DESIGN CONTROL DOCUMENT FOR THE
US-APWR
Chapter 15
Transient and Accident Analyses

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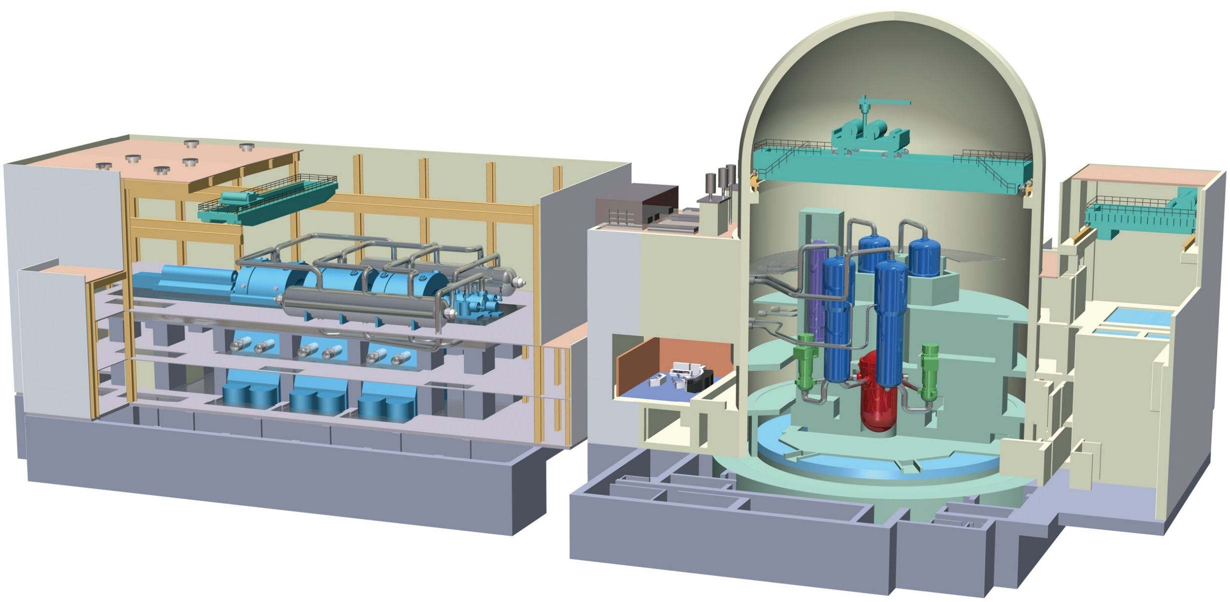

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APPENDIX 15A EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

TABLES

FIGURES

ACRONYMS

ac	alternating current
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
APWR	advanced pressurized water reactor
ASME	American Society of Mechanical Engineers
AST	alternative source term
ATWS	anticipated transient without scram
BOL	beginning of life
BOC	beginning-of-cycle
BOC	beginning-of-life
C/V	containment vessel
CCW	component cooling water
CD	coefficient of discharge
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
COLA	Combined License Application
COLR	Core Operating Limits Report
CRDM	control rod drive mechanism
CRE	control room envelope
CS/RHR	containment spray/residual heat removal
CSS	containment spray system
CVCS	chemical and volume control system
CWO	core wide cladding oxidation
DAS	Diverse Actuation System
DBA	design-basis accident
dc	direct current
DCD	Design Control Document
DDE	deep dose equivalent
DE	dose equivalent
DECLG	double-ended cold leg guillotine
DF	decontamination factor
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio

DV	depressurization valve
DVI	direct vessel injection
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDE	effective dose equivalent
EFW	emergency feedwater
EFWS	emergency feedwater system
EOC	end-of-cycle
EOP	emergency operating procedure
EPS	emergency power source
ESF	engineered safety features
EZB	exclusion zone boundary
GDC	General Design Criteria
GTG	gas turbine generator
HFP	hot full power
HHIS	high head injection system
HEPA	high-efficiency particulate air
HVAC	heating ventilation and air conditioning
HZP	hot zero power
ICCC	incore control component
LMO	local maximum cladding oxidation
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPZ	low-population zone
LWR	light-water reactor
MCR	main control room
MHI	Mitsubishi Heavy Industries, Ltd.
MSDV	main steam depressurization valve
MSIV	main steam isolation valve
MSRV	main steam relief valve
MSSV	main steam safety valve
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PA	postulated accident
PCMI	pellet/cladding mechanical interaction
PCT	peak cladding temperature

PORV	power operated relief valve
PRT	pressurizer relief tank
PSMS	protection and safety monitoring system
PWR	pressurized-water reactor
QA	quality assurance
RADTRAD	radionuclide transport, removal, and dose
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RCPB	reactor coolant pressure boundary
RG	Regulatory Guide
RHR	residual heat removal
RTDP	revised thermal design procedure
RTP	rated thermal power
RTS	reactor trip system
RV	reactor vessel
RWSP	refueling water storage pit
SAFDL	specified acceptable fuel design limits
SDM	shutdown margin
SDV	safety depressurization valve
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SIS	safety injection system
SRP	Standard Review Plan
SRV	safety relief valve
TBV	turbine bypass valve
TEDE	total effective dose equivalent
TMI	Three Mile Island
TSC	technical support center
UPS	uninterruptible power supply
VCT	volume control tank

15.0 Transient and Accident Analyses

The accident analysis in this Design Control Document (DCD) is organized consistent with the Standard Review Plan (SRP) NUREG-0800 and the most recent version of Regulatory Guide 1.206 (RG 1.206).

15.0.0.1 Classification of Plant Conditions

Initiating events are categorized by event type and by frequency of occurrence. Categorization by event type provides for logical comparison between events with similar effects on the plant, which allows for the identification of limiting events. Classification by frequency of occurrence provides a basis for the selection of applicable acceptance criteria.

Initiating events are first categorized by their effect on the plant (i.e., event type), such as an increase in heat removal by the secondary system, and then further classified according to their expected frequency of occurrence. Historically, the frequency of each event was categorized as an incident of moderate frequency (ANSI N18.2 Category II), an infrequent event (ANSI N18.2 Category III), or a limiting fault (ANSI N18.2 Category IV) (Ref. 15.0-1). However, the current SRP does not use the historical ANSI N18.2 frequency classification but rather classifies an event as either an anticipated operational occurrence (AOO) or a postulated accident (PA). This DCD utilizes the categorization and classification schemes adopted by the current SRP.

It is important to note that AOOs and PAs apply to certain initiating events, but that there are transients and accidents that are more severe and infrequent than AOOs, but not as severe and infrequent as PAs. Examples of these events include AOOs with an assumed coincident single failure or operator error, as well as infrequent events that can only result from coincident component active failures or passive failures. AOOs that occur with such a coincident failure are no longer considered AOOs. Such events are either evaluated as if they were AOOs or less restrictive acceptance criteria are applied.

Due to the similarities between the MHI US-APWR and the current generation of PWRs operating in the United States, MHI has determined that no new event types are required to bound the possible initiating events.

15.0.0.1.1 Normal Operation and Anticipated Operational Occurrences

Normal operation includes plant heat-up and cool-down, power level increases (up to the specified maximum rates corresponding to a step load increase of 10 percent (%) and ramp load increases of 5% per minute), and load decreases (up to a full load rejection). Historically, these types of normal operational occurrences were categorized as ANSI N18.2 Category I (Ref. 15.0-1) and are not addressed in the accident analyses.

Anticipated operational occurrences (AOOs) are events in which the reactor plant conditions are disturbed beyond the normal operating range. AOOs are expected to occur one or more times during the lifetime of the plant. During a transient caused by an assumed AOO, the reactor core must be undamaged and be ready to return to normal operation. AOOs are also referred to as incidents of moderate frequency and infrequent incidents in RG 1.206 (Ref. 15.0-2). AOOs generally result from one of the following:

- A single component failure.
- A single malfunction, including passive failures such as leaks or minor pipe breaks, which could occur during the life of the plant while the plant is operating.
- A single operator error.

