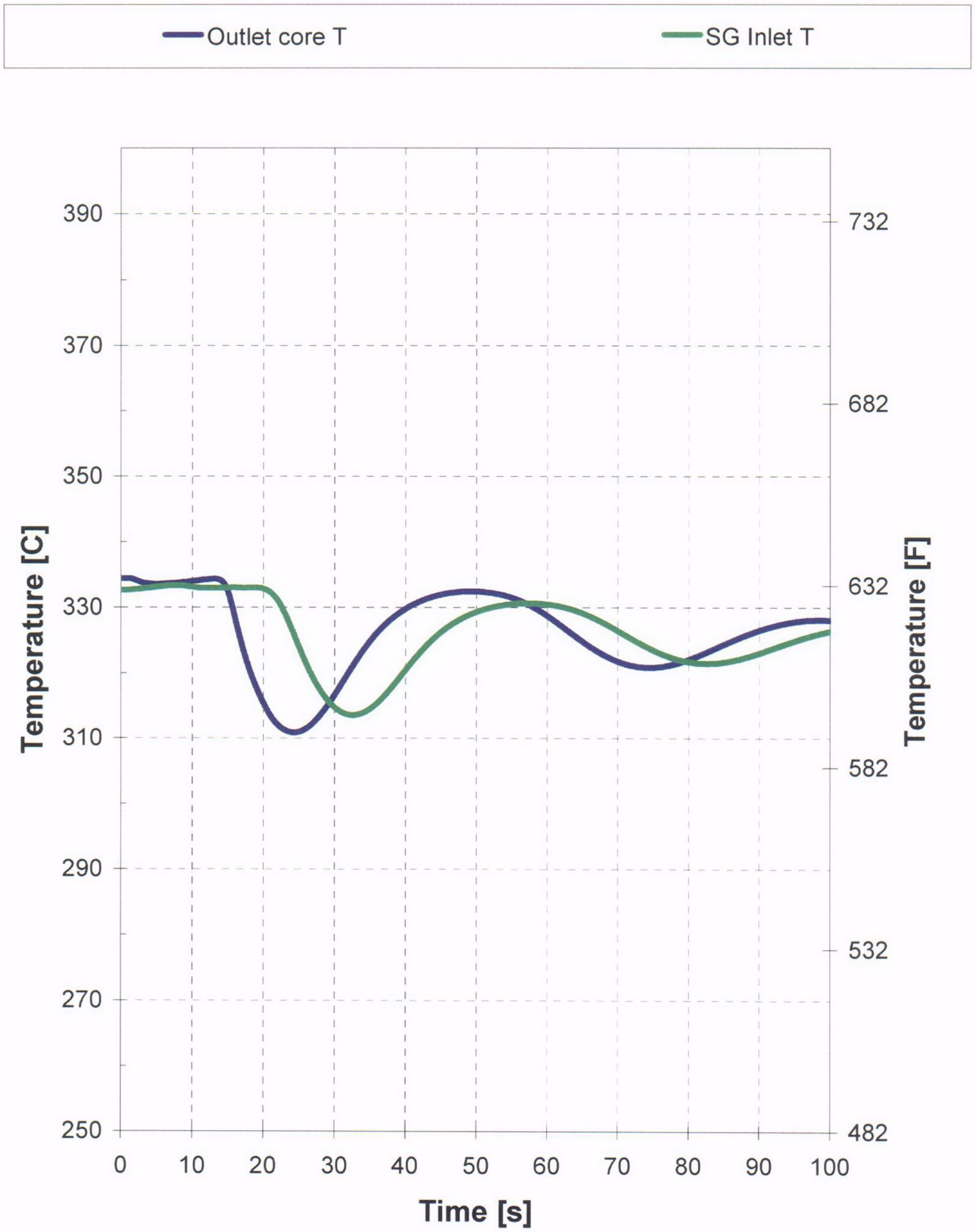


Figure 2.2.1-6
RCS Temperatures Transient for Turbine Trip with Offsite Power Available [2]

C04

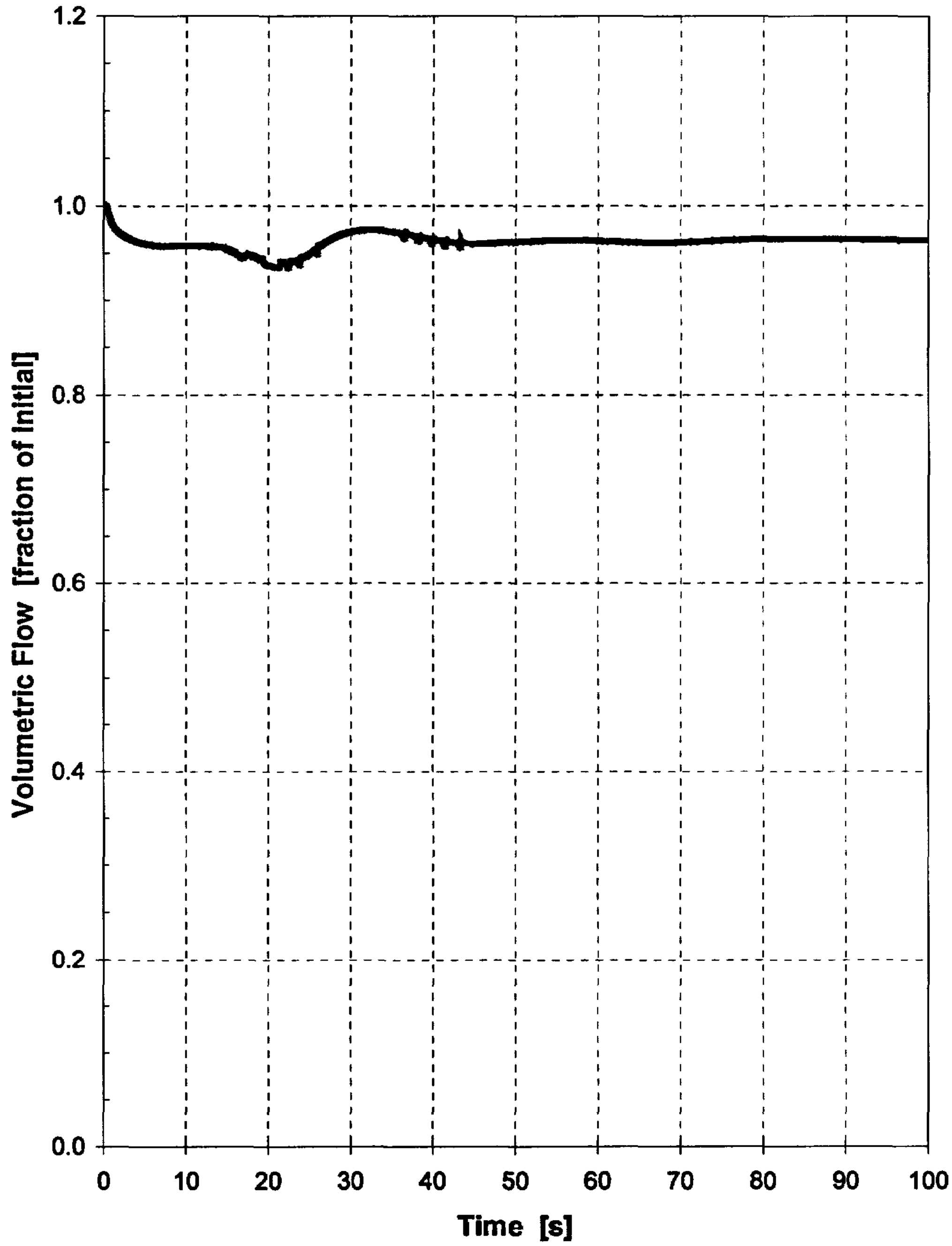


Figure 2.2.1-7
Pump Volumetric Flow Transient for Turbine Trip with Offsite Power Available

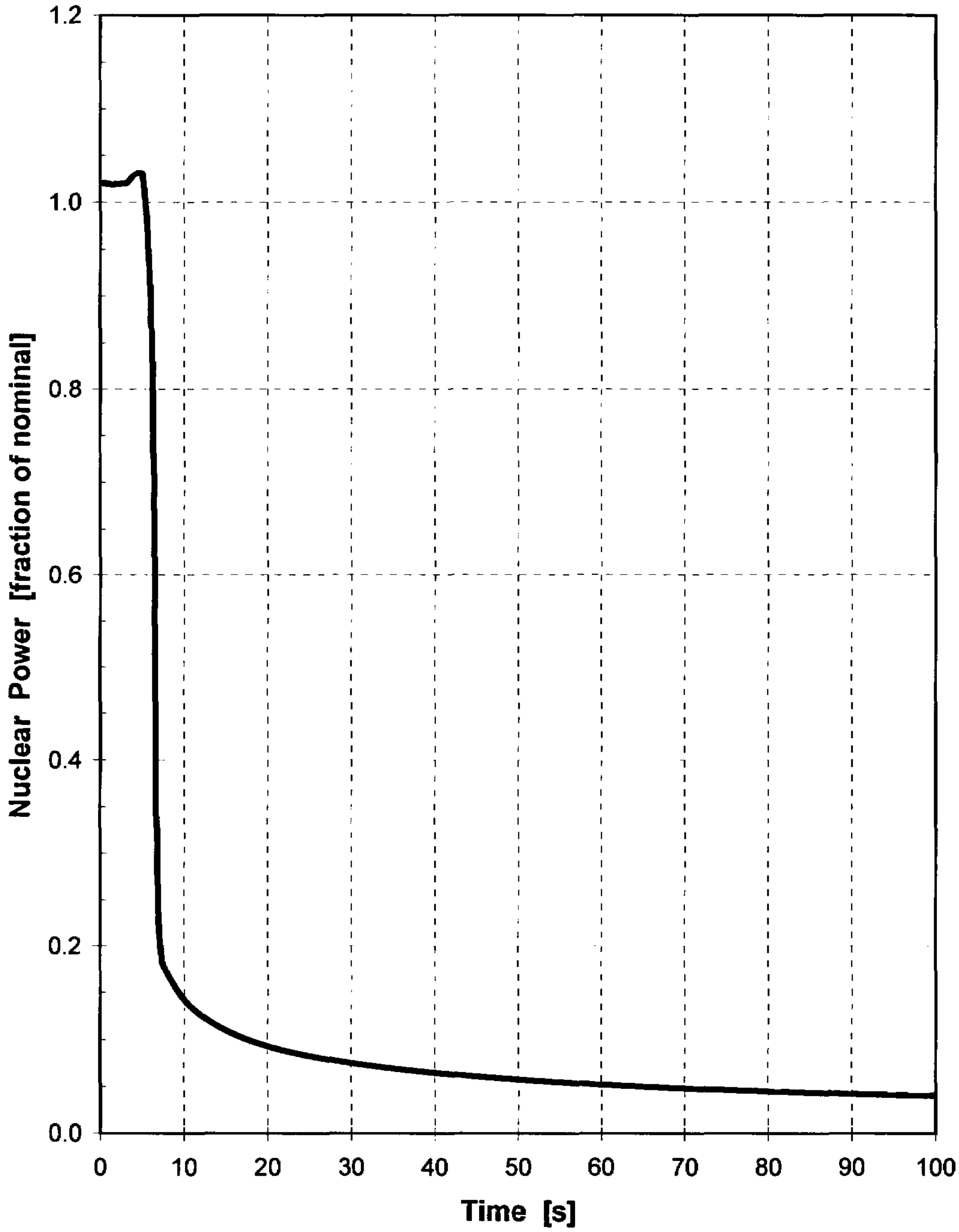


Figure 2.2.1-8
Nuclear Power Transient for Turbine Trip without Offsite Power

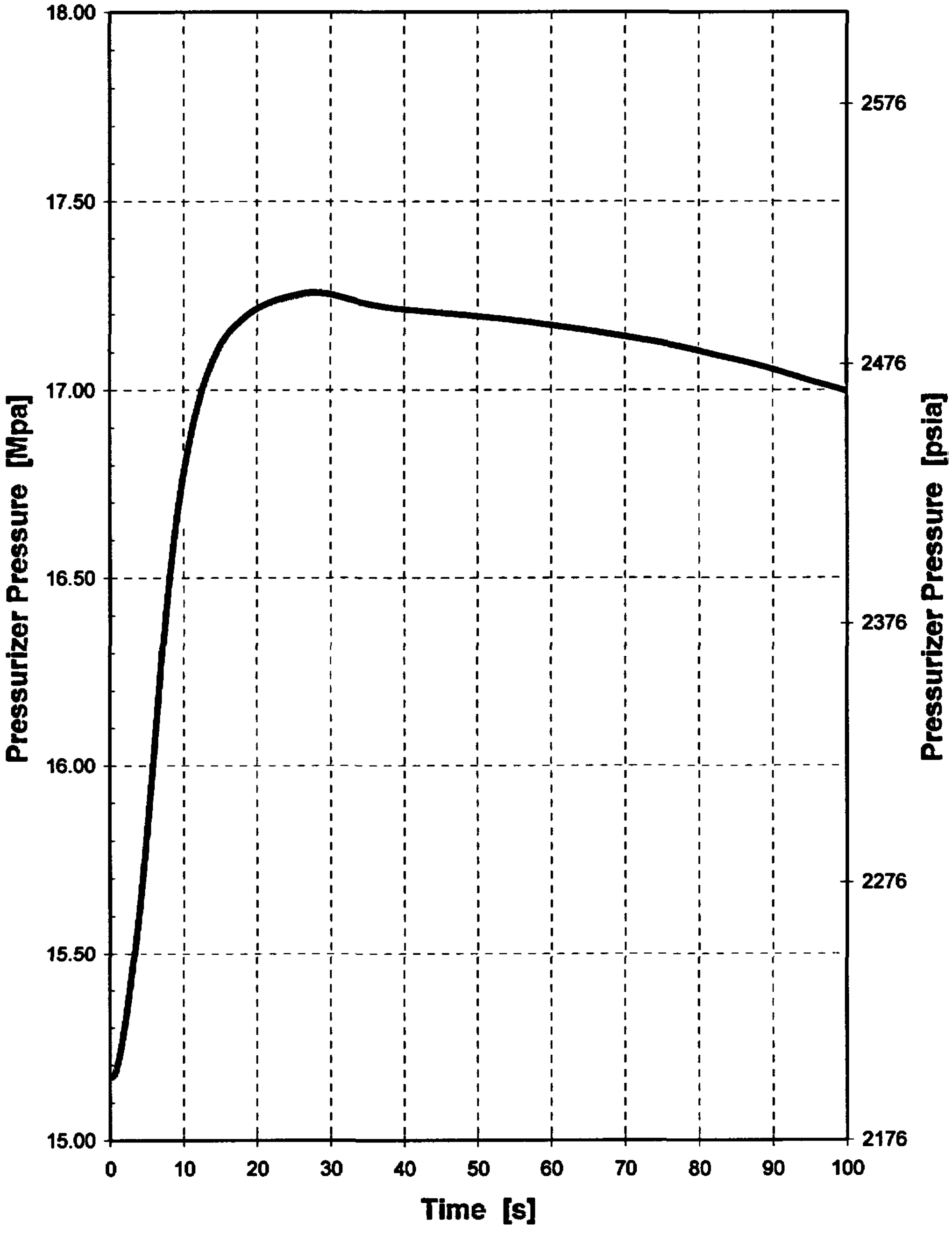


Figure 2.2.1-9
Pressurizer Pressure Transient for Turbine Trip without Offsite Power

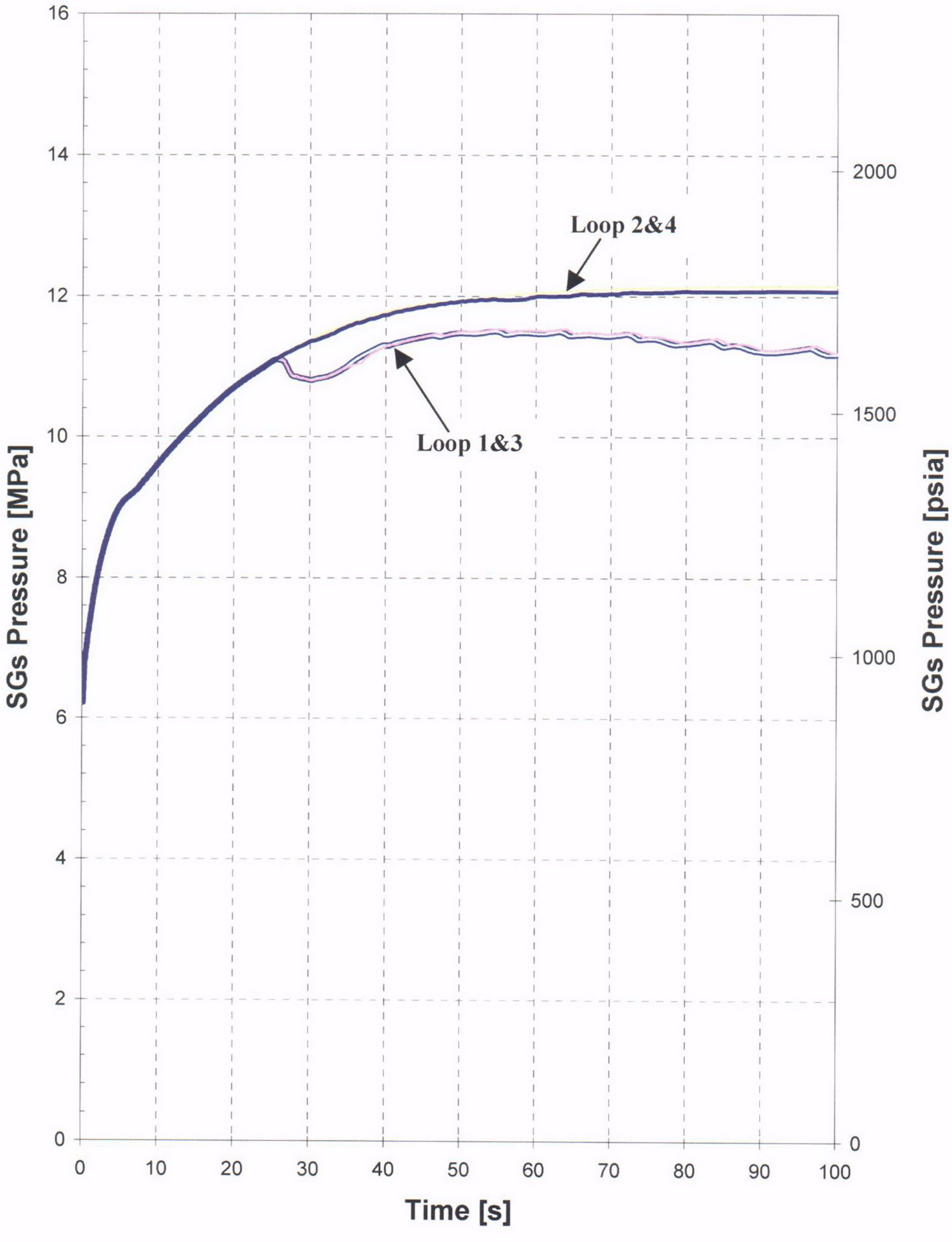


Figure 2.2.1-10
Steam Generator Pressure Transient for Turbine Trip without Offsite Power

C05

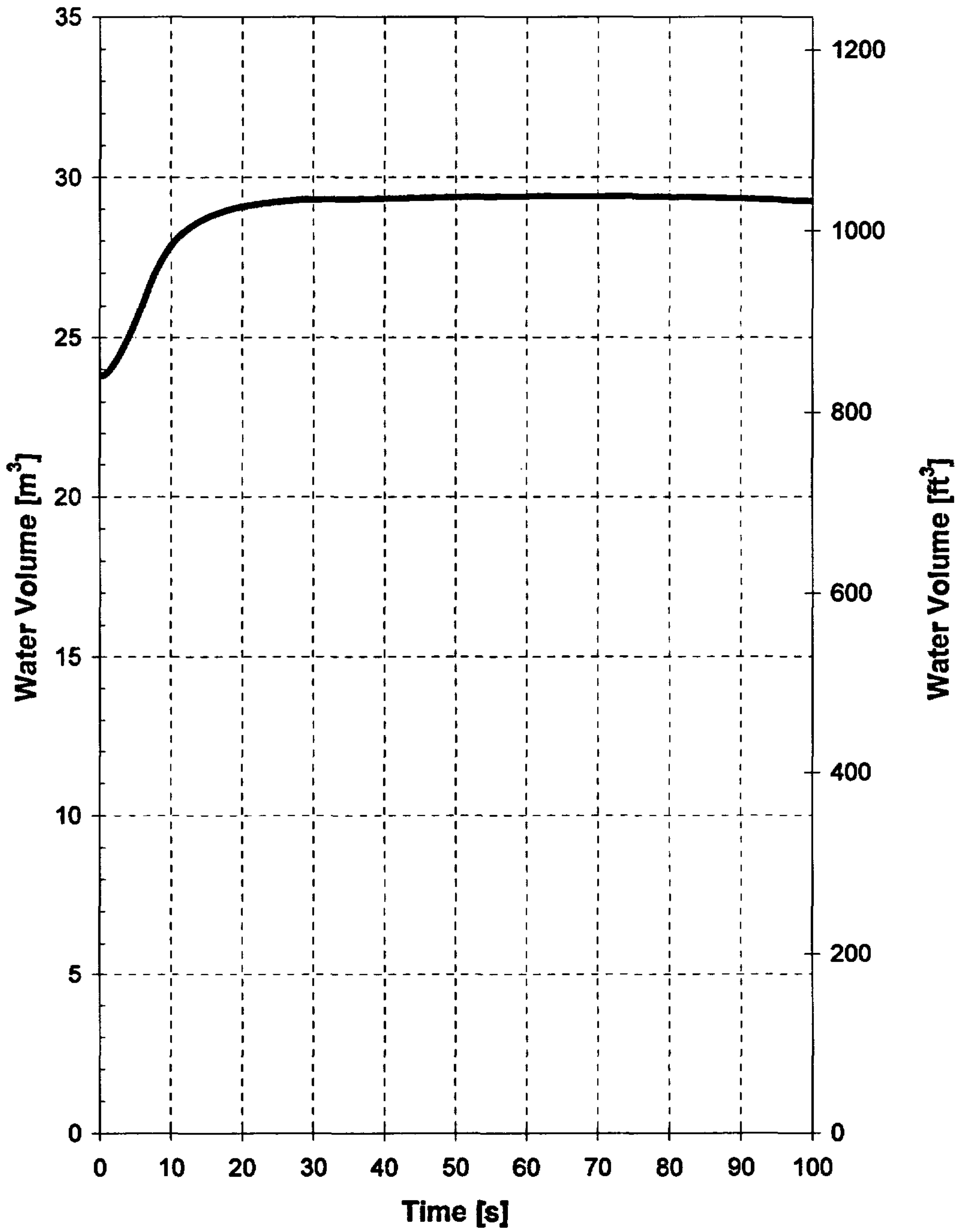


Figure 2.2.1-11
Pressurizer Water Volume Transient for Turbine Trip without Offsite Power

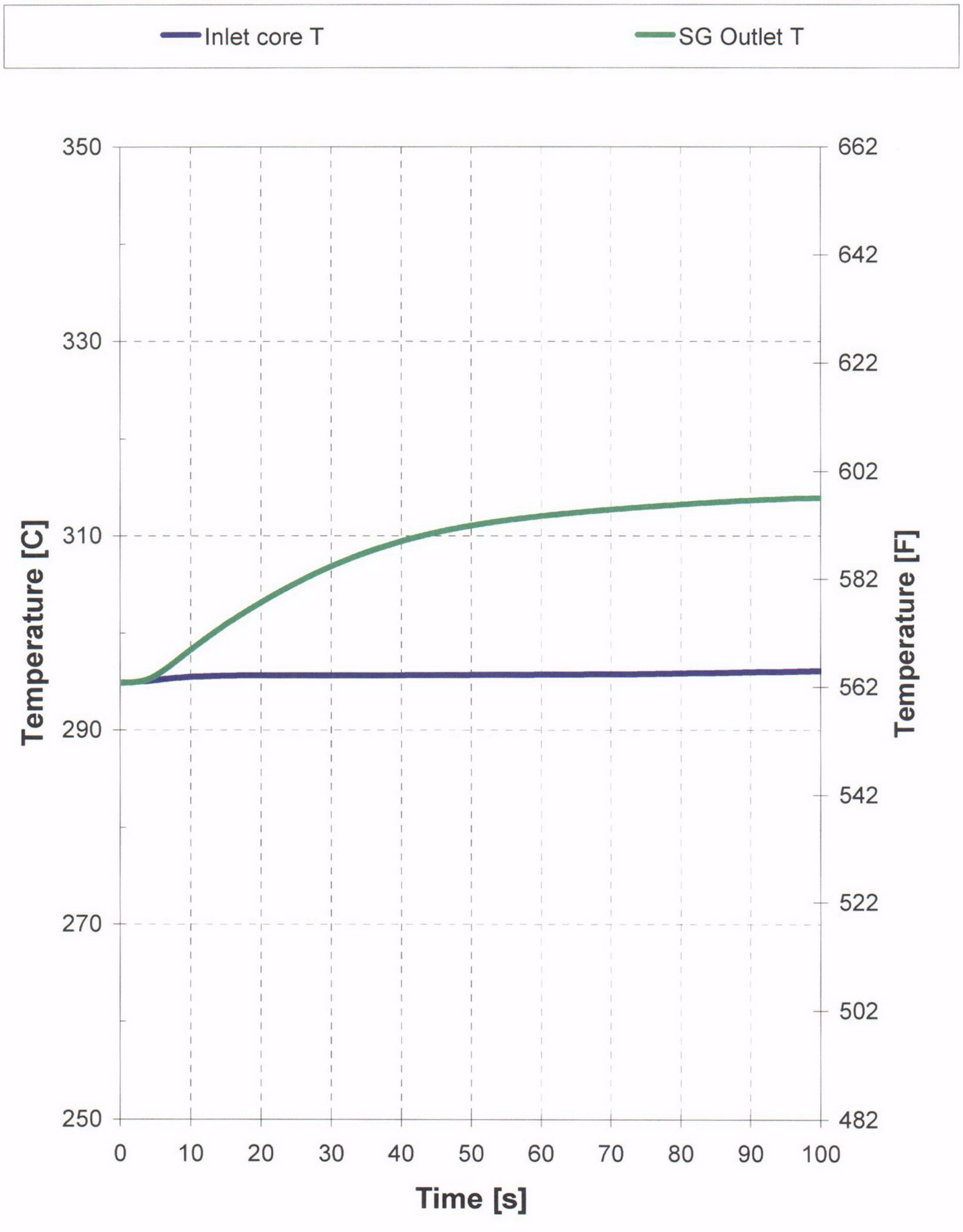


Figure 2.2.1-12
RCS Temperatures Transient for Turbine Trip without Offsite Power [1]

C06

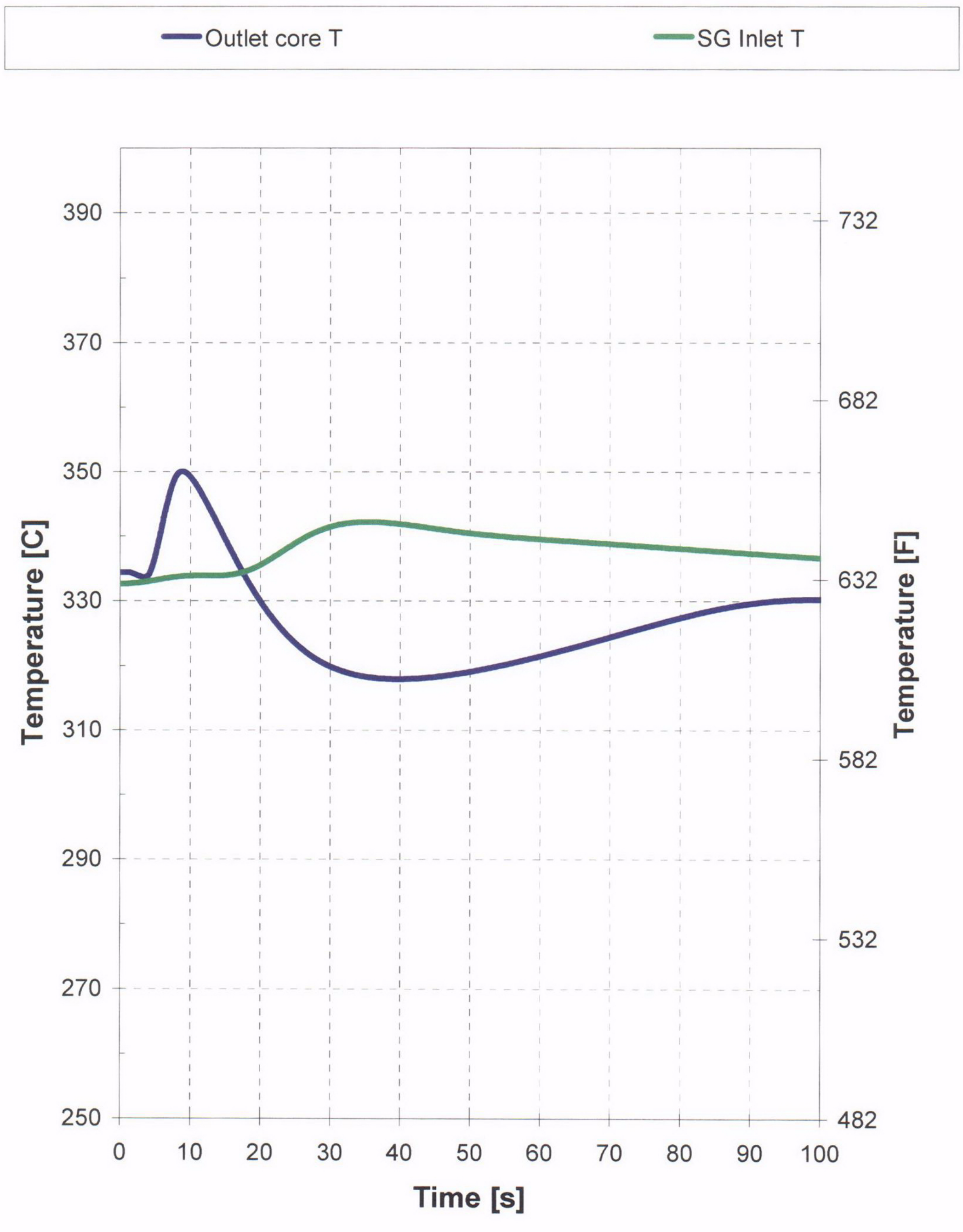


Figure 2.2.1-13
RCS Temperatures Transient for Turbine Trip without Offsite Power [2]

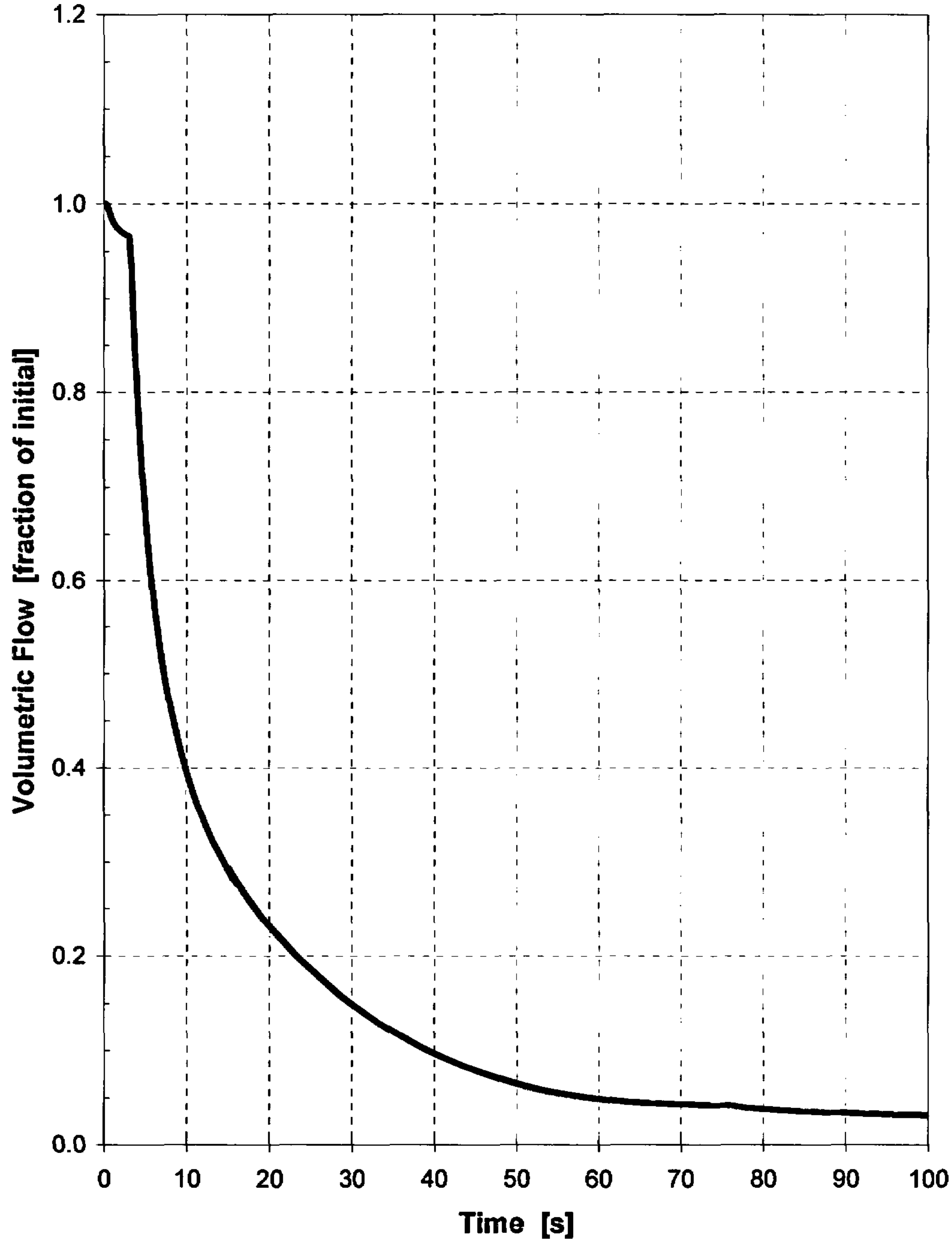


Figure 2.2.1-14
Pump Volumetric Flow Transient for Turbine Trip without Offsite Power