Acceptance criteria generally applied to AOOs include the following (SRP 15.0 and 4.2):

- The minimum departure from nucleate boiling ratio (DNBR) shall be greater than or equal to the 95/95 DNBR limit.
- Pressure in the reactor coolant system (RCS) and main steam system shall be equal to or less than 1.1 times the system design pressure.
- The maximum fuel centerline temperature shall be less than the fuel melting point so that the fuel cladding will not be mechanically damaged.
- An AOO shall not generate a PA without other faults occurring independently or result in a consequential loss of function of the reactor coolant system or reactor containment barriers.

The SRPs provide additional criteria for certain initiating events, which are described on a case-by-case basis in each respective event analysis section.

The third column of Table 15.0-1 indicates which initiating events are classified as AOOs.

15.0.0.1.2 Postulated Accidents

Postulated Accidents (PAs) are unanticipated events or transients that are not expected to occur over the life of the plant, but which could cause the release of radioactive materials from the plant. PAs are analyzed to confirm the adequacy of the plant safety systems including the engineered safety features (ESF).

Acceptance criteria generally applied to postulated accidents include the following (SRP 15.0):

- Pressure in the RCS and main steam system shall be maintained below acceptable design limits, considering potential brittle fracture as well as ductile failures.
- Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR falls below the limit, the fuel is assumed to fail.
- The maximum radiological consequences shall be less than 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE).

- A postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

For the rod ejection accidents, the following acceptance criteria apply (SRP 4.2 Appendix B and SRP 15.4.8):

- Fuel enthalpy shall not exceed the permissible design limit of 230 cal/gm for UO_2.
- Several additional event-specific criteria, which are described in Section 15.4.8.2.5, are applied to the rod ejection accidents.

For loss-of-coolant accidents (LOCA), the analysis criteria of 10 CFR 50.46 also apply (SRP 15.0):

- The calculated maximum fuel clad temperature shall not exceed 2200^{o}F.
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for an extended period of time as required by the long-lived radioactivity remaining in the core.

The SRPs provide additional criteria for certain initiating events, which are described on a case-by-case basis in each respective event analysis section.

The third column of Table 15.0-1 indicates which initiating events are classified as PAs.

15.0.0.2 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.0.2.1 Design Plant Conditions

Table 15.0-2 lists key rated (nominal) power conditions. Two power ratings are considered:

- The design core thermal power output.
- The design nuclear steam supply system (NSSS) thermal power output, which includes the thermal power generated by the reactor coolant pumps (RCPs).

Most of the event analyses assume the NSSS thermal power output; the remaining event analyses assume the design core thermal power output. The thermal power values used for each analyzed event are provided in the last column of Table 15.0-1.

Nominal control system characteristics, developed from operational experience, are modeled in the analyses as they are expected to perform.

Power operation with an inactive loop is not allowed by the technical specifications for the US-APWR; therefore, unless otherwise stated in the individual event analysis section, all RCPs are assumed to be operating at the time of event initiation.

Values for other pertinent plant operating parameters used in the event analyses are provided in Table 15.0-3.

15.0.0.2.2 Initial Conditions

Accident analyses are performed with conservative bases. Inputs to the analyses are developed to include allowances for operational uncertainties (such as allowances for instrument, measurement, and setpoint uncertainties) and analysis uncertainties (such as allowances for variations in core parameters and measurement uncertainties for certain basic data, such as delayed neutron fractions). In principle, conservative values are employed for each parameter that is input to calculations performed as part of the accident analyses. The uncertainty allowance for each parameter is applied in the direction that produces the most adverse calculated consequences for the transient.

The nominal conditions (power, temperature, and pressure) are assumed as the initial conditions for departure from nucleate boiling (DNB) limited events that are analyzed using the revised thermal design procedure (RTDP) discussed in Section 4.4.1.1.2. The RTDP accounts for uncertainty allowances by a statistical combination technique that is part of the procedure.

For certain other events, the initial conditions are obtained by adding the maximum steady-state errors to the rated values in the conservative direction. The conservatively assumed steady-state errors are as follows:

- Core power — ±2 percent allowance for calorimetric error
- Average RCS temperature — ±4°F allowance for controller deadband and measurement error
- Pressurizer pressure — ±30 psi allowance for steady-state fluctuations and measurement errors

The initial power values used in the transient analyses are summarized in Table 15.0-1.

15.0.0.2.3 Power Distribution

The response of the reactor to a transient condition depends in part on the core power distribution at the beginning of the transient. The reactor core is designed to have a relatively uniform power distribution. This is accomplished by the arrangement of fuel assemblies, the location of rod cluster control assemblies (RCCAs), the grouping of specific RCCAs into banks, the selection of the sequence of RCCA withdrawal steps, and the initial design of the fuel assemblies, including fuel pellet enrichment, arrangement of gadolinia integral fuel rods within fuel assemblies, the type of burnable poison, and the number and location of burnable poison rods within the fuel assemblies.

Power distributions are characterized by the nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$), which is essentially a radial power peaking value; the heat flux hot channel factor (F_Q), which is a local ("point") power peaking value; and the axial power distribution.

In the analyses, all transients are assumed to begin with the most severe power distributions that are consistent with operation within the technical specifications. Power peaking factors employed in the analyses are listed in conjunction with specific event analyses discussed in this chapter. Additional general information about F_Q and $F^N_{\Delta H}$ is described in Section 4.4.4.3 and design axial power distributions are described in Section 4.3.2.2.

The value of $F^N_{\Delta H}$ increases as the power level decreases due to effects caused by the insertion of RCCAs. An increase in $F^N_{\Delta H}$ consistent with technical specification limits is factored into the over temperature ΔT trip limits shown in Figure 15.0-1.

Power increase transients that are relatively slow (e.g., a step increase in steam flow) may cause the reactor to establish a new steady state condition at higher power without causing a reactor trip. The fuel rod thermal evaluations performed for this type of transient may be performed as part of the fuel design process using steady state analysis methods as described in Section 4.2.

Power increase transients that are relatively fast with respect to the fuel rod thermal time constant (e.g., uncontrolled control rod assembly withdrawal from a subcritical and rod ejection accidents) may cause a large rapid power increase that can challenge one or more fuel design limits. Detailed fuel transient heat transfer calculations are performed for this type of transient using bounding design power distributions, accident-specific power distributions, or in certain cases, time-dependent power distributions calculated during the accident. Power distribution assumptions are described as part of specific event analyses discussed in this chapter.

15.0.0.2.4 Reactivity Coefficients

The transient response of the reactor depends on the reactivity feedback effects, particularly the moderator temperature or density coefficient and the Doppler power coefficient. These reactivity coefficients are discussed in Section 4.3.2.3.

A bounding maximum or minimum value of the reactivity coefficients is used in each event analysis. In many cases, the conservative combination of moderator temperature and Doppler power coefficients represents either beginning or end of core fuel cycle conditions. In certain cases, conservative combinations of parameters that are not representative of realistic situations are used in order to bound their combined effects at all times during the fuel cycle. The maximum and minimum Doppler power coefficient values used in the transient analyses are provided in Figure 15.0-2. For most accidents analyzed using the MARVEL-M code (described in Section 15.0.2.2.1), one of two constant values of the moderator density coefficient is used. The minimum value is 0.0 (Δk/k)/(g/cc) and the maximum value is 0.51 (Δk/k)/(g/cc). The justification for the use of specific values of these coefficients is described on a case-by-case basis in the respective analysis section. A summary of the reactivity coefficient assumptions used for each event is provided in Table 15.0-1.

15.0.0.2.5 Rod Cluster Control Assembly Insertion Characteristics

A reactor trip signal causes all of the RCCAs to be inserted by gravity to the bottom of the active fuel region. In the analyses, the single highest-reactivity-worth RCCA is conservatively assumed to fail to insert (i.e., to remain fully withdrawn).