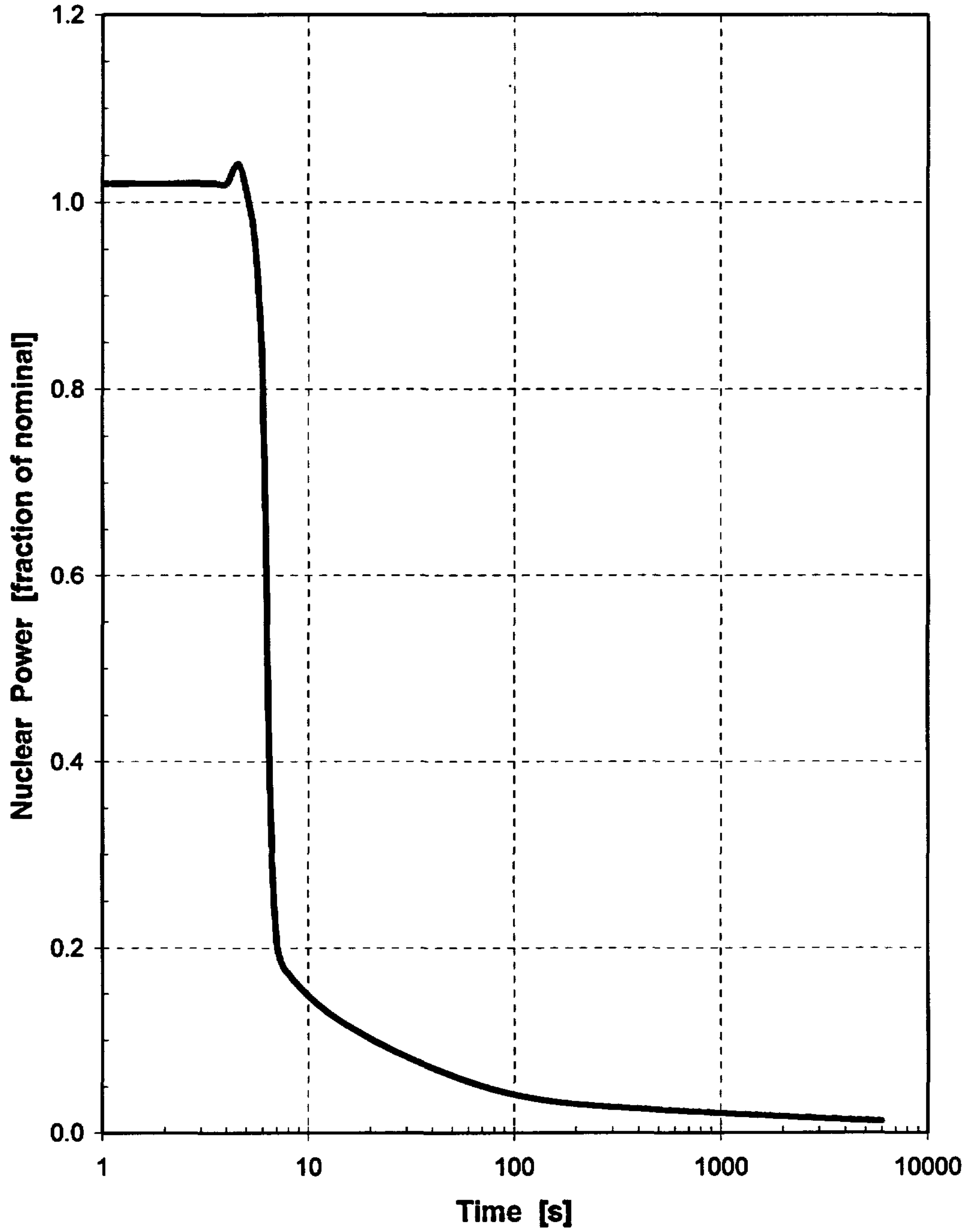


Figure 2.2.2-1
Nuclear Power Transient for Loss of Offsite Power

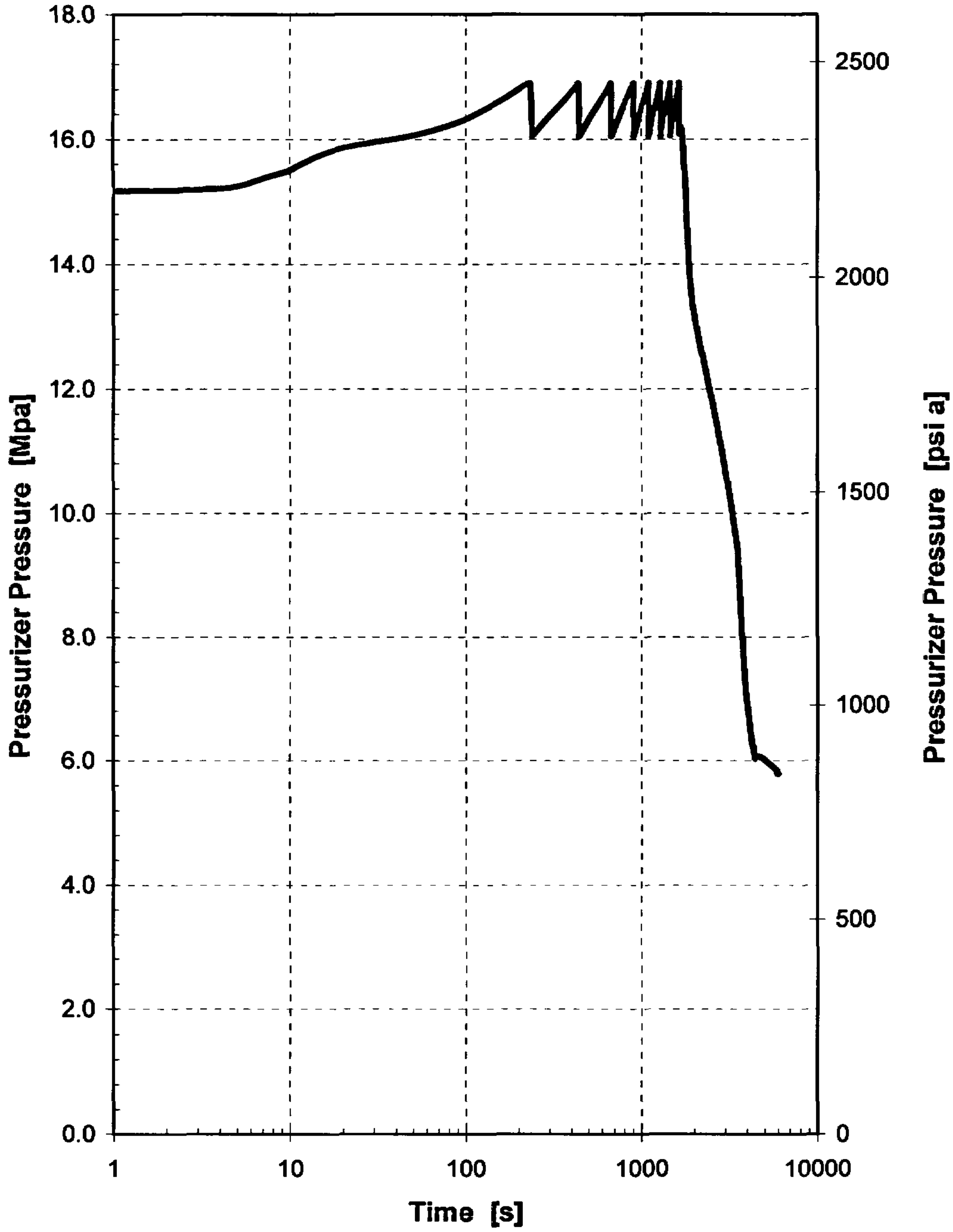


Figure 2.2.2-2
Pressurizer Pressure Transient for Loss of Offsite Power

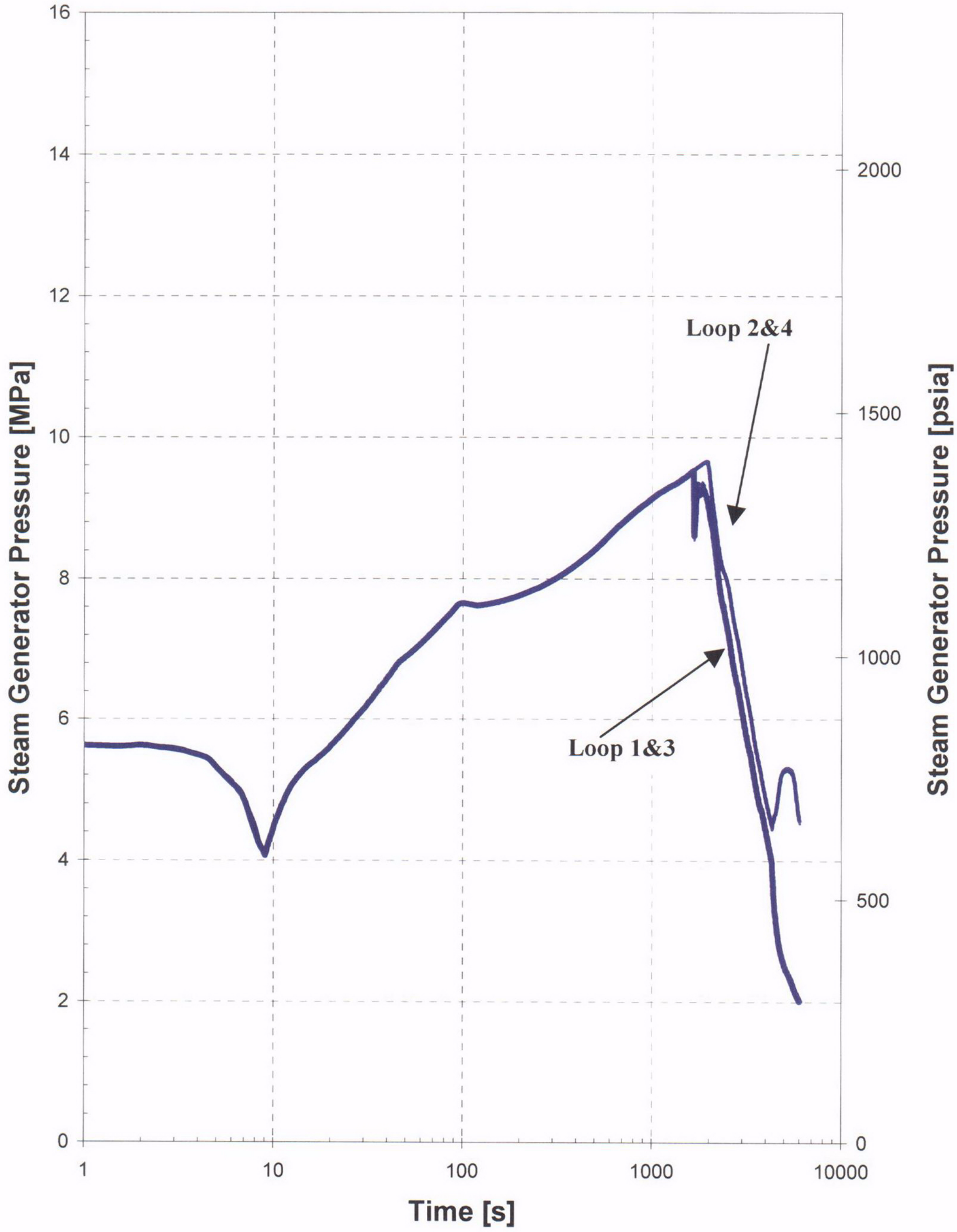


Figure 2.2.2-3
Steam Generator Pressure Transient for Loss of Offsite Power

C07A

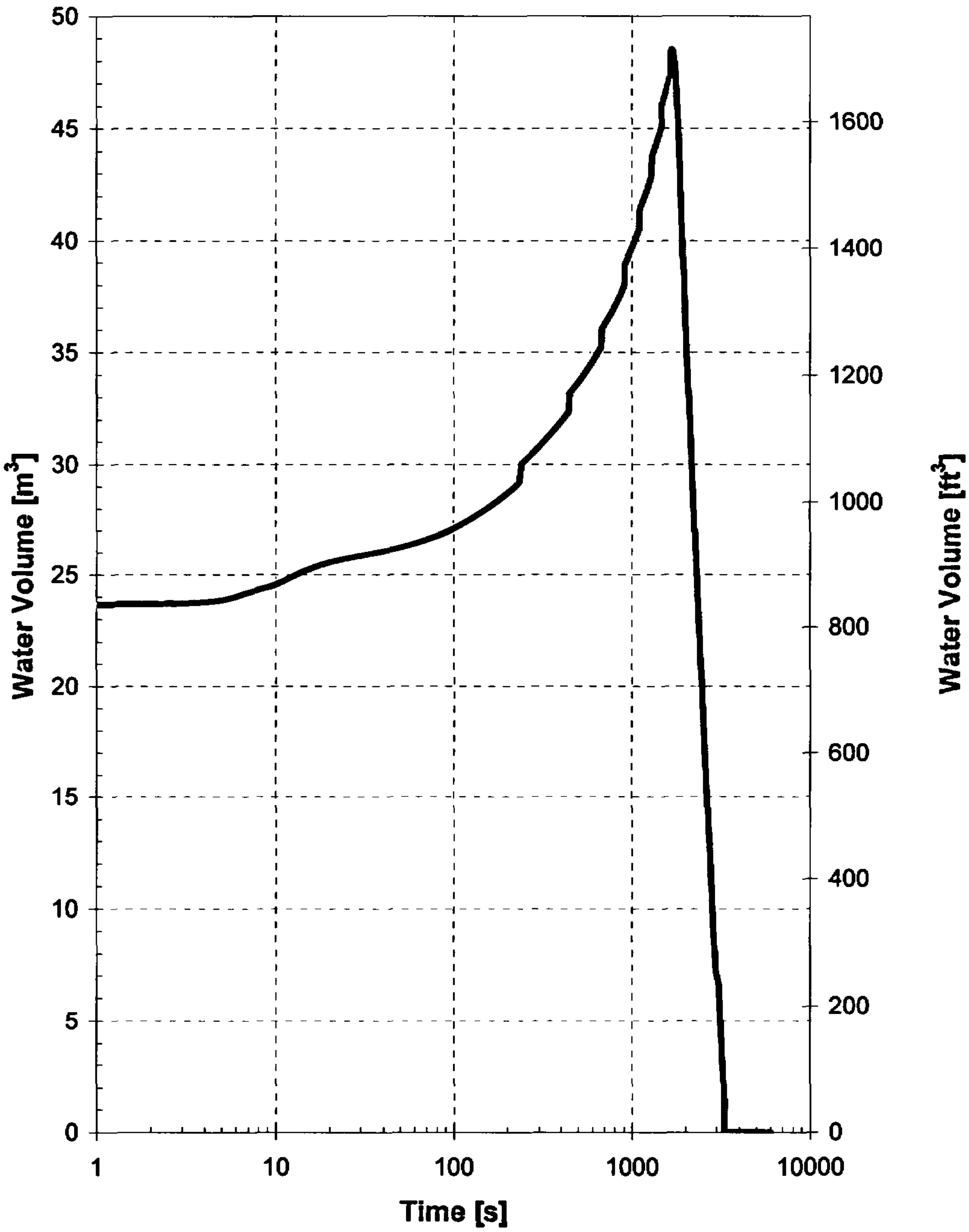


Figure 2.2.2-4
Pressurizer Water Volume Transient for Loss of Offsite Power
(Total Pressurizer Volume 2500 ft3)

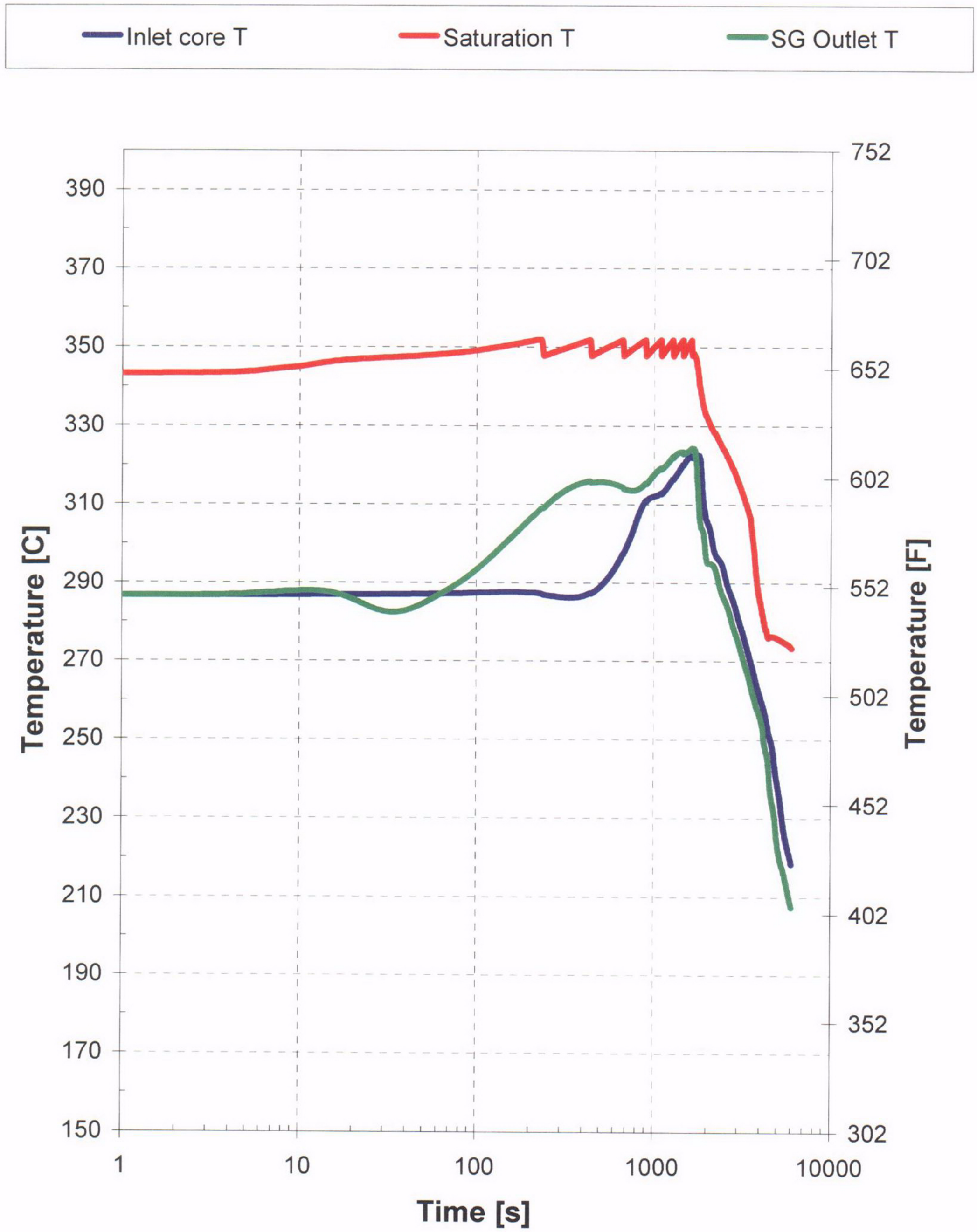


Figure 2.2.2-5
RCS Temperatures Transient for Loss of Offsite Power [1]

c08

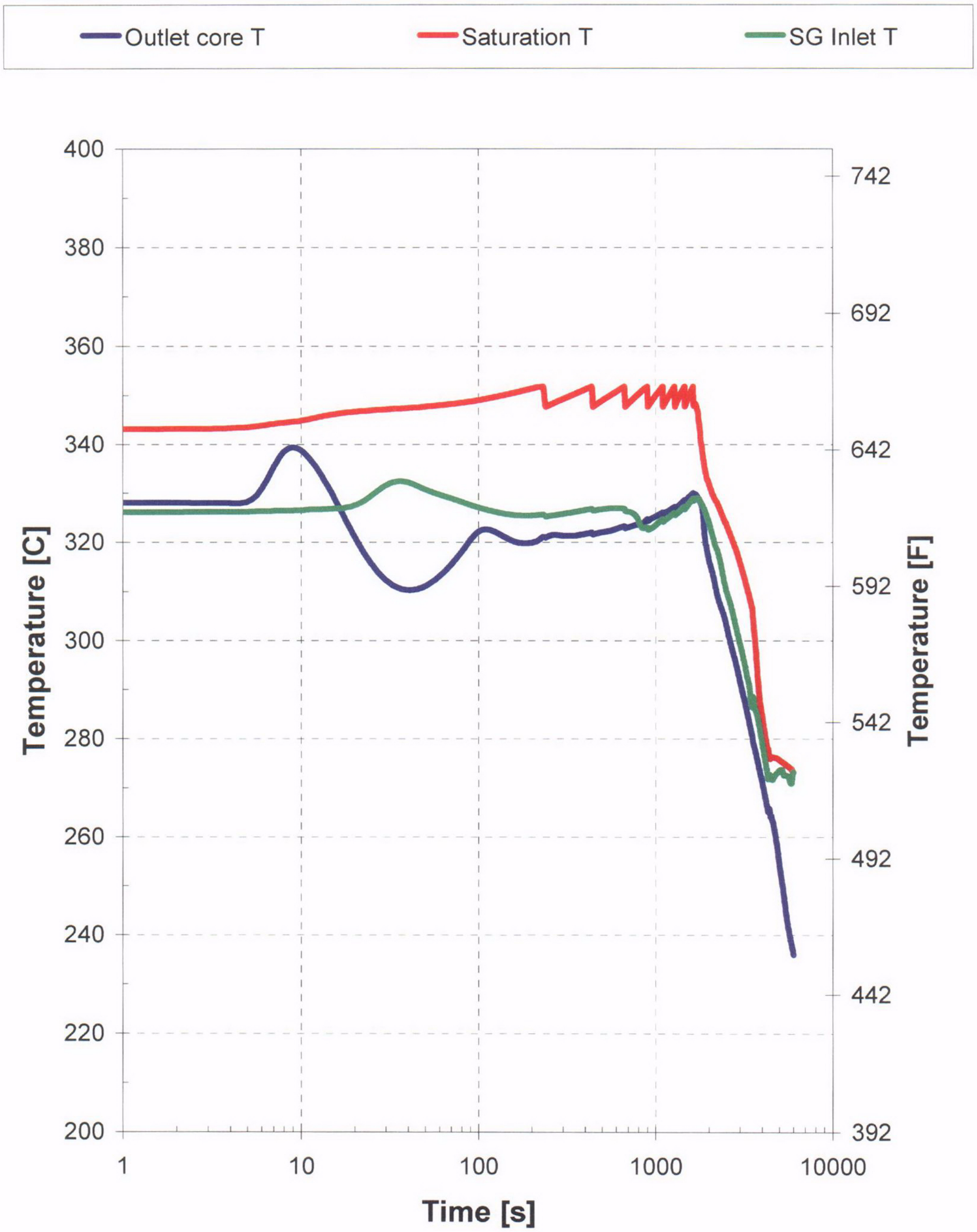


Figure 2.2.2-6
RCS Temperatures Transient for Loss of Offsite Power [2]

C09

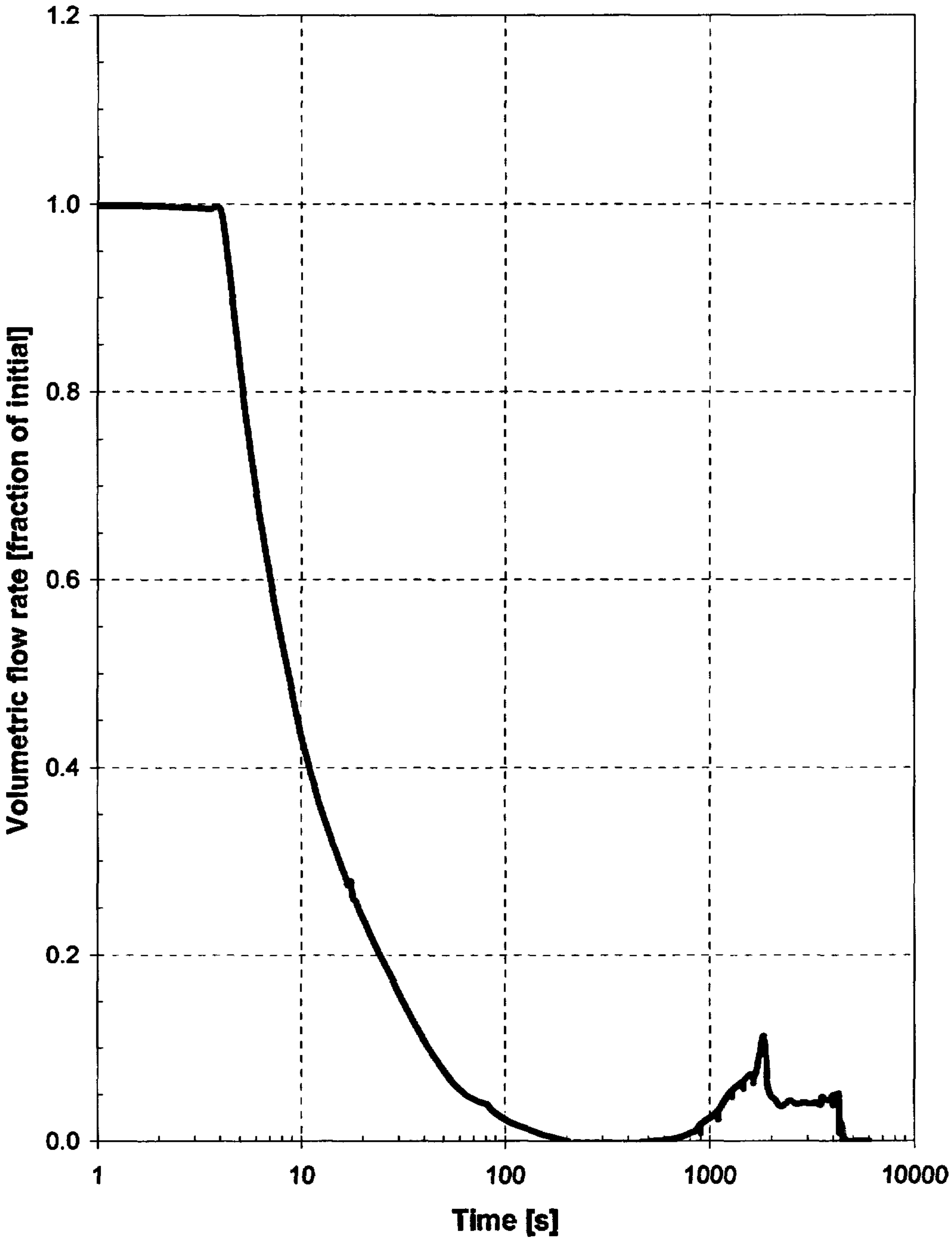


Figure 2.2.2-7
RCP Volumetric Flow Rate Transient for Loss of Offsite Power

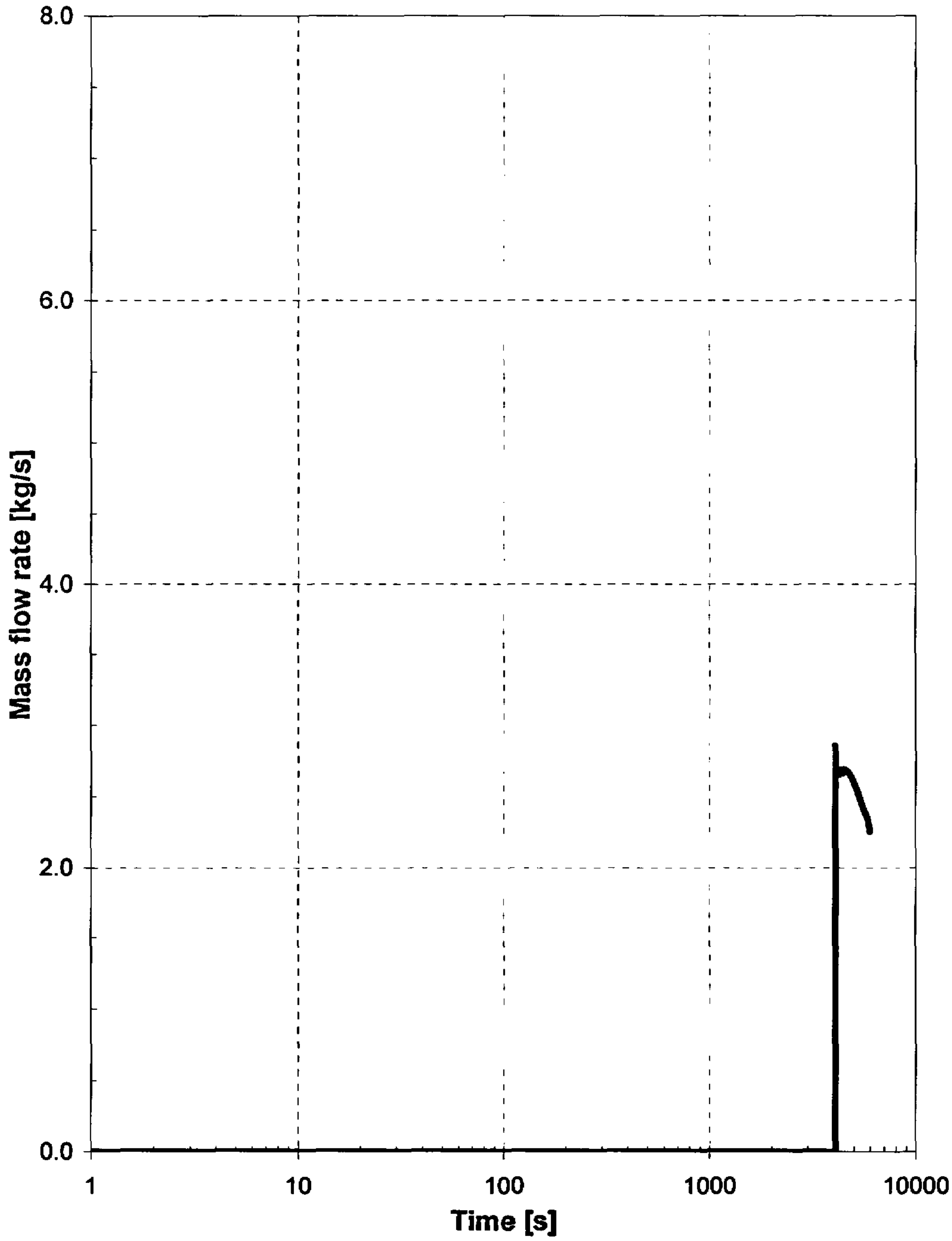


Figure 2.2.2-8
Emergency Boration Tank Flow Rate Transient for Loss of Offsite Power

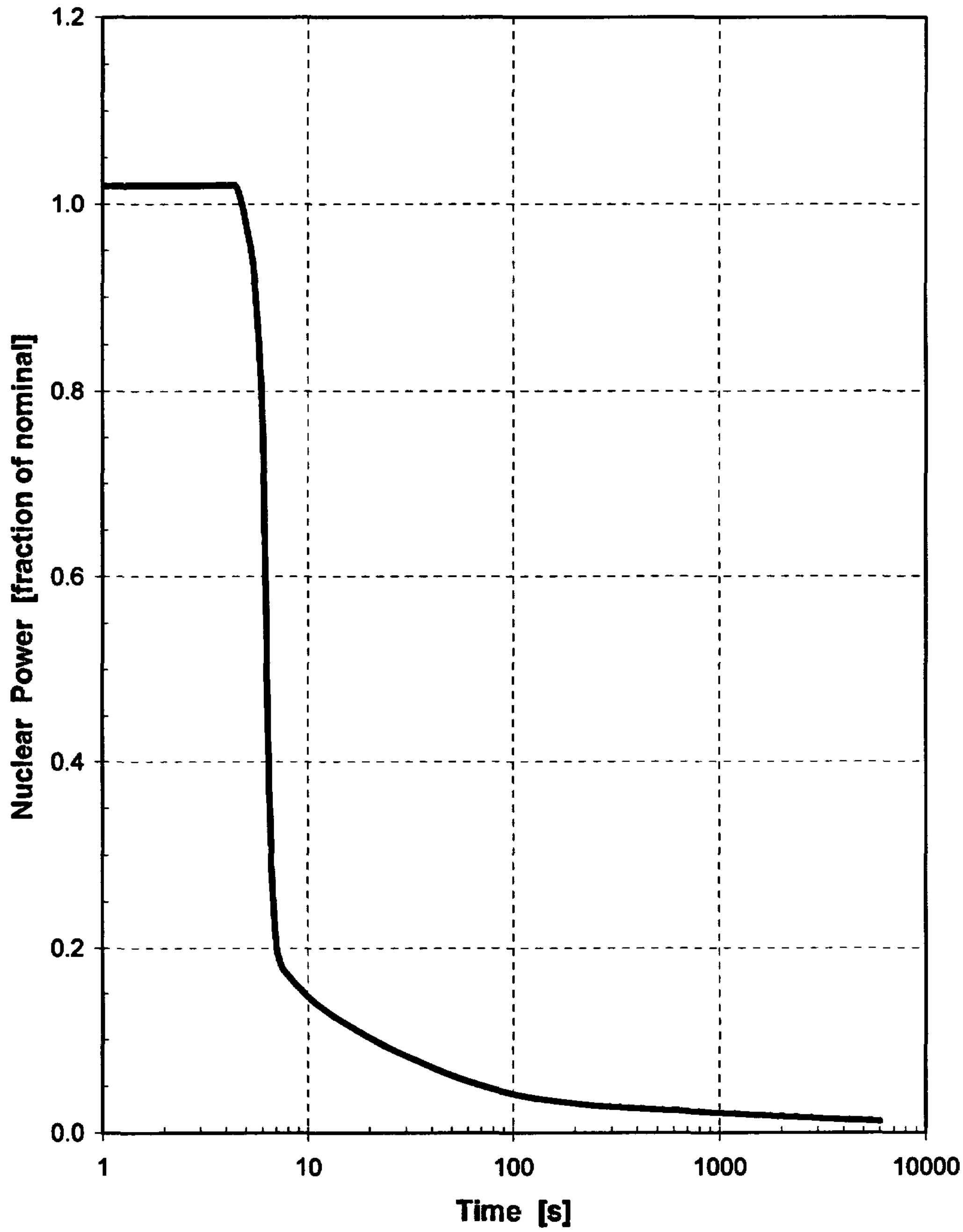


Figure 2.2.2-9
Nuclear Power Transient for Loss of Normal Feedwater

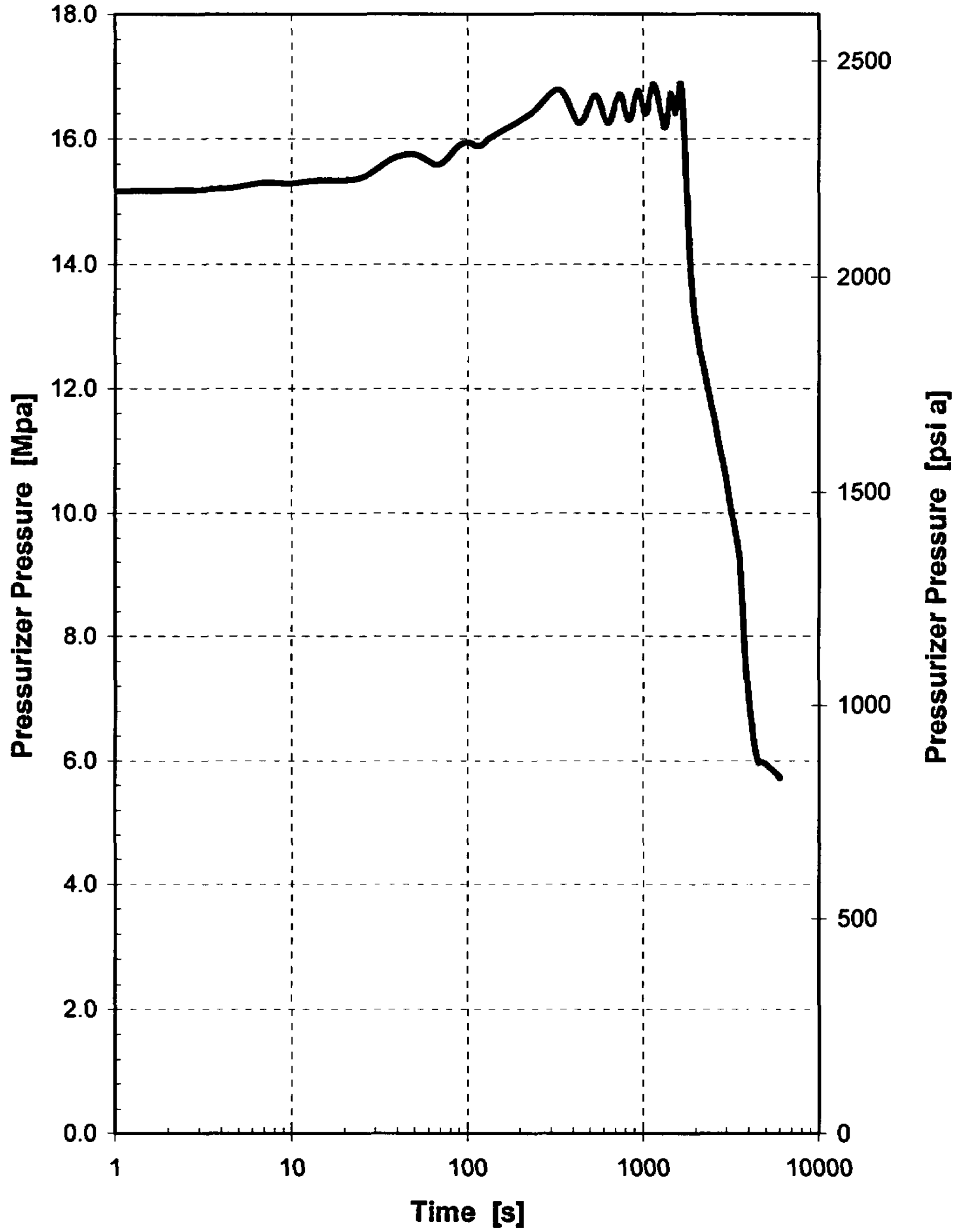


Figure 2.2.2-10
Pressurizer Pressure Transient for Loss of Normal Feedwater

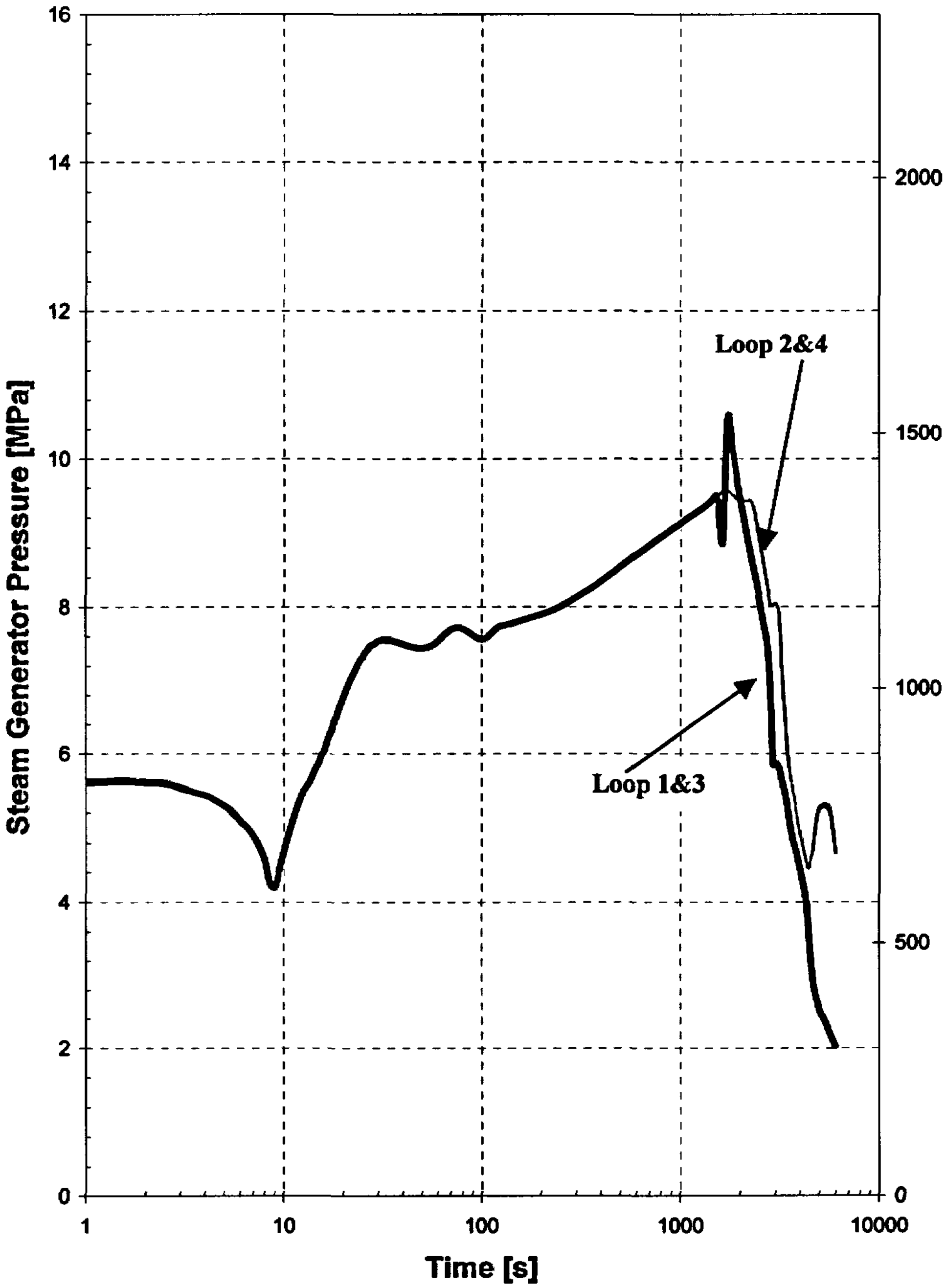


Figure 2.2.2-11
Steam Generator Pressure Transient for Loss of Normal Feedwater

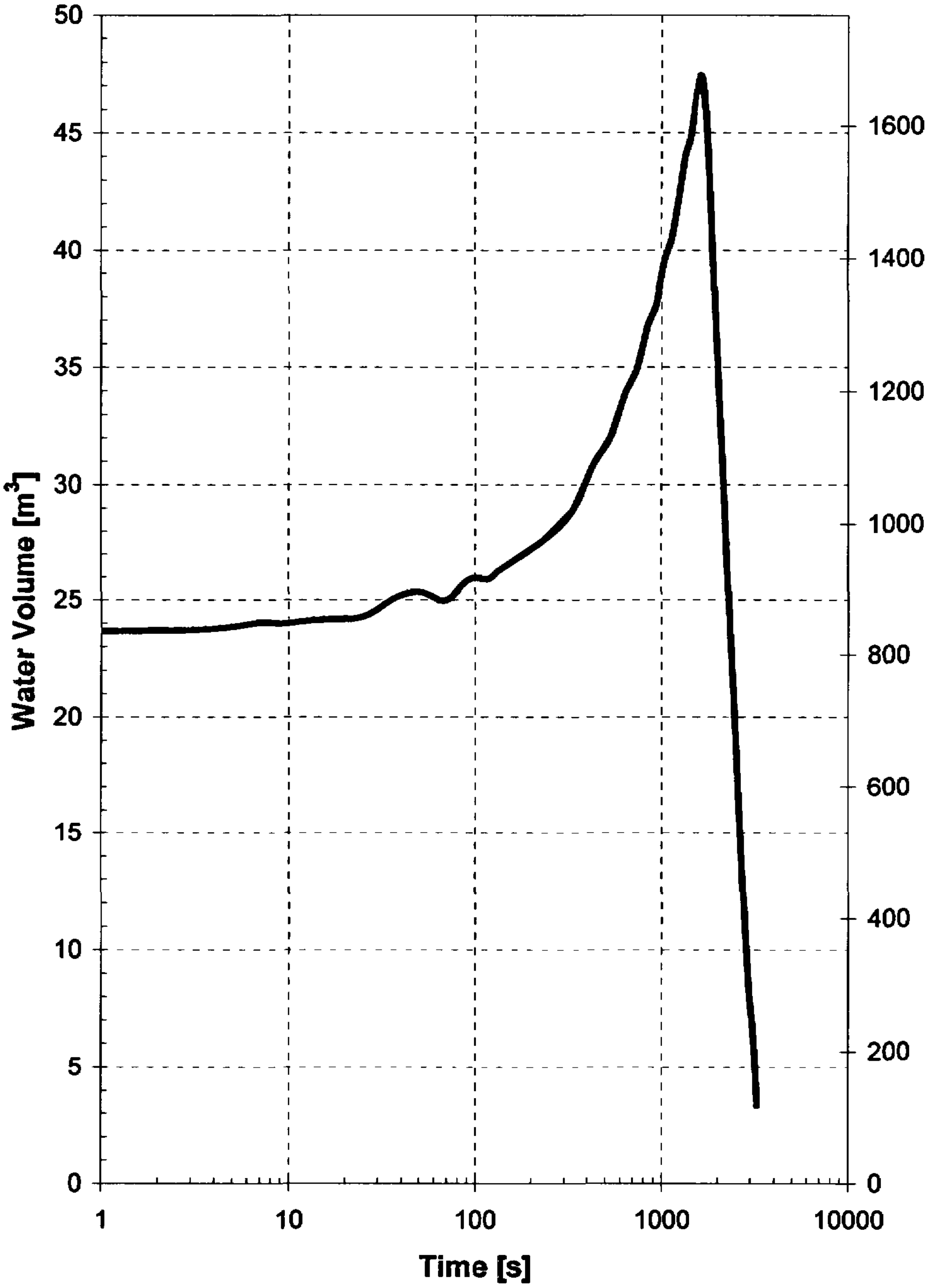
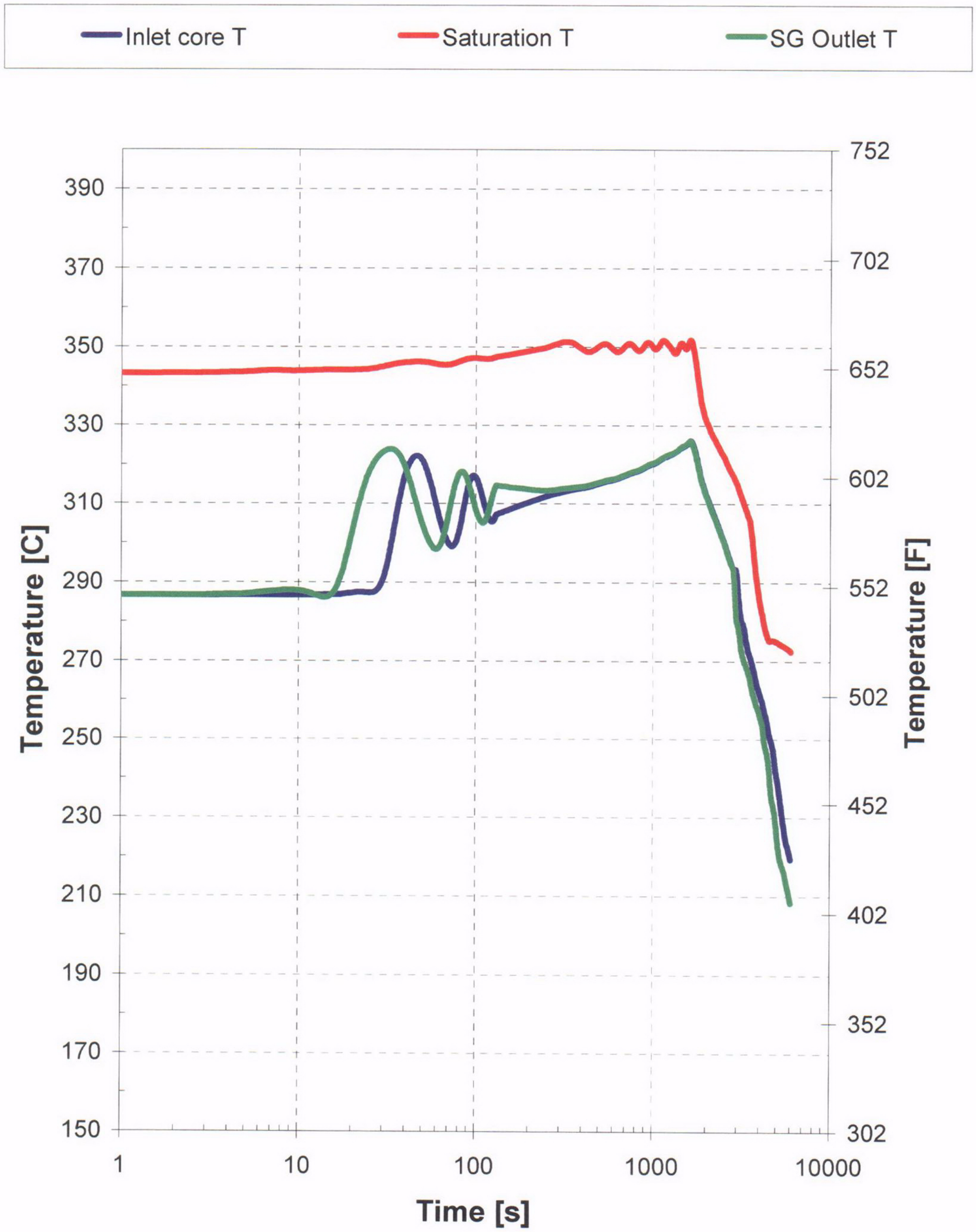


Figure 2.2.2-12
Pressurizer Water Volume Transient for Loss of Normal Feedwater
(Total Pressurizer Volume 2500 ft3)



2.2.2-13
RCS Temperatures Transient for Loss of Normal Feedwater [1]

C10

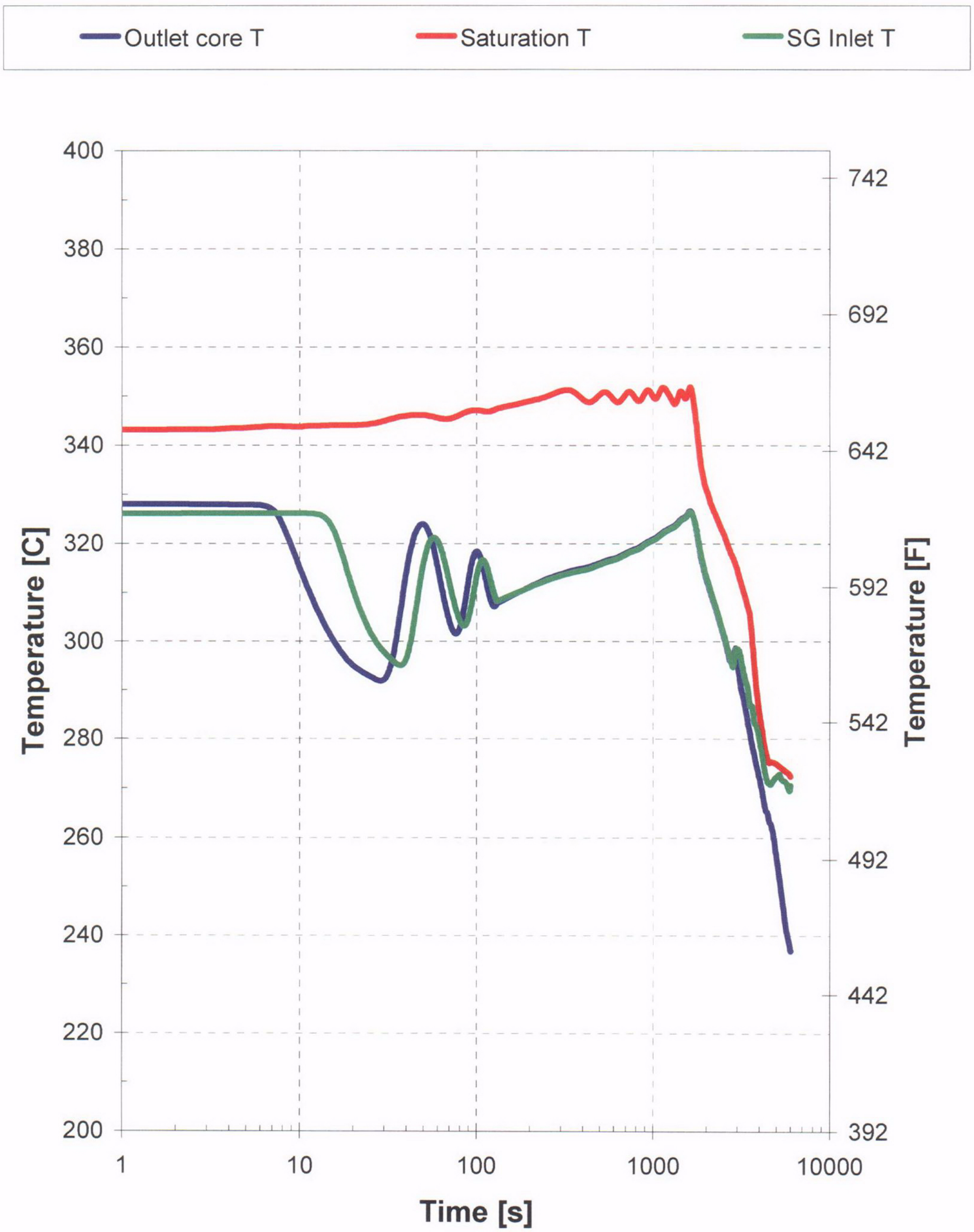


Figure 2.2.2-14
RCS Temperatures Transient for Loss of Normal Feedwater [2]