Figure 15.0-3 is a plot displaying the conservative RCCA displacement as a function of time that is used in the analyses for the RCCA insertion following a reactor trip until RCCAs are fully inserted. A time of 3 seconds is used for insertion time to dashpot entry.

Figure 15.0-4 shows the negative reactivity addition as a function of time that is used in the analysis for the RCCA insertion following a reactor trip. This curve is based on: (1) the same conservatively slow RCCA insertion rate discussed in the preceding paragraph and (2) a conservative bottom-skewed axial power distribution within the core.

The RCCA negative reactivity insertion versus time shown in Figure 15.0-4 is input into the computer codes used in the analyses. Unless otherwise described in the individual event analysis sections, the scram reactivity is -4%Δk/k for hot full power condition.

15.0.0.2.6 Residual Decay Heat

15.0.0.2.6.1 Total Residual Decay Heat

Residual heat in a subcritical core, including decay heat from fission products and actinides, is calculated for the large break LOCA and the non-LOCA transient in accordance with the methodology of ANSI/ANS-5.1-1979 (Ref. 15.0-15).

For the small break LOCA and post-LOCA long-term cooling analysis, the decay heat from fission products is assumed to be equal to 1.2 times the values for infinite operating time in the ANS standard 5.1-1971, conforming to the requirement of 10 CFR 50 Appendix K (Ref. 15.0-7). The heat from the decay of actinides is calculated in accordance with the methodology of ANSI/ANS-5.1-1979.

Input parameters used with ANSI/ANS-5.1-1979 are selected so as to envelope conceivable core conditions for the US-APWR.

15.0.0.2.6.2 Distribution of Decay Heat Following a Loss of Coolant Accident

Early in a LOCA, the neutron chain reaction in the core is terminated due to void formation or RCCA insertion, or both. After this shutdown, most of the heat generation in the core results from absorption of gamma rays generated by the fission products. The location where gamma rays are absorbed may be some distance from the parent fission product nuclei. Therefore, the distribution of heat generation within the core is changed from that at the steady-state. The fraction of heat generated within the clad and pellet is 97.4 percent at the steady state. This factor reduces for the high-power rod after shutdown. The large-break LOCA analyses account for this effects.

15.0.0.3 Reactor Trip System and Engineered Safety Feature Systems Analytical Limit and Delay Times

The reactor trip system (RTS) initiates signals to open the reactor trip breakers when operating parameters that are monitored by the RTS approach pre-determined limits. This action removes power to the control rod drive mechanism (CRDM) coils permitting the rods to fall by gravity into the core.

Instrumentation system time delays are associated with each of the RTS trip functions. These include delays in sensor, signal generation, opening the reactor trip breakers, and the release of the rods by the CRDM. The total response time delay for a reactor trip is the interval from the time the operating parameter reaches the analytical limit to the time the control rods are released and start to drop into the core. The delay for each trip signal is selected so as to give conservative analysis results. Chapter 7 provides a general discussion of the reactor trip and engineered safety features (ESF) actuation signals.

Table 15.0-4 summarizes the reactor trip and ESF actuation analytical limit and response delay times for functions used in the event analyses. The difference between the trip analytical limit and the nominal trip setpoint specified in the plant technical specifications accounts for instrumentation channel error and setpoint error. Both the availability and range of each type of instrumentation are consistent with the corresponding predicted parameter values in the specific event analyses. The instrumentation and control characteristics used in the specific event analyses (including values used for analytical limit and delay times) are consistent with the information documented in Sections 7.2 and 7.3.

Table 15.0-5 summarizes the time delays associated with accident-mitigating equipment such as valve movement or pump start delays. In addition, instrumentation is provided to monitor key plant parameters during events (e.g., EFW flow indication in the main control room) as described in Section 7.5.

15.0.0.4 Component Failures

The accident analyses documented in this chapter account for certain component failures. This treatment is an element of the defense in depth safety philosophy. The discussion of component failures in this section is based in part on Reference 15.0-11.

Certain events (such as a steam system piping failure) are initiated by component failures. The analysis of these accidents assumes that any equipment that can be damaged as a consequence of the initiating event is not available for mitigation of the accident. In addition, the analysis must demonstrate that the acceptance criteria are met with concurrent single failures in safety system employed to mitigate the postulated event in accordance with the single failure criterion.

A single failure is an occurrence which results in the loss of capability of a single component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. (For example, a single failure could disable an onsite AC emergency power source that supplies power to multiple safety systems. The loss of function of all the multiple safety systems powered by the same power source would be counted as a single failure.)

Active failures are considered for immediate actuations of safety-related systems to shut down the reactor, provide for core cooling, and mitigate the consequences of events. Passive failures are generally considered for the safety-related systems in post-accident service after 24 hours from initiation of events and only a limited subset of possible passive component failures are considered. Active and passive component failures are described in more detail in the following two sections.

15.0.0.4.1 Active Failures

An active failure of a component in a fluid system is where mechanical movement necessary for the component to complete its intended function does not occur on demand. An active failure may also apply if the motion takes place spuriously when it is not demanded. As such, an active failure can be defined differently for different components. Passive component failures are described in the following section.

15.0.0.4.2 Passive Failures

Passive failures in safety-related fluid systems are generally a limited breach or leakage in the fluid pressure boundary or a mechanical failure which adversely affects a flow path, such as a failure of a seal at a valve or pump. A failure of component which performs its function to change fluid flow with simple mechanical movement of a part of the component without external force on demand might be regarded as a passive failure if the function has a sufficiently high reliability.

Failures of major passive fluid system components are considered as accident initiators (such as in the large-break LOCA and the steam system piping failure).

15.0.0.4.3 Limiting Single Failures or Operator Errors

The design of the US-APWR is such that no single equipment failure or single operator error will result in an inability to successfully mitigate postulated events. The effect of single failures or operator errors has been considered in the analysis of Chapter 15 events.

For accident analysis, reactor transients are carried out in time until parameters safely stabilize, the event-specific acceptance criteria are met, and the reactor can be expected to be brought to safe shutdown within 24 hours from the event initiation. Single active failures are therefore accounted for in the accident analyses by performing the evaluations accounting for the most limiting single active failure (the one that has the most adverse impact on the consequences of the accident) to meet the acceptance criteria such as DNBR limit, pressure limit, core cooling capability limits, radioactivity confinement, etc. This limiting single failure (where one exists) is identified in each analysis description. These analysis descriptions also describe how the failure impacts the sequence of events associated with the transient. In most instances, even the most severe single failures have no adverse effect on the consequences of the transient because of redundancy in protection equipment. A summary of the assumed limiting single failure for the core response for each event is provided in Table 15.0-6.

Operators might make errors of omission of actions required or may take the wrong action. A single operator error is the set of actions that is a direct consequence of a single erroneous decision. Operator errors are assumed as event initiators, but are not expressly accounted for in the accident analysis.

The man-machine interface is designed using state-of–the-art ergonomics and advanced hardware to preclude operator errors. In addition, automatic actuation of safety systems is provided in the design for complex sequential actions or when actions are required within a short period of time in response to abnormal events. Accident analysis is performed to confirm that required operator actions are feasible by ascertaining that sufficient and adequate audible alarms as indications of abnormal condition are available to the main control room operators and that the actions can be completed in the required time.

Operator actions credited in the accident analysis are discussed in section 15.0.0.6.

15.0.0.5 Non Safety-Related Systems Assumed in the Analyses

The US-APWR design is such that the plant is capable of protecting both the public and workers against the possible effects of natural phenomena, postulated environmental conditions, and the dynamic effects of postulated accidents. The design also incorporates features to minimize the occurrence and effects of fires and/or explosions.

A quality assurance (QA) program has been implemented to ensure that plant systems are capable of satisfactorily performing their assigned safety functions. The combination of the QA Program and reliability of the design provide confidence that the normally operating systems and components are available for the mitigation of the various events described in Chapter 15.

Non safety-related systems are not required to mitigate the consequences of events. Only safety-related systems are credited in the US-APWR safety analyses. Nominal control system characteristics are modeled (best estimate) in the accident analyses only if they adversely impact the results.

15.0.0.6 Operator Action

Many of the events analyzed in Chapter 15 are terminated by an automatic reactor trip. Following such a trip, the plant is usually in a safe, stable, hot standby condition, similar to a normal shutdown. In these cases, actions taken by the operator are essentially normal operating procedures.

After some events, however, the fault that initiated the trip also renders certain equipment unavailable or ineffective after the trip (such as when the initiating event for the transient involves a break in the RCS, the feedwater system, or the steam system piping). The actions taken following these transients, and the time at which these actions occur, depend on what systems are available and the plans for post-transient plant operation.

For all shutdowns, decay heat must be removed through the steam generators to maintain a stable hot standby condition.

In addition, for all shutdowns from power, if a hot standby condition is maintained for an extended time, operator action may be required to add boric acid through the chemical and volume control system (CVCS) or ECCS with emergency letdown to compensate for xenon decay, which could otherwise reduce shutdown margin.

Operator actions required to mitigate accidents are described in the individual event evaluation sections. The non-LOCA events whose analyses credit operator actions are inadvertent dilution of boron concentration in the RCS (Section 15.4.6), CVCS malfunction that increases RCS inventory (Section 15.5.2), and steam generator tube failure (Section 15.6.3). The radiological consequence events whose analyses credit operator actions are RCCA ejection (Section 15.4.8) and failure of small lines carrying primary coolant outside containment (Section 15.6.2). In addition, operator actions are credited to prevent boric acid precipitation to assure post-LOCA long term cooling (Section 15.6.5).

15.0.0.7 Loss of Offsite AC Power

The analyses for AOOs and other accidents consider transients both with and without offsite power available for cases where the transient includes a reactor trip. This analysis approach is consistent with 10 CFR 50, Appendix A, General Design Criteria (GDC) 17 (Ref. 15.0-12).

The unavailability of offsite power is not considered in characterizing the frequency of the event sequence (i.e., for transients that are AOOs, the AOO acceptance criteria are applied even when the offsite power is considered to be unavailable).

The loss of offsite power is considered in addition to the limiting single failure assumed for the event sequence where offsite power is available.

The US-APWR is designed such that the normal source of electrical power for the RCPs is the plant generator. The plant design incorporates a time delay between turbine and generator trips, assuring that power to the RCPs is maintained for a period of time following a turbine trip. This design feature is conservatively ignored. The reactor trip is assumed to cause a disturbance in the utility grid, which causes the loss of offsite power (LOOP). The accident analyses assume a loss of offsite power occurs 3 seconds after the reactor trip. This 3-second delay accounts for the time it would take for a grid instability caused by the reactor trip to propagate through the grid to the plant offsite power source.

The principal concern with a LOOP occurring at the time of reactor trip is that a complete loss of flow transient would be superimposed on the initiating event. With the beginning of the reactor coolant pump coastdown delayed more than 3 seconds after reactor trip, the rods are inserted to the dashpot by the time the LOOP (and corresponding loss of flow) is initiated. (Refer to Figure 15.0-3) This time delay between the reactor trip and pump coastdown assures that the portion of the transient following a postulated LOOP occurs after the limiting DNBR. Therefore, the minimum DNBR at any time during the transient is the same with offsite power available or unavailable. For this reason, the LOOP cases following reactor trip are not presented in the event-specific analyses.

For peak pressure analyses, loss of offsite power at the time of turbine trip is considered only if this assumption is conservative.

15.0.0.8 Long Term Cooling

The reactor trip and ESF actuation systems are designed to mitigate accident conditions and to stabilize the plant at hot standby conditions. After the plant has been stabilized the operators may continue to maintain the plant at hot standby or transition to and maintain the plant at cold shutdown conditions. In either case, adequate core cooling must be maintained to assure residual heat removal.

The residual heat removal system (RHRS) is designed to remove heat energy from the core and the RCS during shutdown and refueling conditions. Section 5.4.7 provides a detailed description of the US-APWR RHRS and safe shutdown operation. The RHRS consists of four independent subsystems (flow paths), each containing a containment spray/residual heat removal (CS/RHR) heat exchanger, a CS/RHR pump, and associated piping and valves. The CS/RHR heat exchangers and the CS/RHR pumps have functions in both the containment spray system (CSS) and the RHRS.

During system operation, each CS/RHR pump takes suction from one of the RCS hot legs by a separate suction line, and discharges through its respective CS/RHR heat exchanger. The reactor coolant is then returned to the RCS cold legs.

One of the RHRS safety functions is to remove the reactor core decay heat and other residual heat, thus reducing the reactor coolant temperature during safe shutdown. Safe shutdown is defined for the US-APWR as achieving cold shutdown conditions following design base events such as earthquakes and anticipated operational occurrences using safety-related equipment. Any two subsystems of the RHRS are sufficient to achieve safe shutdown conditions. With only two subsystems operating, the RHRS is designed to reduce the RCS temperature to 200°F within 36 hours after reactor shutdown.

The RHRS is placed in operation when the RCS temperature and pressure are reduced to approximately 350°F and 400 psig, respectively. In safe shutdown, the RHR flow is controlled by throttling RHR flow control valves installed in each of the four RHR return lines.

In the large break LOCA event, containment spray is actuated. The CS/RHR heat exchangers are used for long term cooling by removing heat from the refueling water storage pit (RWSP) water, which is recirculated to the reactor vessel by the safety injection pumps. When the containment pressure is sufficiently reduced, the operator terminates containment spray by closing the containment spray header isolation valve and realigns the CS/RHR pumps to discharge back into the RWSP to establish the recirculation path.

The duration of the safety analysis evaluation of the Chapter 15 events is generally only long enough to assure that the appropriate acceptance criteria have been met and does not typically include the transition to shutdown conditions using the RHRS. Therefore, the event-specific discussion does not typically address long-term cooling. However, any event-specific assumptions regarding the actuation and operation of the RHRS are described in the appropriate event-specific safety analysis section.

15.0.0.9 Pump Seal Cooling With Containment Isolation

Normally, the RCP seal is cooled by seal water injection from CVCS. In addition to seal injection, supplemental seal cooling is also provided by the RCP thermal barrier heat exchanger using component cooling water (CCW). Loss of seal cooling could lead to seal degradation if the condition persists. In the event of containment vessel (C/V) isolation and loss of seal water injection, both cooling means are lost. As discussed in Section 9.2.2, the C/V isolation valves on the CCW supply and return lines can be manually reopened from the main control room to restore RCP seal cooling using the RCP thermal barrier heat exchanger. The CCW system is designed to restore CCW to the RCP thermal barrier heat exchanger assuming a single failure and loss of offsite power.

15.0.1 Radiological Consequence Analyses Using Alternative Source Terms

Although the US-APWR utilizes the alternative source term (AST) methodology, this section is focused on the application of AST to operating reactors and is not applicable to the US-APWR. See Section 15.0.3 for the details of the radiological consequence analyses for the US-APWR.

15.0.2 Review of Transient and Accident Analysis Methods

15.0.2.1 Analysis Methods

The following table summarizes the various MHI topical reports that are relevant to the safety analyses provided in Chapter 15.

Topical Report Title	Document Number	Date / Rev.
Fuel System Design Criteria and Methodology	MUAP-07008-P	May 2007 / 0
Thermal Design Methodology	MUAP-07009-P	May 2007 / 0
Non-LOCA Methodology	MUAP-07010-P	July 2007 / 0
Large Break LOCA Code Applicability Report for US-APWR	MUAP-07011-P	July 2007 / 0
Small Break LOCA Methodology for US-APWR	MUAP-07013-P	July 2007 / 0

15.0.2.2 Computer Codes Used

The fourth column of Table 15.0-1 includes a summary listing of the computer codes used for analyzing specific events. Additional information about these computer codes is provided in the following sections. Any specialized modeling capabilities that are unique to a specific given event are summarized in the respective event analysis section.