C11

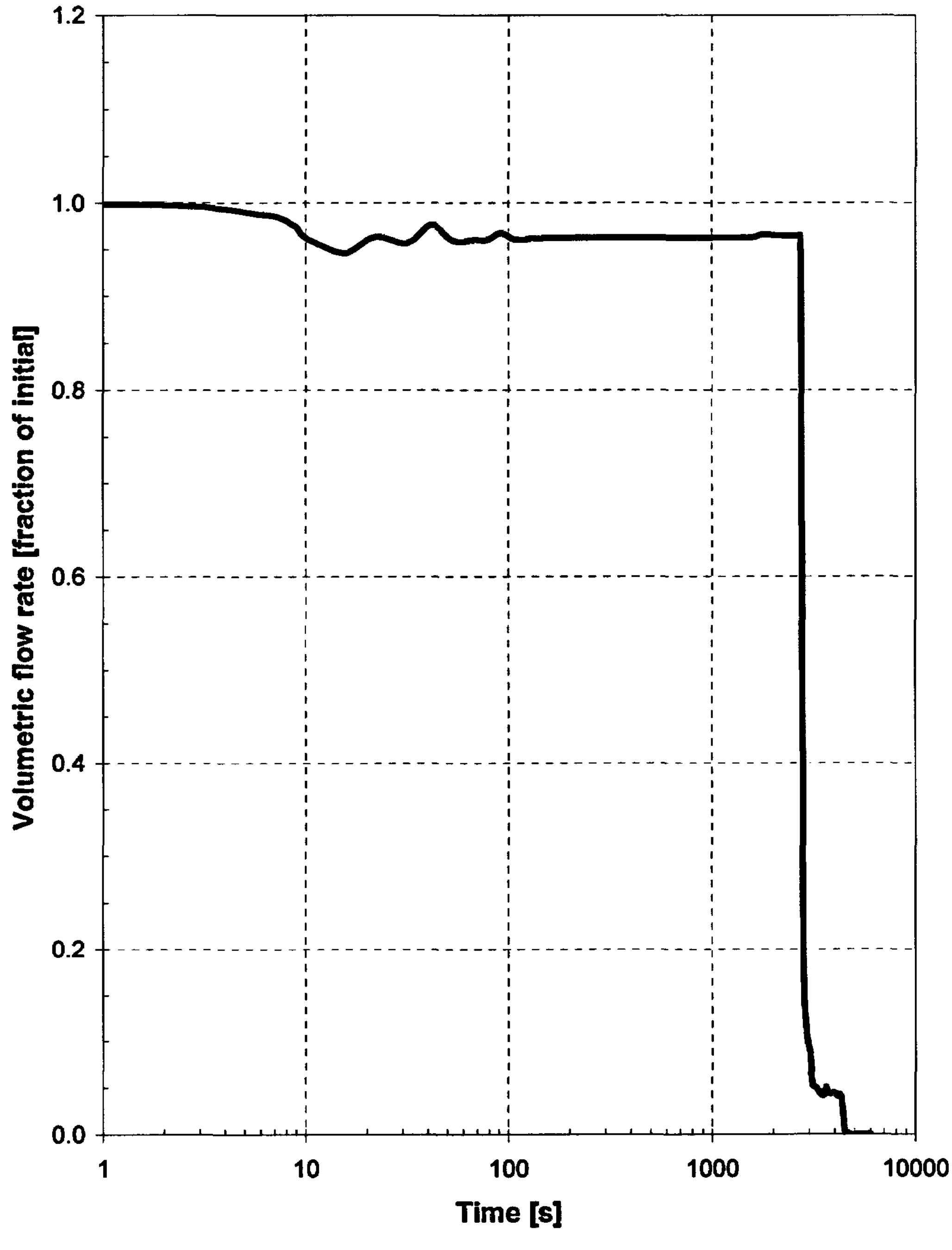


Figure 2.2.2-15
RCP Volumetric Flow Rate Transient for Loss of Normal Feedwater

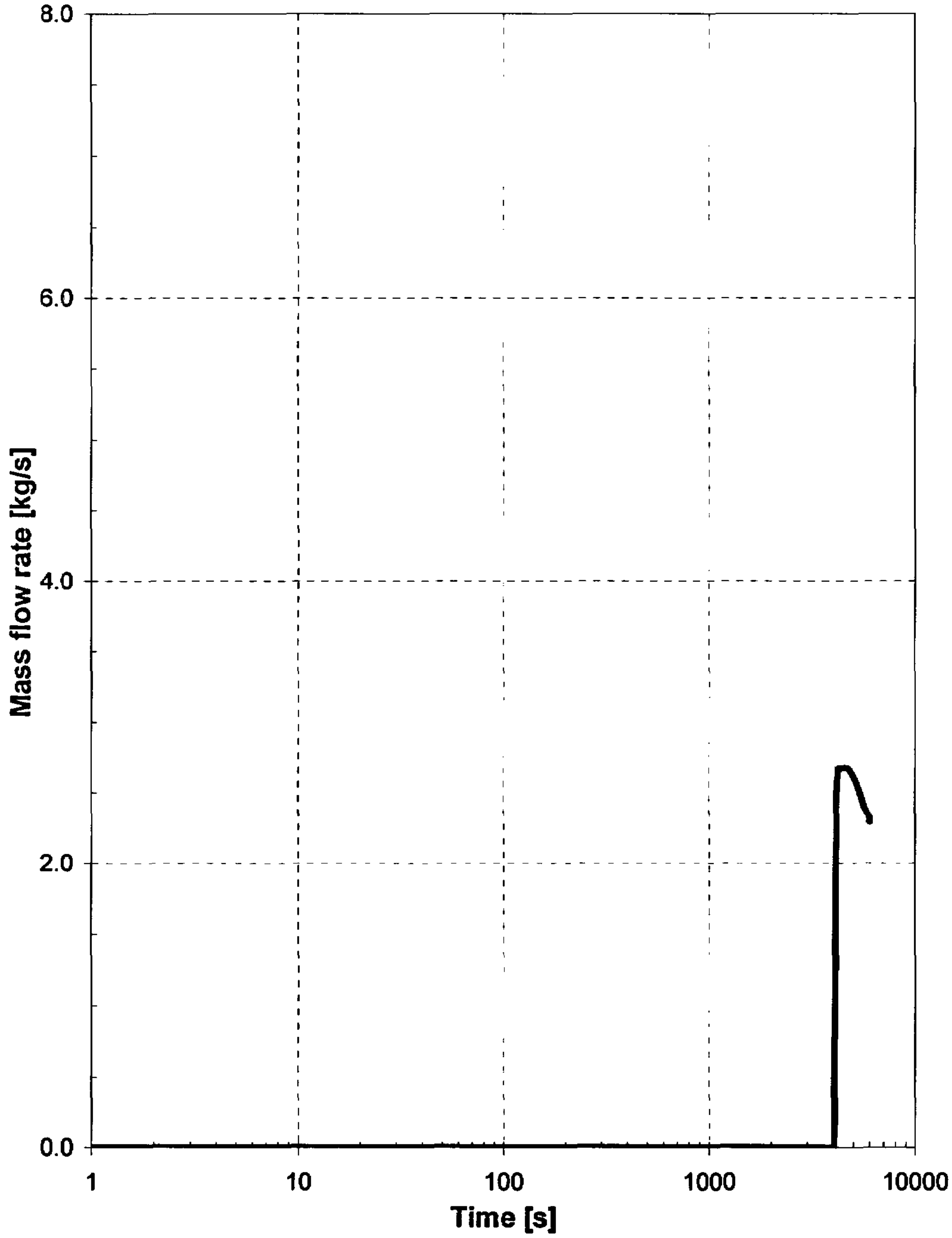


Figure 2.2.2-16
Emergency Boration Tank Flow Rate Transient for Loss of Normal Feedwater

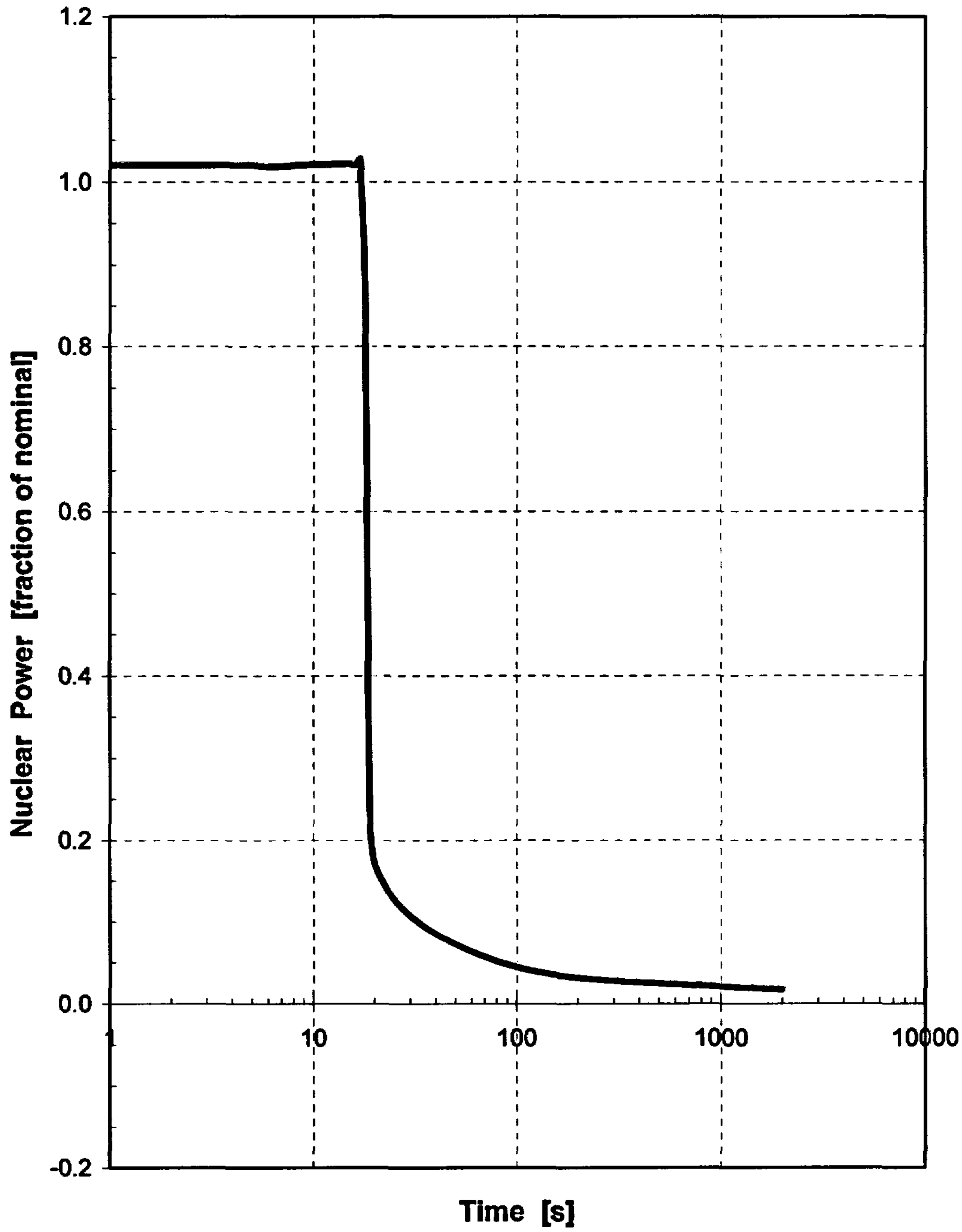


Figure 2.2.3-1
Nuclear Power Transient for Feedwater System Piping Failure

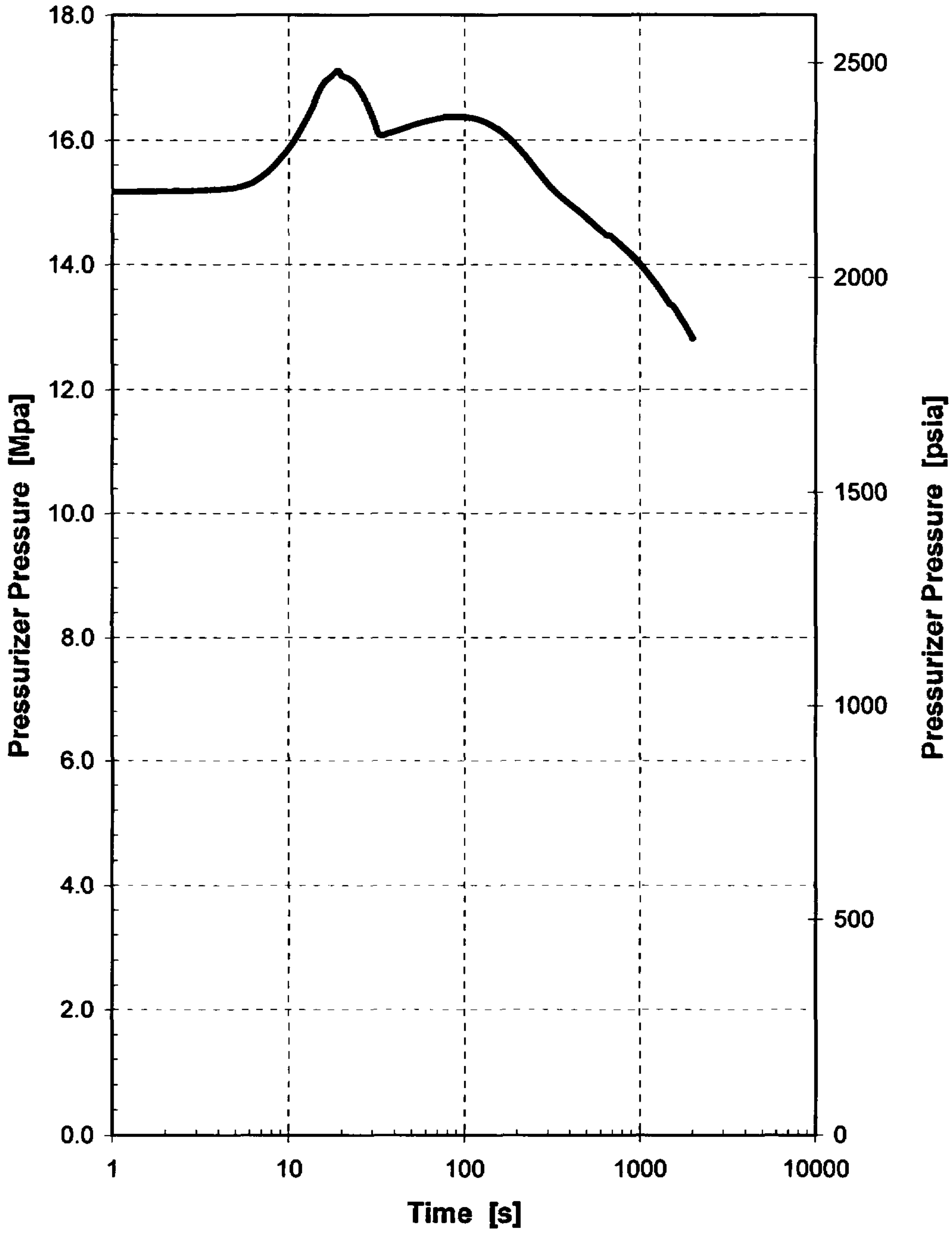


Figure 2.2.3-2
Pressurizer Pressure Transient for Feedwater System Piping Failure

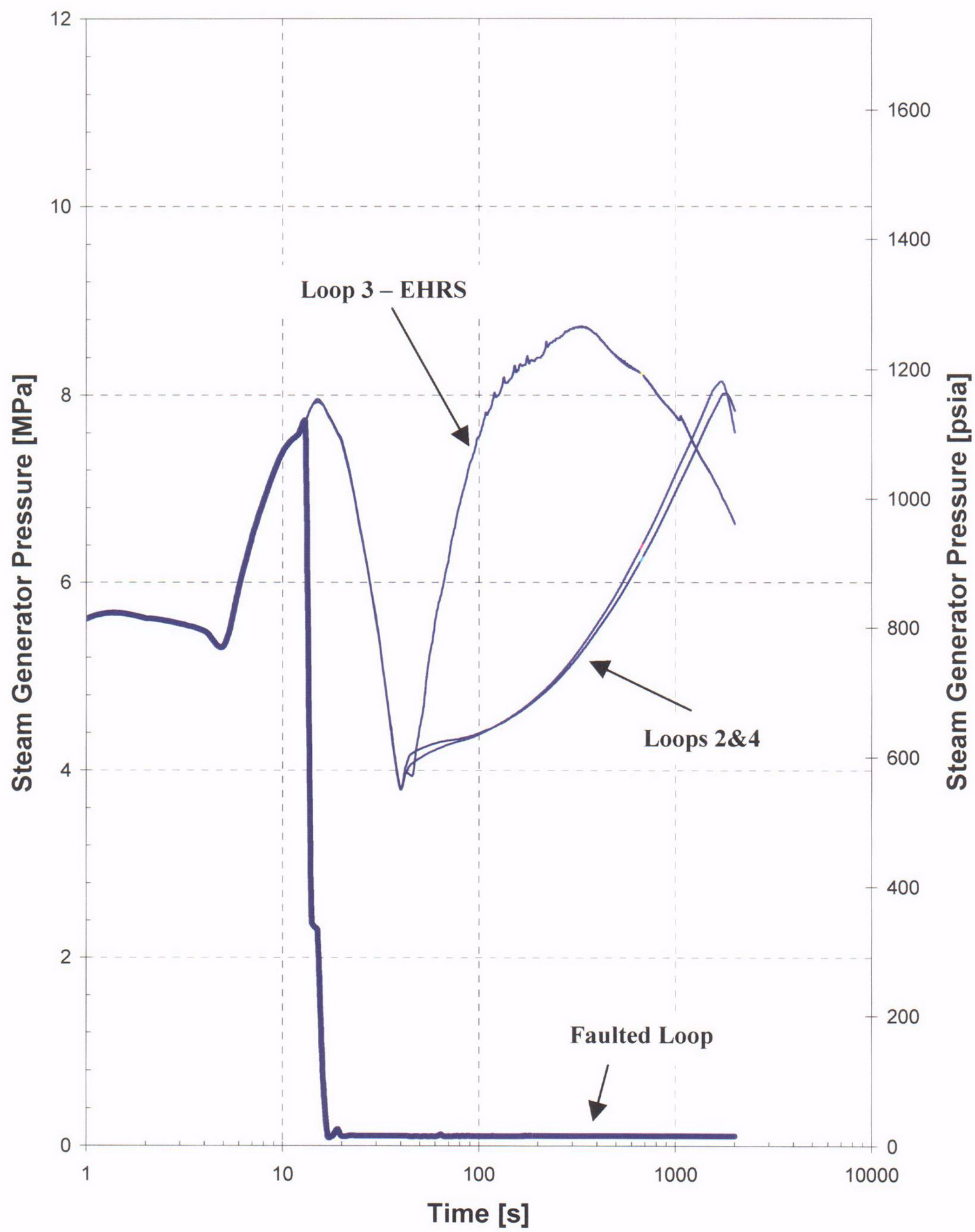


Figure 2.2.3-3
Steam Generator Pressure Transient for Feedwater System Piping Failure

C11A

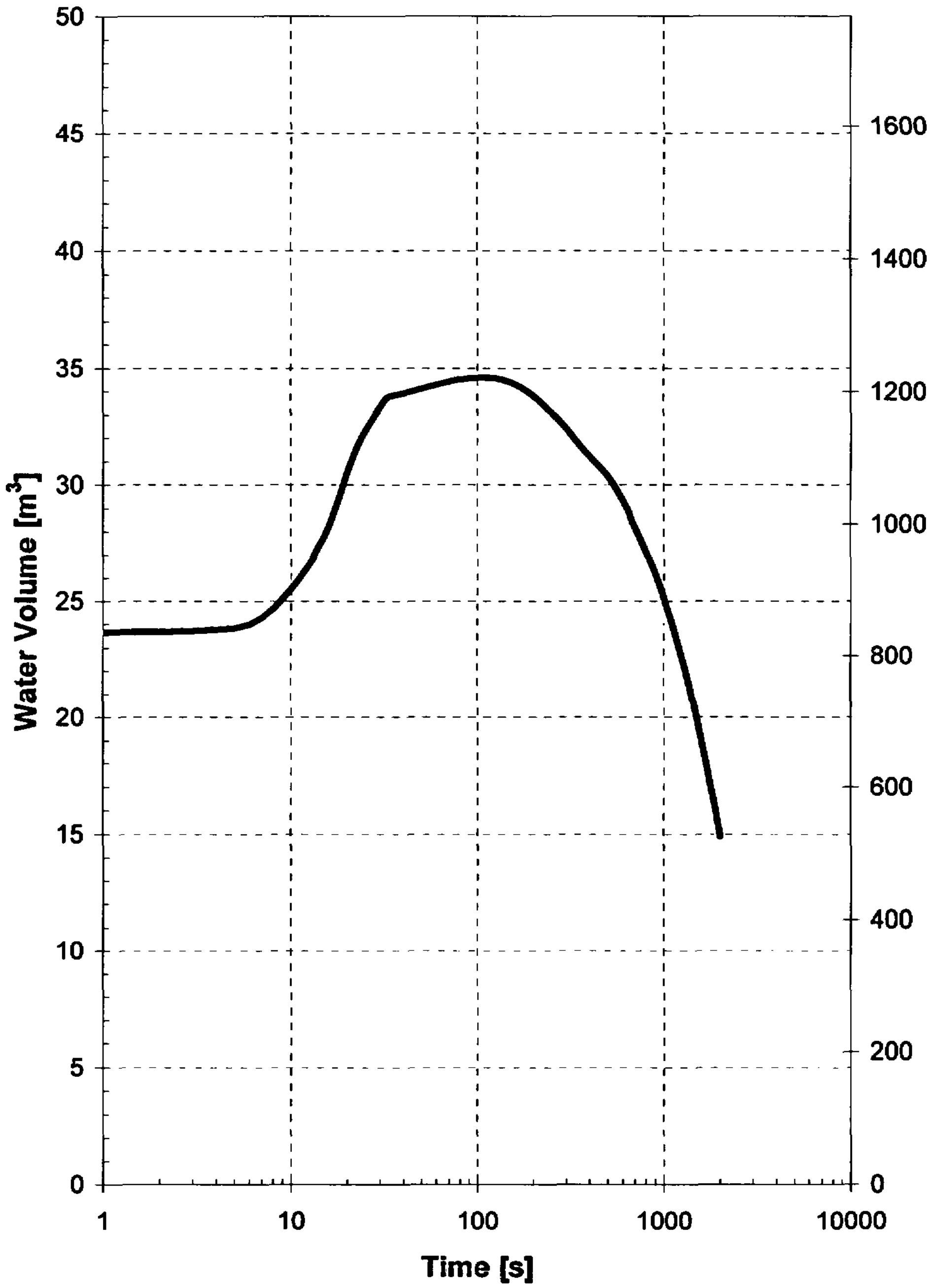


Figure 2.2.3-4
Pressurizer Water Volume Transient for Feedwater System Piping Failure (Total Pressurizer Volume 2500 ft3)