15.0.2.2.1 MARVEL-M

MARVEL-M (Ref. 15.0-14) is a multi-loop plant system transient analysis code used to calculate detailed transient behavior of pressurized-water reactor systems. MARVEL-M has a maximum modeling capability of four coolant loops with four steam generators and associated systems. It simulates reactor kinetics, thermal-hydraulics of the core and reactor coolant system, the pressurizer, main and secondary steam and feedwater systems, and the reactor control and protection system. It also simulates the ESF systems and other subsystems, which are representative of conventional pressurized-water reactor (PWR) power plants.

The MARVEL-M program utilizes a space-independent single point reactor kinetics model with six delayed neutron groups. The thermal and hydraulic characteristics of the RCS are described by time- and space-dependent differential equations. The RCS is represented by flow nodes, which model transient behaviors of mass and energy for the ranges of sub-cooled and homogenous two phase fluid typically encountered in the analysis of non-LOCA transients. Pressurizer heaters, spray, and safety valves are also considered in the program. Reactivity effects from the moderator, fuel, boron, and rods are also included. MARVEL-M also simulates the protection and monitoring system and control systems.

MARVEL-M has the ability to calculate the value of DNBR during a transient using a simple calculation model. The model employs user-input values of the DNBR at nominal core conditions and selected DNBR limits represented by operating parameters of core inlet temperature, pressure and power levels. The simplified DNBR model closely agrees with design calculations when the core operating conditions do not exceed the design flux distribution or core protection limits. When conditions exceed these limitations, DNBR analysis is performed by the more detailed external calculation code, VIPRE-01M, which is discussed in Section 15.0.2.2.2.

MARVEL-M outputs the transient response of reactor power, reactor pressure, primary coolant temperature, DNBR, and other parameters. Inputs into the code include initial conditions such as primary coolant temperature and the reactor power, primary coolant volume and other plant data, nuclear characteristics data, and setpoints for actuation of the reactor trip system and ESF systems. The program is applicable to both conventional as well as advanced PWR plants (APWRs).

15.0.2.2.2 VIPRE-01M

VIPRE-01M (Ref. 15.0-3) is a subchannel thermal hydraulic analysis code with both steady state and transient capabilities, including a fuel thermal transient model. It divides the core into three-dimensional mesh elements and then solves the appropriate equations by applying the mass, momentum, and energy conservation principles to each mesh element.

Inputs into VIPRE-01M include initial conditions such as reactor power, coolant temperature, coolant flow, power distributions, core geometry and fuel properties.

VIPRE-01M calculates time-dependent changes in parameters, such as coolant temperature, coolant density, void fraction, fuel temperature, and minimum DNBR in the core. Boundary conditions are the transient data generated by MARVEL-M or TWINKLE-M.

15.0.2.2.3 TWINKLE-M

TWINKLE-M (Ref. 15.0-14) is a multidimensional spatial neutron kinetics code which solves the two-group transient diffusion equations using a finite difference technique. This code is used to analyze changes in dynamic behavior of space- and time-dependent neutron flux in response to reactivity accidents.

TWINKLE-M also uses a six-region model for fuel rod heat transfer between the fuel pellet, clad, and primary coolant and a primary coolant thermal hydraulic model, which handles behavior in the vertical-axis using the mesh points used for the dynamic analysis of the neutron flux. This capability enables feedback effects, including Doppler and moderator feedback effects, to be modeled as space-dependent. The feedback effects are taken into account by absorption cross-section compensation at each mesh point.

Inputs into TWINKLE-M include time-dependent changes at each mesh point of the neutron cross sections, core inlet temperature, pressure, flow rate though the core, boron concentration, and control rod motion. It outputs the neutron flux level, neutron flux distribution, and the thermal response of the core as space- and time-dependent parameters.

15.0.2.2.4 RADTRAD

The RADionuclide Transport, Removal, and Dose (RADTRAD) (Ref. 15.0-16) computer code is a computer model for estimating doses at offsite locations such as the exclusion area boundary (EAB) and the low-population zone (LPZ), as well as onsite locations (e.g., main control room (MCR)) due to postulated radioactivity releases from design basis accident conditions. RADTRAD uses a compartment model and simulates radioactive material transport through the containment, and related systems, structures and components. RADTRAD calculates dose consequences for different specified time intervals based on user-input information on the amount, form, and species of the radioactive material released in the reactor plant. See Appendix 15A for additional details regarding the RADTRAD code.

15.0.2.2.5 ANC

The nuclear analysis code ANC is described in Sections 4.3.3.1 and 4.3.2.

15.0.2.2.6 FINE

The MHI FINE code incorporates all of the basic fuel rod performance models required to evaluate expected in-reactor fuel behavior as described in topical report MUAP-07008-P (Ref. 15.0-13).

15.0.2.2.7 WCOBRA/TRAC (M1.0)

The WCOBRA/TRAC (M1.0) code, a modified version of the WCOBRA/TRAC code, is used for calculation of thermal-hydraulic behavior during a large break LOCA. It’s applicability to the US-APWR large break LOCA analysis is discussed in the Topical Report (Ref. 15.0-18).

WCOBRA/TRAC is approved by the U.S. Nuclear Regulatory Commission for use in best estimate large break LOCA calculations for three and four loop conventional PWRs, also the AP600 and AP1000 advanced plant designs. The COBRA portion of the code is based on a two-fluid, three-field, multi-dimensional fluid equations to describe thermal-hydraulic behavior of the vessel component. The TRAC portion of the code is based on one-dimensional, two-phase drift flux model to describe thermal-hydraulic behavior of the major components of PWR, such as steam generators, pipes, pumps, valves and pressurizer.

15.0.2.2.8 HOTSPOT

The HOTSPOT (Ref. 15.0-18) code is used for detailed fuel rod model analysis.

HOTSPOT calculates the effect of uncertainties at axial location of the fuel rod. The code uses a transient conduction model identical to that of WCOBRA/TRAC. The code also simulates cladding burst, metal-water reaction and fuel relocation following burst phenomena.

15.0.2.2.9 M-RELAP5

The M-RELAP5 computer code (Ref. 15.0-19) is a modified version of the RELAP5-3D code developed at the Idaho National Laboratory (Ref.15.0-8). M-RELAP5 has the capability to analyze thermal hydraulic behaviors and safety performance of the MHI US-APWR during a small break LOCA in compliance with the requirements specified in 10 CFR 50.46, ”Acceptance Criteria for Emergency Core Cooling System for Light-Water Nuclear Power Reactors” (Ref. 15.0-6) and in 10 CFR 50 Appendix K, ”ECCS Evaluation Models” (Ref. 15.0-7).

The M-RELAP5 code is based on a non-equilibrium separated two-phase flow thermal hydraulic approach with additional models to describe the behavior of the components of reactor systems including heat conduction in the core and reactor coolant system, reactor kinetics, control systems and trips. The code also has generic and specialized component models such as pumps and valves. In addition, special process models are included to represent those effects important in a thermal hydraulic system including form loss, flow at an abrupt area change, branching, and choked flow.

M-RELAP5 is used to model the following subsystems for the US-APWR for the small break LOCA analysis:

- Primary system (reactor and core, reactor coolant system, emergency core cooling system)
- Secondary system (main steam system, main feedwater system, emergency feedwater system)
- Containment vessel

15.0.3 Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors

15.0.3.1 Evaluation Models and Parameters for Analyses of Radiological Consequences of Postulated Accidents

15.0.3.1.1 Methodology

Radiological consequences from postulated accidents are calculated employing the alternative source term (AST) methodology in accordance with Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 15.0-4).

This methodology is implemented using the RADionuclide Transport, Removal, And Dose (RADTRAD) computer code (Ref. 15.0-16). The RADTRAD code is designed to calculate the radiological consequences to the offsite population and to MCR operators following a design-basis accident at light water reactor (LWR) power plants. The code can track the transport of radionuclides as they are released from the reactor vessel, travel through the containment and other buildings, and are released to the environment. As the radioactive material is transported through the containment and other buildings, credit for several removal mechanisms may be taken including sprays, suppression pools, overlying pools, filters, and natural deposition. Simple models are available in the code for these different removal mechanisms that use, as input, information about the conditions in the plant and predict either a removal coefficient or decontamination factor. RADTRAD was executed using the specified input parameters for each accident type as listed in the appendices to RG 1.183 (Ref. 15.0-4).

The AST methodology determines the TEDE, which is the sum of the committed effective dose equivalent (CEDE) and the deep dose equivalent (DDE) external exposure (such as immersion in a plume or cloud of airborne radioactive material).

15.0.3.1.2 Radiological Doses

The TEDE is evaluated for the EAB, the outer boundary of the LPZ, and the MCR (for LOCA). The acceptance criteria for these calculated TEDE values are listed in Table 15.0-7 (Ref. 15.0-4). The EAB dose is calculated for the two-hour period that causes the highest dose at any point on the EAB following the accident. The LPZ dose is calculated as the highest dose for a postulated individual who stays at any point on the outer boundary of the LPZ for the duration of the accident. The MCR dose is calculated as the highest dose for a postulated individual who stays in the MCR for the duration of the accident. Table 15.0-7 contains additional information about accident durations for the LPZ and MCR receptors.

For the EAB and LPZ, the DDE is determined by summing (for each type of airborne nuclide released from the reactor plant that reaches the postulated receptor) the product of: (1) the activity of the radioactive nuclide that is released; (2) the exposure-to-effective dose equivalent (EDE) ratio; and (3) the atmospheric dispersion factor (χ/Q value).

For the EAB and LPZ, the CEDE (inhalation dose) is determined by summing (for each type of airborne nuclide released from the reactor plant that reaches the postulated receptor) the product of: (1) the activity of the radioactive nuclide that is released; (2) the exposure-to-CEDE ratio; (3) the receptor's breathing rate; and (4) the χ/Q value.

For the MCR, dose calculation methods include adjustments for the finite size of the radioactive cloud, time operators may be off duty, and for filtration and air purification equipment. Specific methods used for calculating MCR doses are described in Section 15.6.5.5.

EAB and LPZ radiological doses for each event with a radiological release are summarized in Table 15.0-17. MCR and technical support center doses for the LOCA events are discussed in Section 15.6.5.5.

15.0.3.1.3 Source Terms

Certain accidents can result in the release of primary and secondary coolant to the environment. The technical specifications limit the iodine and noble gas radioactivity in the primary and secondary coolant. Design values for the activity concentration of radioactive fission, corrosion, and activation products in the primary and secondary coolant are described in Section 11.1. Fission products can be present in the primary coolant due to leaking fuel rods (i.e., rods that are leaking prior to the postulated accident). Activity in the primary coolant is assumed to be transferred to the secondary coolant due to leaking steam generator tubes. The quantity of coolant released to the environment during postulated accident sequences is described in the event-specific discussion for those accidents that have radiological consequences due to coolant releases.

The cladding on previously non-leaking fuel rods can become damaged during certain non-LOCA accidents involving fuel in the reactor core. This breached cladding can release fission products in the gap between the fuel pellet and the cladding of the fuel rod. This fuel rod gap inventory can be transferred to the reactor (primary) coolant, then be transported to the secondary coolant via postulated leaking or failed steam generator tubes, and then be released to the environment when the main steam safety valves or relief valves are assumed to be used to dissipate decay heat by releasing steam to the environment.

Table 15.0-8 lists the fraction of the fission product inventory assumed to be in the fuel rod gap (for various nuclides and groups of nuclides) for non-LOCA accidents. Table 15.0-9 lists the elements making up the various groups of fission product nuclides. The information in Tables 15.0-8 and 15.0-9 is adapted from Reference 15.0-4.

For some events, the iodine concentrations in the reactor coolant are calculated using special assumptions that ensure the calculations account for conservatively large quantities of radioactive iodine by assuming: (1) a pre-transient iodine spike and (2) a transient initiated iodine spike.

For the pre-transient iodine spike, a reactor transient is assumed to have occurred prior to the initiating event of the transient and has raised the reactor coolant iodine concentration from 1 μCi/g dose equivalent (DE) I-131 (this quantity is the maximum value permitted by the technical specifications) to 60 μCi/g DE I-131. Table 15.0-10 lists the iodine concentrations consistent with concentrations of 1 and 60 μCi/g DE I-131.

For the transient initiated iodine spike, the transient itself is assumed to cause an iodine spike in the primary system. The increase in reactor coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the reactor coolant (expressed in curies per unit time) increases to a value as much as 500 times (for the steam system piping failure and failure of small lines carrying primary coolant outside containment events) or 335 times (for the steam generator tube failure event) greater than the release rate corresponding to the iodine concentration at the equilibrium value (1 μCi/g DE I-131) specified in the technical specifications. These iodine appearance rates are listed in Table 15.0-11.

The pre-accident noble gas concentrations in the reactor coolant are based on the technical specification limit of 300 μCi/g DE Xe-133. Also, the pre-accident alkali metal concentrations in the reactor coolant are based on 1% fuel defect. The secondary coolant iodine and alkali metal activities are assumed to be 10% of the maximum equilibrium reactor coolant activity. Table 15.0-12 lists the noble gas concentrations consistent with concentrations of 300 μCi/g DE Xe-133.

The total core fission product inventory is described in Section 15.0.3.2.

15.0.3.1.4 Radiological Dose Parameters

The calculated activity of each radioactive nuclide released to the environment during accidents is described in the event-specific event description for those accidents that have radiological consequences.

The exposure-to-CEDE dose conversion factors used in the US-APWR analysis are taken from Table 2.1 of "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11 (Ref. 15.0-9). These values are derived from data in International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers," (Ref. 15.0-17). As prescribed in RG 1.183, the factors in the column headed "effective" are used to obtain the CEDE for various radionuclides.

The receptor breathing rates are assumed to be 3.5×10^{-4}, 1.8×10^{-4}, and 2.3×10^{-4} m^3/s for the first 8 hours after the accident, the period from 8 to 24 hours after the accident, and the period from 24 hours until the end of the accident, respectively. Table 15.0-13 lists the receptor breathing rates. These receptor breathing rates are consistent with information in Section 4.1.3 of Reference 15.0-4.

As discussed in RG 1.183 (Ref. 15.0-4), the external dose DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the EDE from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. The analysis discussed in the sections for Chapter 15 uses Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 15.0-10), for external EDE conversion factors. As prescribed in RG 1.183, the factors in the column headed "effective" are used to obtain the EDE for various radionuclides.

The dose conversion factors listed in Table 15.0-14 have been considered dose effect of daughter's decay. These conversion factors are listed in NUREG/CR-6604 (Ref. 15.0-16). NUREG/CR-6604 is based on Federal Guidance Report 11 and 12.

This DCD is based on a reference reactor site, which is assumed to have atmospheric conditions that will bound most sites in the United States. The χ/Q values used in determining the radiological consequences of postulated accidents are listed in Table 15.0-13.

Radiological consequence results are listed in the section describing each postulated accident (for those postulated accidents that have radiological consequences). Acceptance criteria for calculated EAB and LPZ dose values are listed in Table 15.0-7. These acceptance criteria are based on 10 CFR 50.34 and Table 6 of Reference 15.0-4. The acceptance criterion for the MCR is 5 rem.

15.0.3.2 Fission Product Inventories

The radiological source terms assumed released for specific postulated accidents are calculated using methods consistent with Section 3 of RG 1.183 (Ref. 15.0-4).