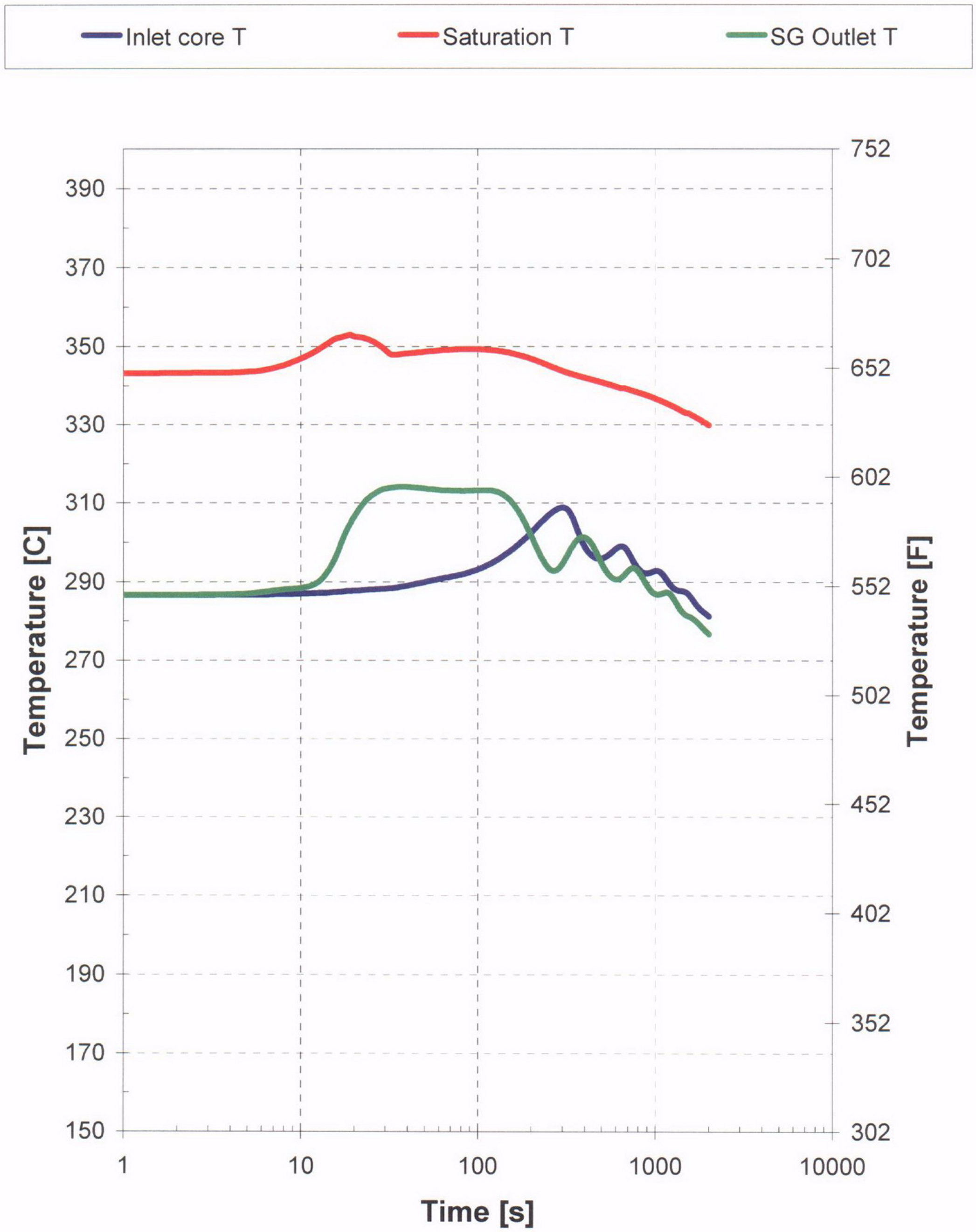


Figure 2.2.3-5
RCS Temperatures Transient for for Feedwater System Piping Failure [1]

C12

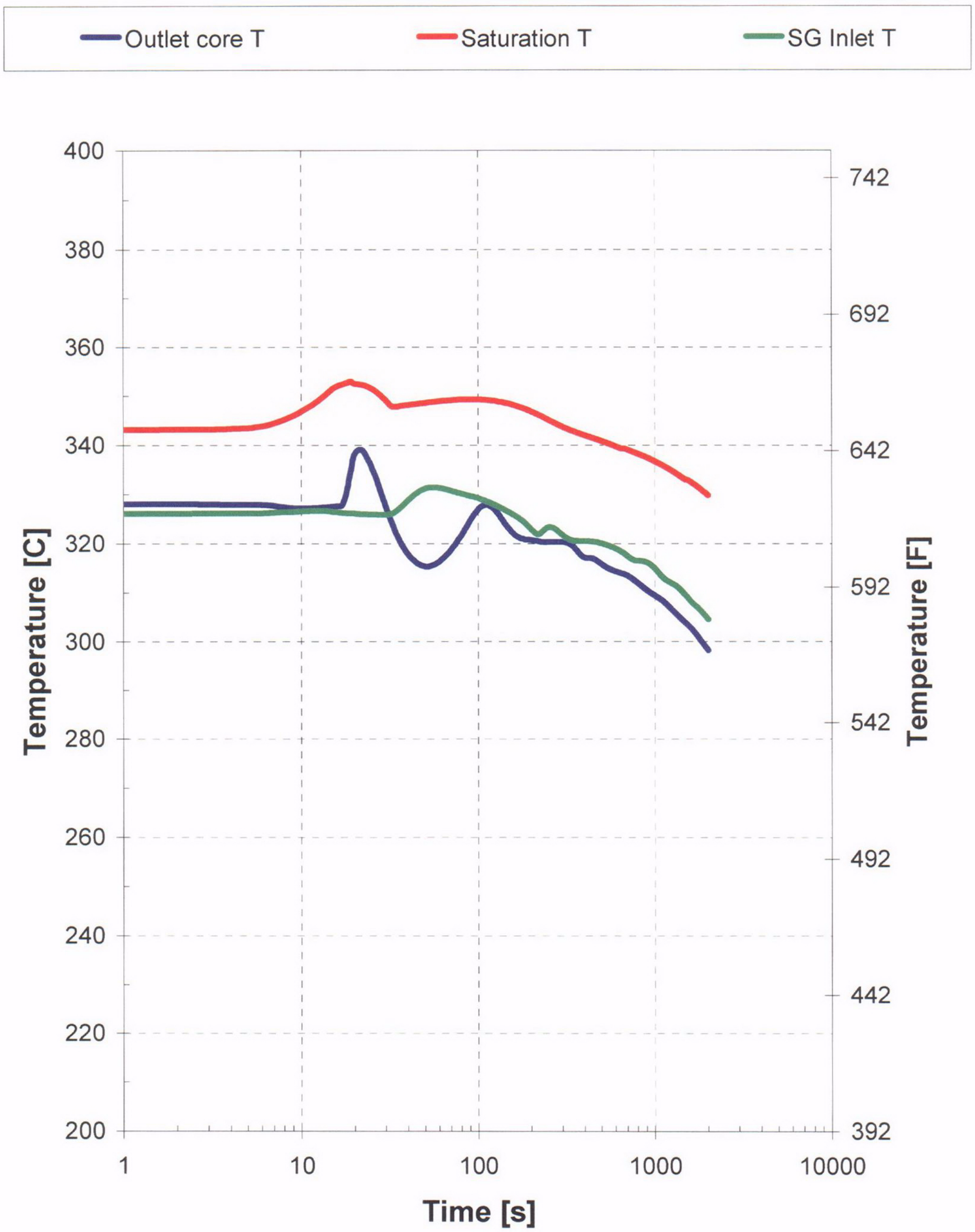


Figure 2.2.3-6
RCS Temperatures Transient for Feedwater System Piping Failure [2]

C13

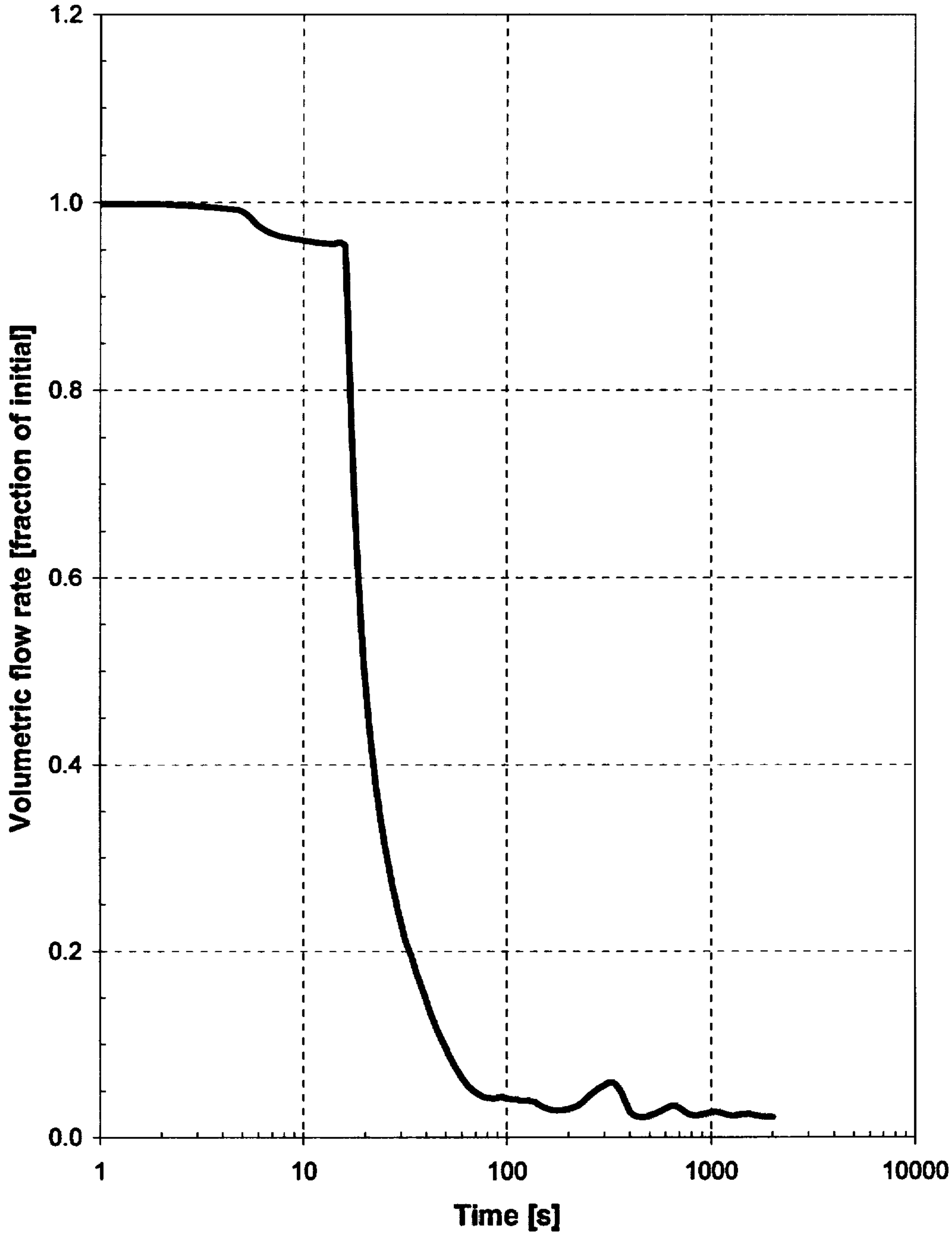


Figure 2.2.3-7
RCP Volumetric Flow Rate Transient for Feedwater System Piping Failure

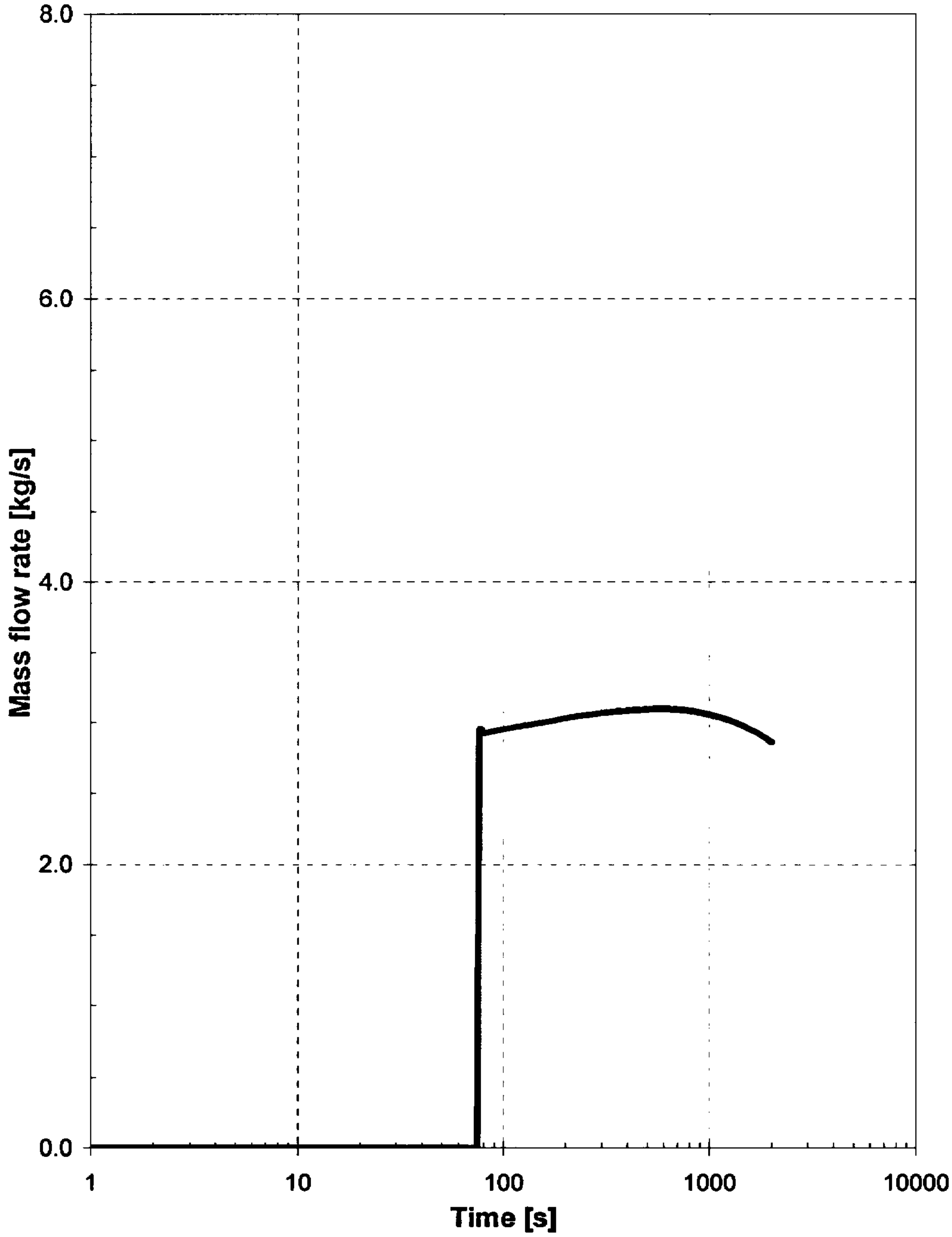


Figure 2.2.3-8
Emergency Boration Tank Flow Rate Transient for Feedwater System Piping Failure

2.3 Decrease in Reactor Coolant System Flow

Discussed in this Section are events that could result in a decrease in reactor coolant system flow rate.

Analyses will be presented in Section 15.3 of the IRIS Design Control Document for the following events that are identified as more limiting:

A. Partial loss of forced reactor coolant flow (15.3.1)

B. Complete loss of forced reactor coolant flow (15.3.2)

C. Reactor coolant pump shaft seizure (locked rotor) (15.3.3)

D. Reactor coolant pump shaft break (15.3.4)

Item A above is considered to be an ANS/ANSI Condition II event, Item B an ANS/ANSI Condition III event, and Items C and D are ANS/ANSI Condition IV events. Section 2.0.1 contains a discussion of ANS/ANSI classifications.

In this document, specific consideration is provided for the following events,

(1) Complete loss of forced reactor coolant flow, Section 2.3.2

(2) Reactor coolant pump shaft seizure (locked rotor), Section 2.3.3

Also, some qualitative considerations are provided in Section 2.3.1 for a partial loss of forced reactor coolant flow.

The events in this category present the potential for a sudden reduction in the heat transfer rate in the core following a loss of reactor coolant. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly. Protection against these events is provided by a Reactor Trip before fuel damage can occur.

IRIS does not present significant phenomenological differences in the evolution of these events compared to other PWRs, and to AP600/AP1000 in particular. Aside from other specific differences that will be discussed in the following section, the main design difference from other PWRs is in the response to a Locked Rotor event: due to the increased number of reactor coolant pumps in IRIS (8 versus 4 for a typical 4-Loop plant or for AP600/AP1000) the loss of forced flow from a single pump leads to a milder scenario. The analyses will demonstrate that the same acceptance criteria used for the Complete loss of flow event (i.e., a 95% probability with a 95% confidence than departure from nucleate boiling does not occur) can be satisfied for the Condition IV locked rotor event.

2.3.1 Partial Loss of Forced Reactor Coolant Flow

A partial loss-of-coolant flow accident can result from a mechanical or electrical failure in an RCP or from a fault in the power supply to the pump or pumps. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly.

Normal power for the pumps is supplied through individual buses connected to the generator. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core.

A partial loss of forced reactor coolant flow is classified as an ANS/ANSI Condition II incident (an incident of moderate frequency) as defined in Section 2.0.1.

In Westinghouse loop PWRs, protection for a partial loss-of-coolant flow accident is provided by the low primary coolant flow reactor trip which is actuated by two out of three low flow signals in any reactor coolant loop. Above permissive P8, low flow in any loop will actuate a reactor trip. Between approximately 10-percent power (permissive P10) and the power level corresponding to permissive P8, low flow in any two loops will actuate a reactor trip.

IRIS is not a loop design, and therefore a protection similar to that given by the reactor trip on Low Primary Coolant Flow Rate must be defined. This is still an open design issue: one alternative would envision just an overall measure of the reactor coolant system flow rate, for example based on core DP, while other alternatives are being evaluated to identify the possibility of measuring flow for each of the eight RCP/SG modules. The design will take advantage of the flexibility provided by the large thermal margin available. Since the final design choice will influence the response to the partial loss of flow event, the analysis of this event has been postponed. Also, it should be noted that while the current design point assumes a reactor trip on loss of one pump, an alternative that is being considered to provide additional operation flexibility is to allow operation at a reduced power level with one or two pumps out of service.

As specified in GDC 17 of 10 CFR Part 50, Appendix A, the effects of a loss of offsite power are considered when evaluating partial loss of coolant flow rate transients. As discussed in Section 2.0.14, the loss of offsite power is considered to be a potential consequence of the event due to disruption of the grid following a turbine trip following the event. The primary effect of the loss of offsite power is to cause the remaining operating reactor coolant pumps to coast down.

2.3.2 Complete Loss of Forced Reactor Coolant Flow

2.3.2.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of

the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature.

Electrical power for the reactor coolant pumps is supplied through buses, connected to the generator through the unit auxiliary transformer. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core.

A complete loss of flow accident is classified as an ANS/ANSI Condition III incident (an infrequent fault) as defined in Section 2.0.1.

The following signals provide protection for a complete loss of flow accident:

A. Reactor coolant pump power supply undervoltage (or underfrequency or underspeed).

B. Low reactor coolant flow

The reactor trip on RCP undervoltage is provided to protect against conditions that can cause a loss of voltage to all RCPs, i.e., loss of offsite power. This function is blocked below approximately 10 percent power (permissive P10).

The reactor trip on RCP underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid.

The reactor trip on low primary coolant flow is provided to protect against loss of flow conditions that affect only some reactor coolant pump (see discussion on the partial loss of flow, Section 2.3.1).

2.3.2.2 Method of Analysis

The complete loss of flow is analyzed for a loss of power to all eight reactor coolant pumps.

This transient has been analyzed in this study using two computer codes. First, the RELAP5 code (see Section 2.0.11) has been used to evaluate the reactor coolant system transient following a loss of power to all eight reactor coolant pumps. The program simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, steam generators. The program computes pertinent plant variables including temperatures, pressures, and power level.

The VIPRE code (see Section 2.0.11) is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the nuclear power and core flow from RELAP.
The major assumptions for the case analyzed, a complete loss of voltage followed by the reactor coolant pumps coasting down, are as follows.

- **Initial Operating Conditions** - Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Plant characteristics and initial condition

are discussed in Section 2.0.3. Uncertainties in initial conditions are included in the limit DNBR as described in Section 2.0.11

- **Reactivity Coefficients** - A conservatively large absolute value of the Doppler-only power coefficient is used (see Table 2.0-5). This is equivalent to a total integrated Doppler Reactivity from 0- to 100-percent of 0.016 Δk. The least negative moderator temperature coefficient is assumed because this results in the maximum core power during the initial part of the transient, when the minimum DNBR is reached. For this analysis, a curve of trip reactivity versus time based on a 2.5-second rod cluster control assembly insertion time to the dashpot is used (see Figure 2.0.5-3).