The time dependent fission product inventories in the reactor core are calculated by the ORIGEN 2.2 code (Ref. 15.0-5). For core inventory calculation, it is assumed that core has 2 regions. In core inventory calculation, irradiation time for a cycle is assumed 28months. (The planned cycle duration is 24 months.) In this calculation, the average specific power 32.1 MW/MTU is assumed. These calculation conditions lead fission and activation products generated in fuel with burnup of about 55 GWD/MTU in 2 cycles. The core thermal power is 102% of design thermal power. Table 15.0-14 lists the fission product inventories. Table 15.0-14 also identifies the radionuclide group to which each of the nuclides considered in the analysis belongs.

The fission products in the gap region are assumed to be instantaneously released due to any fuel rod cladding failures that are calculated or assumed to occur during accidents other than LOCA. These gap inventories are also assumed to be released during an early phase of LOCA where fuel rods are calculated to be damaged.

Table 15.0-8 summarizes the fraction of fission products by fission product radionuclide group assumed to be in the fuel rod gap for fuel rods experiencing cladding damage in non-LOCA events. These fractions are intended to be adjusted for the relative power fraction (i.e., power peaking factor) for the specific fuel rods calculated or assumed to be damaged during the non-LOCA transient.

Table 15.0-15 summarizes the fraction of fission products released into containment in the (1) gap and (2) early-in-vessel release phases of LOCA analyses performed for the US-APWR for specific fission product radionuclide groups. These values are intended to be applied to the core-average inventories. Table 15.0-16 summarizes the time for onset and the duration of the LOCA release phases for which core inventory fractions are listed in Table 15.0-15.

The chemical forms of the iodine released from the fuel to the containment are prescribed by Reference 15.0-4 to be:

- Cesium iodide (Particulates) 0.95
- Elemental iodine 0.0485
- Organic iodide 0.0015

The values listed in Tables 15.0-8, 15.0-15, and 15.0-16 are based on Section 3 of Reference 15.0-4.

15.0.3.3 Atmospheric Dispersion Factors

The analyses documented in this chapter assume that airborne radioactive material may be released for some events. Radiation dose consequences from these postulated releases are calculated for hypothetical individuals who are assumed to be at the EAB and LPZ surrounding the plant site for these events. These consequences take credit for dispersion as the airborne radioactive materials are transported between the release point and the postulated (receptor) individuals.

The offsite χ/Q values are determined by representative values at the corresponding EAB and LPZ distance selected from the χ/Q value of a reasonable number of the existing sites. The χ/Q values used in determining the radiological consequences of postulated accidents are listed in Table 2.0-1 of Chapter 2. Table 15.0-13 reiterates these χ/Q values.

The MCR χ/Q values are also defined in Table 2.0-1 of Chapter 2. The χ/Q values for the different required time intervals for the MCR are listed by design basis accident event in Tables 15A-18 through 15A-23. The locations of the potential release points and their relationship to the MCR air intake and inleak are shown in Figure 15A-1.

In the COLA, if the site-specific χ/Q values exceed DCD χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 50.34 and the control room dose limits in 10 CFR 50, Appendix A, General Design Criteria 19 are met for affected events using site-specific χ/Q values.

15.0.3.4 Radiological Dose Conversion Factors

The radiological dose conversion factors and half-life employed to calculate radiological consequences for those events that may result in releases of radioactive material from the reactor plant are listed in Table 15.0-14. These values are based on References 15.0-9 and 15.0-10.

15.0.3.5 Airborne Radioactivity Removal Coefficients

Appendix 15A describes the airborne radioactivity removal coefficients that are used in the US-APWR radiological analyses. Equations are presented for two removal mechanisms. One mechanism is elemental iodine removal by wall deposition and the second mechanism is particulate removal.

15.0.4 Combined License Information

COL 15.0 (1) In the COLA, if the site-specific χ/Q values exceed DCD χ/Q values, then the COL Applicant is to demonstrate how the dose reference values in 10 CFR 50.34 and the control room dose limits in 10 CFR 50, Appendix A, General Design Criteria 19 are met for affected events using site-specific χ/Q values.

15.0.5 References

15.0-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.0-2 Combined License Applications for Nuclear Power Plants (LWR Edition), NRC Regulatory Guide 1.206, June 2007.

15.0-3 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.0-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.0-5 RSIC Computer Code Collection CCC-371, ORIGEN 2.2 Isotope Generation and Depletion Code - Matrix Exponential Method, June, 2002.

15.0-6 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, 39 FR 1002, Jan. 4, 1974, as amended at 53 FR 36004, Sept. 16, 1988; 57 FR 39358, Aug. 31, 1992; 61 FR 39299, July 29, 1996; 62 FR 59726, Nov. 3, 1997.

15.0-7 ECCS Evaluation Models, 10CFR 50, Appendix K.

15.0-8 Idaho National Laboratory, RELAP5-3D© Code Manual Volume I: Code Structure, System Models, And Solution Methods, INEEL-EXT-98-00834 Revision 2.4, June 2005.

15.0-9 K.F. Eckerman et al., Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factor for Inhalation, Submersion, and Ingestion, Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988.

15.0-10 K.F. Eckerman and J.C. Ryman, External Exposure to Radionuclide in Air, Water and Soil, Federal Guidance Report 12, EPA-402-R-93-801, Environmental Protection Agency, 1988.

15.0-11 Single Failure Criterion. SECY-77-439, August 1977.

15.0-12 Electric Power Systems, 10CFR Part 50, Appendix A, General Design Criterion 17, ".

15.0-13 Mitsubishi Fuel Design Criteria and Methodology, MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007.

15.0-14 Non-LOCA Methodology, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.0-15 ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, Approved August 29, 1979.

15.0-16 S.L. Humphreys et al., RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose estimation, NUREG/CR-6604, U.S. Nuclear Regulatory Commission, April 1998.

15.0-17 International Commission on Radiological Protection, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, 1979.

15.0-18 Large Break LOCA Code Applicability Report for US-APWR, MUAP-07011-P (Proprietary) and MUAP-07011-NP (Non- Proprietary), July 2007.

15.0-19 Small Break LOCA Methodology for US-APWR, MUAP-07013-P (Proprietary) and MUAP-07013-NP (Non-Proprietary), July 2007.

15.0-20 Small Break LOCA Sensitivity Analyses for US-APWR, MUAP-07025-P (Proprietary) and MUAP-07025-NP (Non-Proprietary), December 2007.

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 1 of 4)

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler	
15.1.1	Decrease in feedwater temperature	AOO	MARVEL-M	max	--	min feedback Figure 15.0-2	4466
15.1.2	Increase in feedwater flow	AOO	MARVEL-M	max	--	min feedback Figure 15.0-2	4466
15.1.3	Increase in steam flow	AOO	MARVEL-M	min and max	--	min feedback Figure 15.0-2	4466
15.1.4	Inadvertent opening of a steam generator relief or safety valve	AOO	MARVEL-M, ANC, VIPRE-01M	See Figure 15.1.4-1	--	See Figure 15.1.4-2	0
15.1.5	Steam system piping failures - Minor/Major	AOO/PA	MARVEL-M, ANC, VIPRE-01M[*1]	Hot standby: Figure 15.1.4-1 HFP: max	--	Hot standby: Figure 15.1.4-2 HFP: min feedback Figure 15.0-2	0%, 75%, & 100% of 4466
15.2.1	Loss of external load	AOO	MARVEL-M	min	--	min feedback Figure 15.0-2	4466 for DNBR 4555[*2] for RCS pressure
15.2.2	Turbine trip	AOO	Bounded by loss of load	--	--	--	--
15.2.3	Loss of condenser vacuum	AOO	Bounded by loss of load	--	--	--	--

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 2 of 4)