- **Flow Coastdown** - For the IRIS spool pump an Inertia corresponding to a [ ]$^{(a, c)}$ without added flywheel have been calculated. This relatively high inertia is characteristic of the spool pump design. [ ]$^{(a, c)}$

- **Reactor Trip.** Reactor trip is actuated by the first reactor protection system trip setpoint reached, which is expected to be the reactor coolant pump undervoltage.

Note that a loss of forced primary coolant flow can also result from a reduction in the reactor coolant pump motor supply frequency. The results of the complete loss of voltage, followed by the reactor coolant pumps coasting down, bound the complete loss of flow initiated by a frequency decay of up to 5 hertz per second. Therefore, only the results of the complete loss of voltage case are presented.

The present analysis is limited by the following assumption: [

]$^{(a, c)}$

---

[1] [

]$^{(a, c)}$

2.3.2.3 Results

The calculated sequence of events for the case analyzed is shown on table 2.3-1. The RCPs will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in Section 2.2.2. With the reactor tripped, a stable plant condition is attained. Normal plant shutdown may then proceed.

Figures 2.3.2-1 through 2.3.2-5 show the transient response for the loss of power to all RCPs. All figures include a 1 second null transient. The reactor is assumed to be tripped on an undervoltage signal. Figure 2.3.2-5 shows the DNBR to be always greater than the limit value.

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperature do not approach any limiting conditions.

2.3.2.4 Conclusions

For the complete loss of forced reactor coolant flow, the DNBR does not decrease below the limit value at any time during the transient; thus, no fuel or clad damage is predicted, and all applicable acceptance criteria will be met.

2.3.3 Reactor Coolant Pump Rotor Seizure (Locked Rotor)

2.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of an RCP rotor. Flow through the affected pump and associated steam generator is rapidly reduced, leading to initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the tube side of the steam generators is reduced, initially because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the shell side cools down while the tube side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, caused by the flow reduction, combined with reduced heat transfer in the steam generators, results in an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, but is not expected to lead to an actuation of the pressurizer safety valves.

This event is classified as an ANS/ANSI Condition IV incident (a limiting fault) as defined in Section 2.0-1.

From a phenomenological point of view, the evolution of this event in IRIS does not present significant differences from current loop PWRs, and AP600/AP1000 in particular. However, two design features have to be considered when evaluating the system response:

1) IRIS has a larger number of reactor coolant pumps (eight versus four for 4-Loop Plants and AP600/AP1000). This leads to a reduced transient following a locked rotor on a single pump. While the phenomenology is similar to other PWRs, the severity of the system response is greatly mitigated by this inherent design feature. The analysis developed here indicate that the more stringent acceptance criteria specified for Condition II events can be met for this event: it should be noted that the preliminary evaluation model used in this analysis presents sufficient differences, due to the different acceptance criteria, from conventional PWRs that results should only be considered as indicative of the large margin available to achieve this goal (i.e. satisfy acceptance criteria for Condition II events). Additionally, the analyses performed show that the DNB safety limit is not violated following a locked rotor event even if no reactor trip signal is generated.

2) As discussed in Section 2.3.1, the definition of the instrumentation and logic for the low flow trip setpoint is still being assessed for IRIS. In the analysis, it is conservatively assumed that only an overall reactor coolant system flow measure is provided. This delays the reactor trip on a low flow signal, and is therefore is a conservative assumption for this event.

2.3.3.2 Method of Analysis

This transient has been analyzed by two digital computer codes. First the RELAP5 code (see Section 2.0.11) was used to calculate the flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code (see Section 2.0.11) was then used to calculate the departure from nucleate boiling ratio (DNBR) during the transient based on the RELAP5 system analysis.

At the beginning of the postulated locked rotor accident (i.e., at the time the rotor in one of the RCPs is assumed to seize) the plant is assumed to be in operation under the most adverse steady-state operating conditions (i.e., maximum steady-state power level, maximum steady-state pressure, and maximum steady state coolant average temperature). Plant characteristics and initial conditions are further discussed in Section 2.0.3. The accident is evaluated for both cases with and without offsite power available. For the case without offsite power available, the power is conservatively assumed to be lost 3.0 seconds following a turbine/generator trip. Turbine Trip is assumed to occur at the time of reactor trip.

Typically, two evaluations are provided for this event: a case for which assumptions that maximize the pressure peak during the transient are made, and a case for which the assumptions minimize the MDNBR during the transient. In this analysis, only the DNB case is analyzed. The reason is that, based also on the results provided in Section 2.3.2, it is not expected that any significant pressurization will follow this event. Due to the large steam volume in the IRIS pressurizer, the initial pressurization shall be mitigated, and only decay heat removal will be require to terminate the reactor coolant system pressurization. The analyses provided for the turbine trip and for the loss of normal feedwater (Section 2.2.1 and 2.2.2) is expected to be limiting compared to this event. A more complete analyses will be developed as part of the design certification, but these considerations are sufficient for the scope of this safety assessment.

This event is therefore analyzed here only to demonstrate that DNB will not occur with a 95% probability at a 95% confidence. This analysis will be performed using the revised thermal design procedure (RTDP, see Section 2.0.11).

Assumptions similar to those discussed in Section 2.3.2 for the complete loss of flow events have been made in the evaluation of the locked rotor events.

2.3.3.3 Results

Figures 2.3.3-1 to 2.3.3-5 show the transient results for one locked rotor with eight reactor coolant pumps in operation. Only the results for the case without offsite power available are provided. A time sequence for the event is provided in Table 2.3-1. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed (See Section 2.2).

The analyses demonstrate the large margin available in IRIS following a locked rotor event, and provide adequate assurance that the minimum DNB ratio will remain above the safety analysis limit during the transient, and therefore the 95% probability at a 95% confidence that DNB is not reached in the most limiting rod is guaranteed.

More detailed analyses and a more complete development of an adequate evaluation model for this event will be provided as part of the design certification effort.

2.3.3.4 Conclusions

The analysis performed and the qualitative considerations discussed, demonstrate that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the limit value at any time during the transient; thus, no fuel or clad damage is predicted, and all applicable acceptance criteria will be met.

Table 2.3-1
TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW

(time calculated from beginning of transient)

Accident	Event	Time (sec)
Partial loss of forced reactor coolant flow (2.3.1)	Coast down of one pump begins	1.0
	Low flow reactor trip setpoint reached in affected loop	TBD
	Rods begin to fall into core	TBD
	Minimum DNBR occurs	TBD
Complete loss of forced reactor coolant flow (2.3.2)	All operating pumps lose power and begin coasting down; reactor coolant pump undervoltage setpoint reached	1.00
	Rods begin to fall into core	2.50
	Minimum DNBR occurs	4.35
Reactor coolant pump shaft seizure (locked rotor/broken shaft) (2.3.3)	Rotor in one pump locks/breaks	1.00
	Low flow reactor trip setpoint reached	1.20
	Rods begin to fall into core	2.65
	Minimum DNB occurs	3.15
	Loss of offsite power, unaffected reactor coolant pumps begin to coast down	5.34

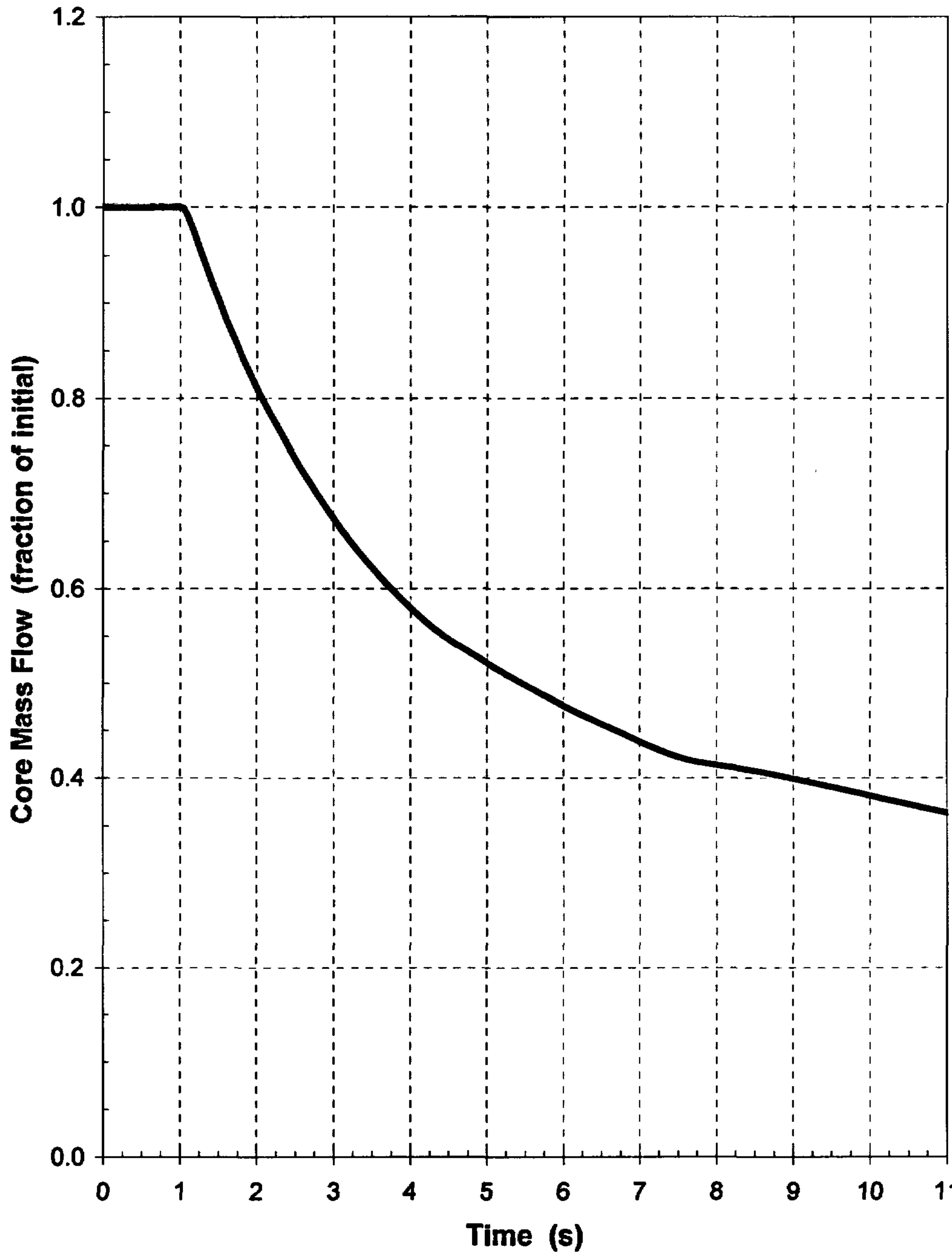


Figure 2.3.2-1
Core Inlet Mass Flow Rate Transient for Complete Loss of Flow Event

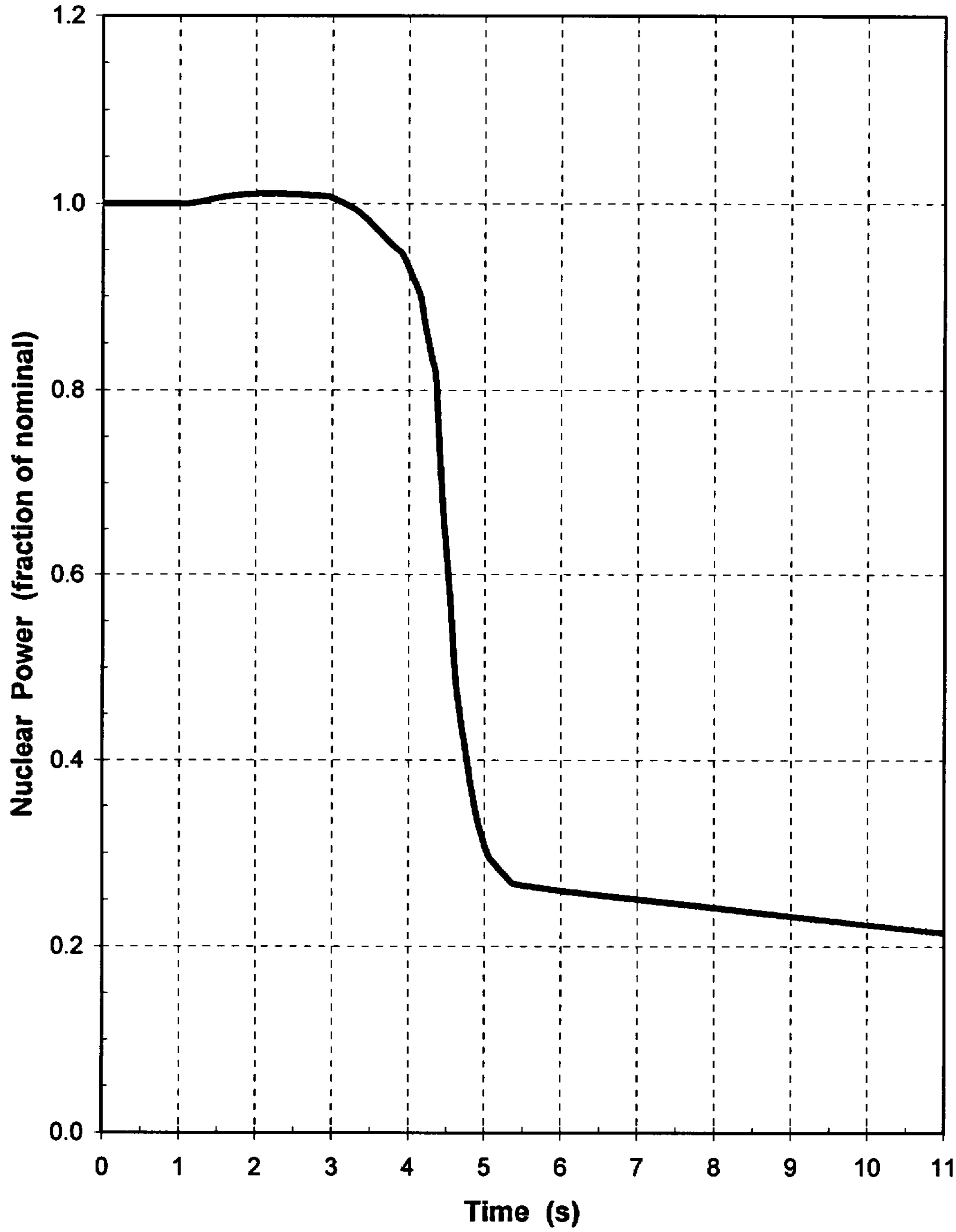


Figure 2.3.2-2
Nuclear Power Transient for Complete Loss of Flow

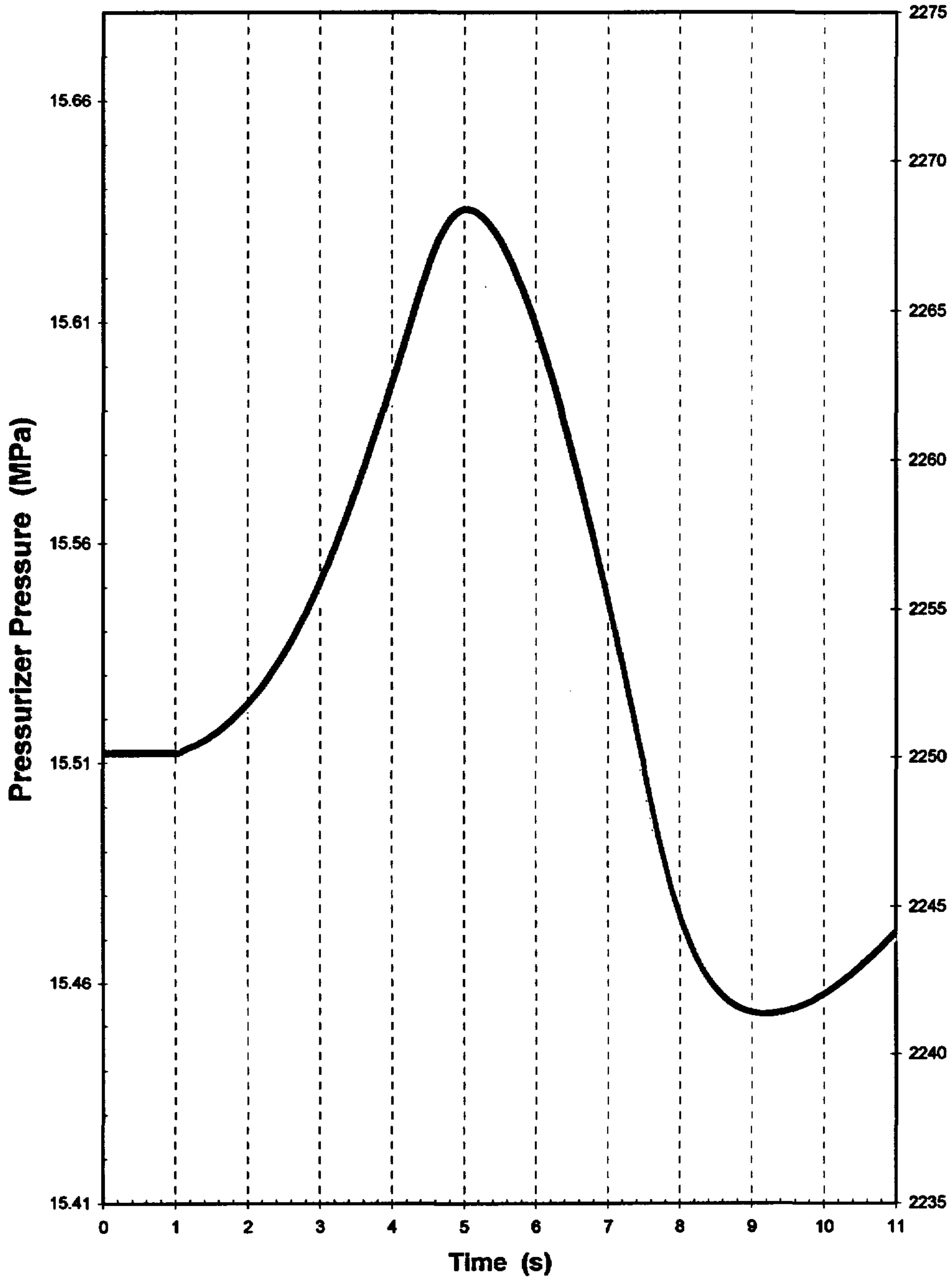


Figure 2.3.2-3
Pressurizer Pressure Transient for Complete Loss of Flow

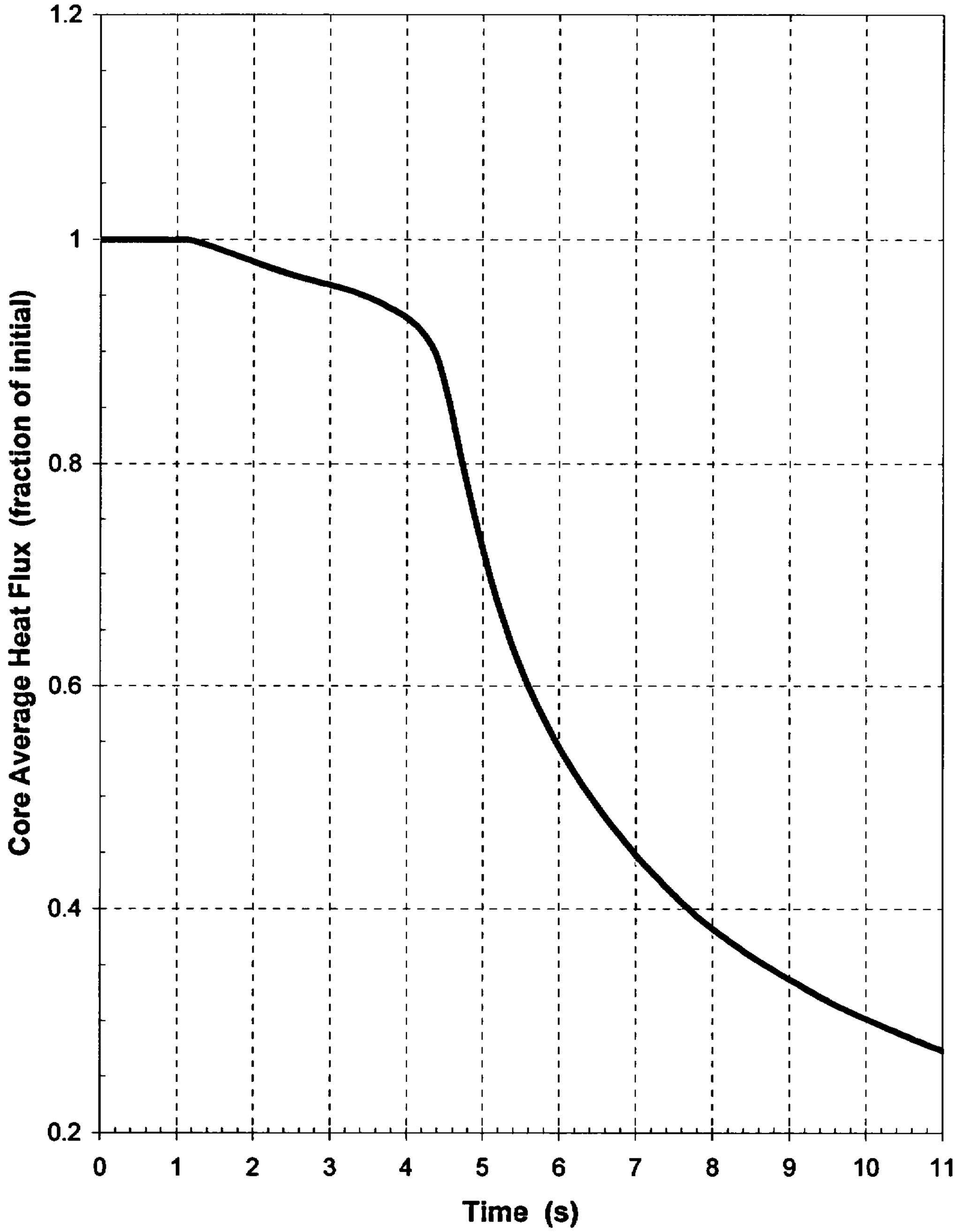


Figure 2.3.2-4
Core Average Heat Flux Transient for Complete Loss of Flow

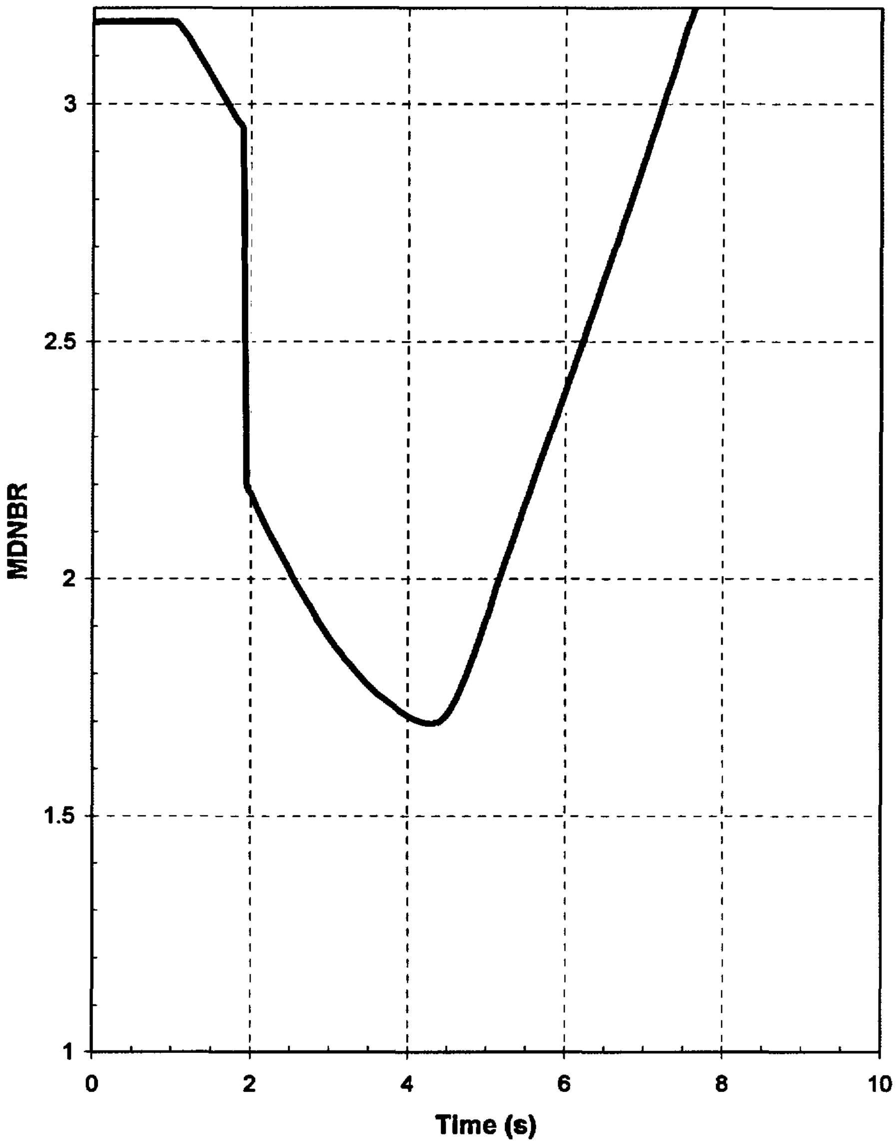


Figure 2.3.2-5
MDNBR for Complete Loss of Flow

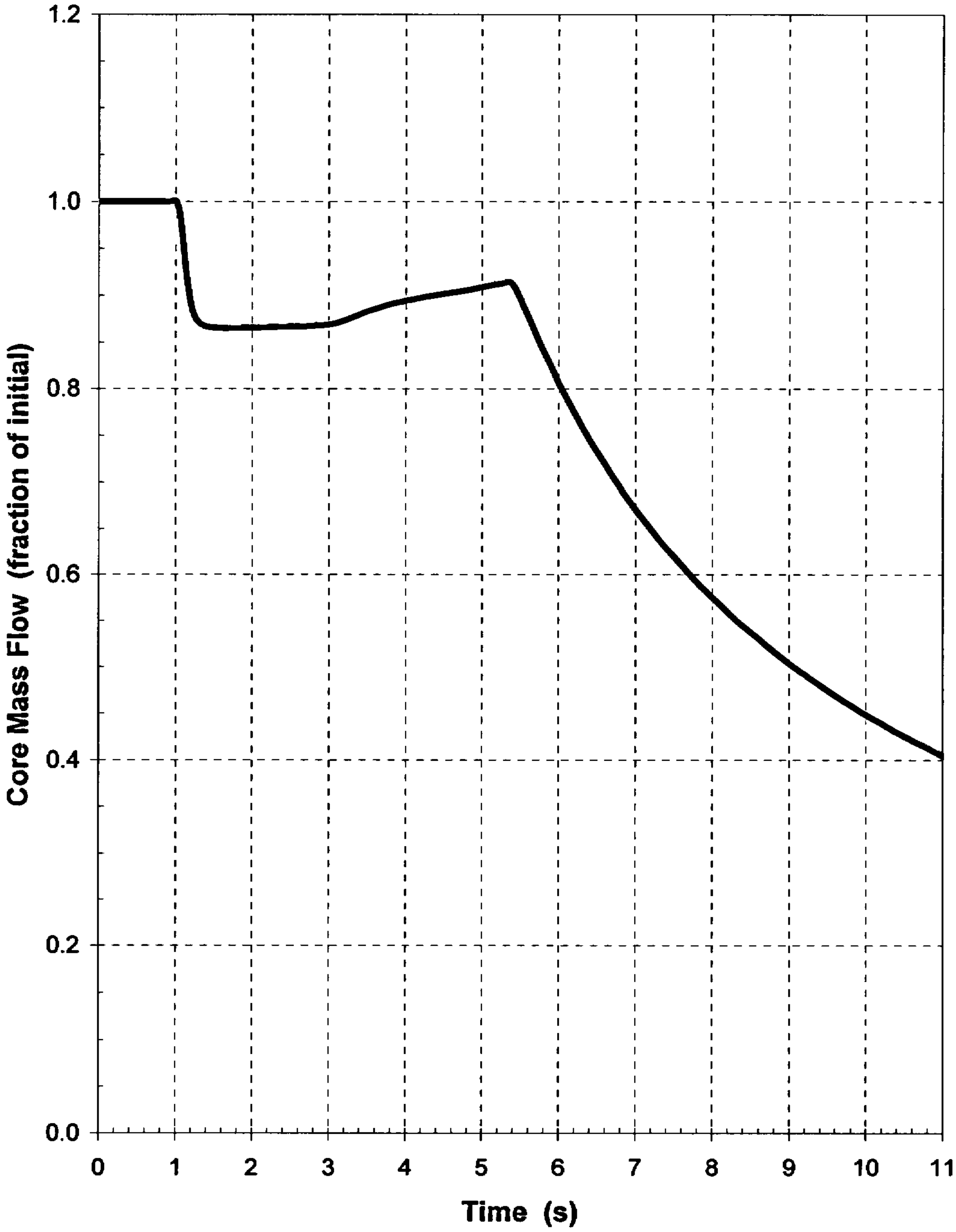


Figure 2.3.3-1
Core Inlet Mass Flow Rate Transient for Locked Rotor Event

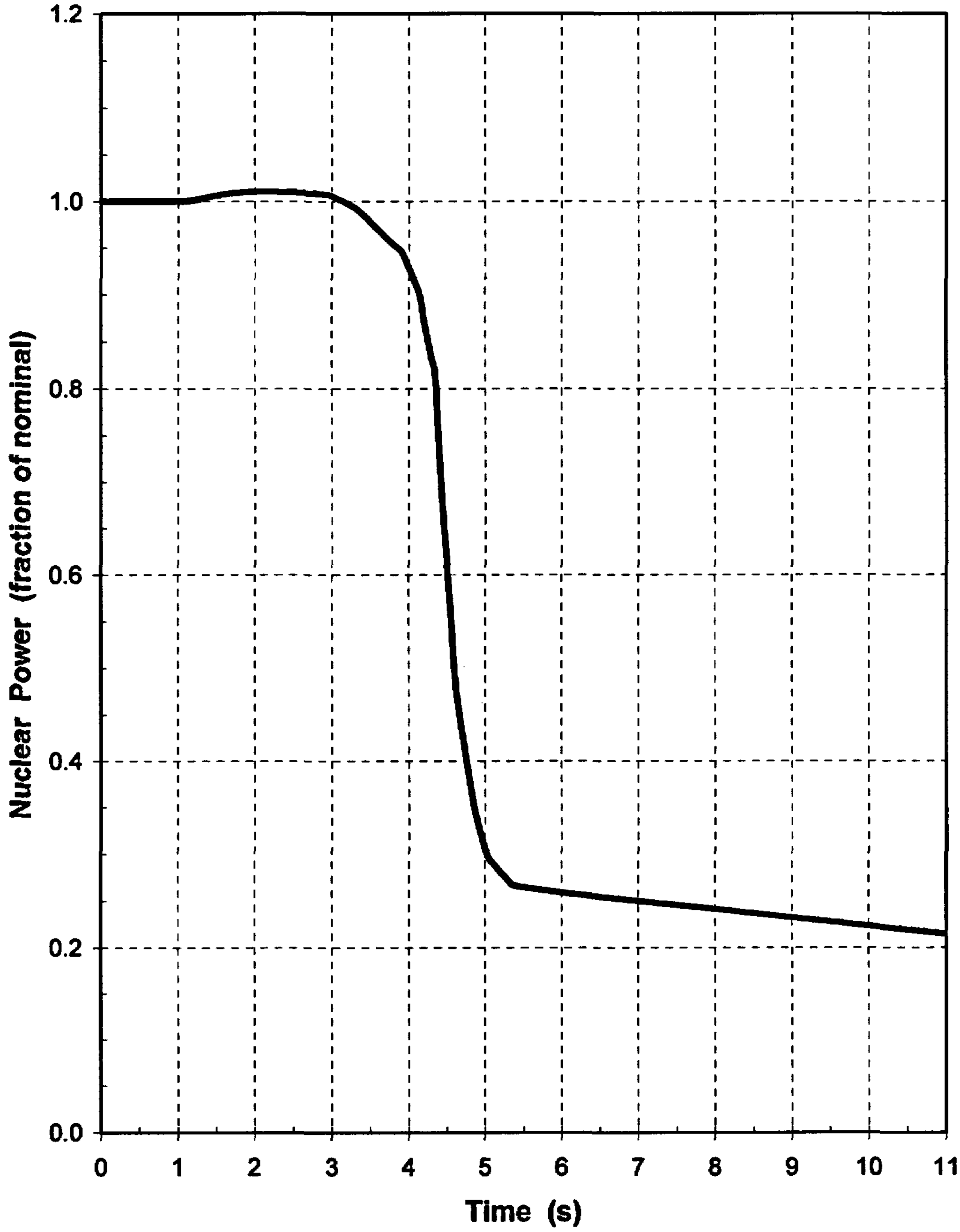


Figure 2.3.3-2
Nuclear Power Transient for Locked Rotor

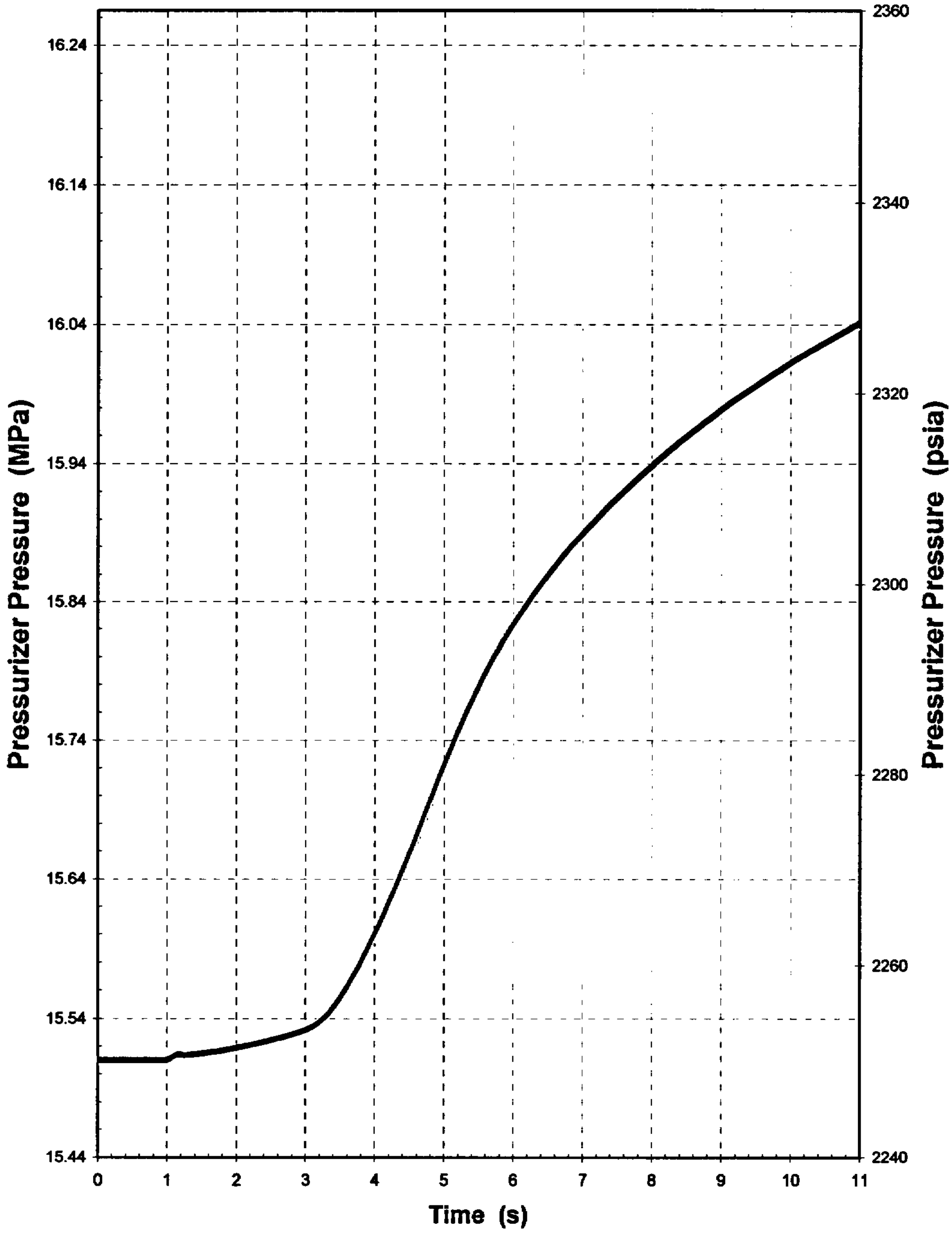


Figure 2.3.3-3
Pressurizer Pressure Transient for Locked Rotor

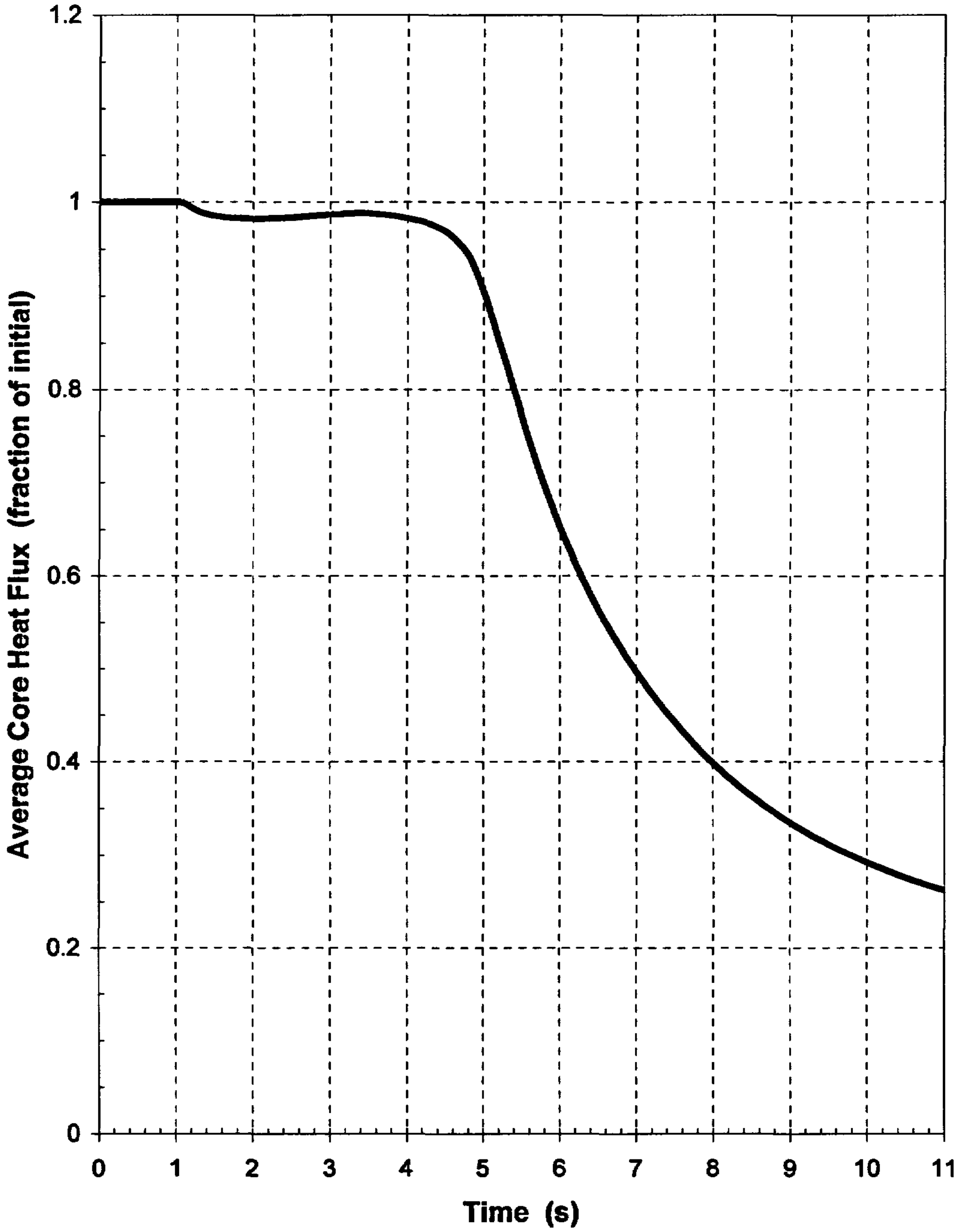


Figure 2.3.3-4
Average Core Heat Flux Transient for Locked Rotor

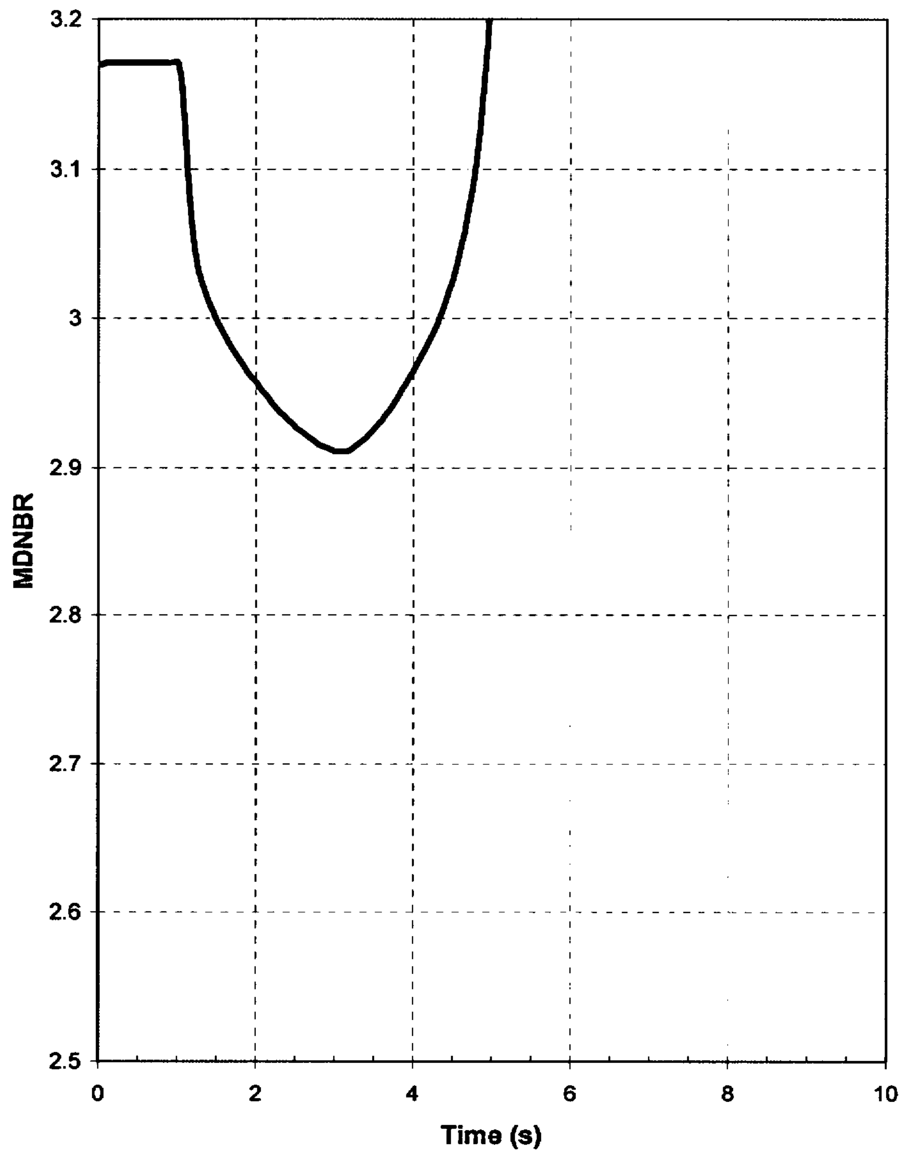


Figure 2.3.3-5
MDNBR Transient for Locked Rotor

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Westinghouse Electric Company, LLC
2000 Day Hill Road
Windsor, Connecticut 06095-0500