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler	
15.2.4	Closure of main steam isolation valves	AOO	Bounded by loss of load	--	--	--	--
15.2.5	Steam pressure regulator failure	N/A to US-APWR					
15.2.6	Loss of non-emergency AC power to the station auxiliaries	AOO	MARVEL-M	min	--	max feedback Figure 15.0-2	4555[*2]
15.2.7	Loss of normal feedwater flow	AOO	MARVEL-M	min	--	max feedback Figure 15.0-2	4466 for DNBR 4555[*2] for RCS pressure, Pzr Level
15.2.8	Feedwater system pipe break - Minor/Major	AOO/PA	MARVEL-M	min	--	max feedback Figure 15.0-2	4555[*2]
15.3.1.1	Partial loss of forced reactor coolant flow	AOO	MARVEL-M, VIPRE-01M	min	--	max feedback Figure 15.0-2	4466
15.3.1.2	Complete loss of forced reactor coolant flow	AOO	MARVEL-M, VIPRE-01M	min	--	max feedback Figure 15.0-2	4466
15.3.3	Reactor coolant pump rotor seizure	PA	MARVEL-M, VIPRE-01M	min	--	max feedback Figure 15.0-2	4555[*2]
15.3.4	Reactor coolant pump shaft break	PA	Bounded by rotor seizure	--	--	--	--

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 3 of 4)

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler	
15.4.1	Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition	AOO	TWINKLE-M, VIPRE-01M, MARVEL-M	--	+2	Temperature coefficient -20% from design	0
15.4.2	Uncontrolled control rod assembly withdrawal at power	AOO	MARVEL-M	min and max	--	min and max feedback Figure 15.0-2	10%, 75%, & 100% of 4466
15.4.3	Control rod misoperation	AOO/PA	MARVEL-M, VIPRE-01M[*1]	min	--	min feedback Figure 15.0-2	4466
15.4.4	Startup of an inactive loop or recirculation loop at an incorrect temperature	AOO	N/A	--	--	--	--
15.4.5	Flow controller malfunction causing an increase in BWR recirculation loop	N/A to US-APWR					
15.4.6	Inadvertent decrease in boron concentration in the RCS	AOO	N/A	--	--	--	0 and 4466
15.4.7	Inadvertent loading and operation of a fuel assembly in an improper Position	PA	ANC	--	--	--	--
15.4.8	Spectrum of rod ejection accidents	PA	TWINKLE-M, VIPRE-01M, MARVEL-M	--	Temperature coefficient -20% from design	Temperature coefficient -20% from design	0 and 4540[*3]

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 4 of 4)

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler	
15.5.1	Inadvertent operation of ECCS that increases reactor coolant inventory	AOO	N/A	--	--	--	--
15.5.2	CVCS malfunction that increases reactor coolant inventory	AOO	MARVEL-M	min	--	min feedback Figure 15.0-2	4555*2
15.6.1	Inadvertent opening of a PWR pressurizer pressure relief valve	AOO	MARVEL-M	min	--	max feedback Figure 15.0-2	4466
15.6.2	Radiological consequences of the failure of small lines carrying primary coolant outside containment	AOO	RADTRAD	--	--	--	4540*3
15.6.3	Radiological consequences of SGTR	PA	MARVEL-M	min	--	max feedback Figure 15.0-2	4555*2
15.6.5	Loss-of-Coolant Accidents	PA	WCOBRA/TRAC, HOTSPOT	*4	--	*4	4466
			M-RELAP5	*5	--	*6	4555*2

Notes:
- *1 Steady state analysis
- *2 102% of 4466MW_t (NSSS thermal power)
- *3 102% of 4451MW_t (core thermal power)
- *4 Applicability confirmed (Ref.15.0-18).
- *5 Conservative Moderator Density Coefficient changes with moderator density assumed (Ref.15.0-20).
- *6 Conservative Doppler Temperature Coefficient changes with moderator density assumed (Ref.15.0-20).

Table 15.0-2
Nominal Power Conditions

Parameter	Value
Design core thermal power (MW_t)	4,451
RCP generated thermal power (MW_t)	15
Design NSSS thermal power (MW_t)	4,466

Table 15.0-3
Nominal Values of Plant Parameters

	RTDP with 10% SG Tube Plugging	Non-RTDP[*1] with 10% SG Tube Plugging
Thermal output of NSSS (MW_t)	4,466	4,466
Core inlet temperature (°F)	551.4	550.6
Vessel average temperature (°F)	583.8	583.8
RCS pressure (psia)	2,250	2,250
RC flow per loop (gpm)	115,000	112,000
Steam flow from NSSS (lbm/hr)	2.02 E+07	2.02 E+07
Steam pressure at SG (psia)	929	922
Assumed feedwater temperature at SG inlet (°F)	457	457
Average core heat flux (Btu/hr-ft^2)	0.162 E+06	0.162 E+06

Note:
*1 Steady-state errors as discussed in Section 15.0.0.2.2 are added to these values to obtain transient analyses initial conditions.

Table 15.0-4
Reactor Trip and ESF Actuation Analytical Limits and Time Delays Assumed for Transient Analyses

Protective Function Description	Analytical Limit	Time Delay (sec)
High Power Range Neutron Flux (low setpoint) Reactor Trip	35% RTP	0.6
High Power Range Neutron Flux (high setpoint) Reactor Trip	118% RTP	0.6
Over Temperature ΔT Reactor Trip	Variable (See Figure 15.0-1)	6.0
Over Power ΔT Reactor Trip	Variable (See Figure 15.0-1)	6.0
Low Reactor Coolant Flow Reactor Trip	87% of rated flow	1.8
Low Reactor Coolant Pump Speed Reactor Trip	95% of rated pump speed	0.6
High Pressurizer Pressure Reactor Trip	2425 psia	1.8
Low Pressurizer Pressure Reactor Trip	1860 psia	1.8
Low Steam Generator Water Level Reactor Trip	0% of narrow range level span	1.8
High-High Steam Generator Water Level Reactor Trip	75% of narrow range level span	1.8
High Pressurizer Water Level Reactor Trip	100% of level span	1.8
ECCS Signal Reactor Trip	--	3.3
ECCS Signal on Low Main Steam Line Pressure	500 psia	3.0
ECCS Signal on Low Pressurizer Pressure	1760 psia	3.0
ECCS Signal on High Containment Pressure	24.0 psia	3.0
Main Steam Line Isolation Signal on Low Steam Line Pressure	500 psia	3.0
Main Steam Line Isolation Signal on High-High Containment Pressure	39.9 psia	3.0
Containment Spray Signal on High-3 Containment Pressure	51.2 psia	3.0
Emergency Feedwater Actuation on Low Steam Generator Water Level	0% of narrow range level span	3.0
Emergency Feedwater Isolation Signal on Low Main Steam Line Pressure	500 psia	3.0
Emergency Feedwater Isolation Signal on High Steam Generator Water Level	55% of narrow range level span	3.0

Table 15.0-5
Mitigation System Time Delays

Component	Time Delay (sec)
Main feedwater isolation valve closure, main feedwater regulation valve closure	5
Main steam isolation valve closure	5
Main steam relief valve block valve closure	30
Emergency feedwater isolation valve closure	20
Emergency feedwater pump - initiation to full flow with offsite electrical power without offsite electrical power	 60 133[*1,*2]
Safety injection pump - initiation to full flow with offsite electrical power without offsite electrical power	 18[*2] 118[*1,*2]
Containment spray system initiation without offsite electrical power	243[*1]

Notes:
*1 including emergency standby gas turbine generator start and load delay (100 seconds)
*2 depending on the event, additional time margin may be taken into consideration

Table 15.0-6
Assumed Single Failures

Event Description	Assumed Failure
Feedwater temperature reduction	None-no mitigating systems required
Excessive feedwater flow	1 train RTS
Excessive steam flow	None-no mitigating systems required
Inadvertent secondary depressurization	1 train ECCS
Steam system piping failure	1 train ECCS (Hot standby), 1 train RTS (HFP)
Loss of external load	1 train RTS
Turbine trip	1 train RTS
Loss of condenser vacuum and other events resulting in turbine trip	1 train RTS
Inadvertent closure of main steam isolation valves	1 train RTS
Steam pressure regulator malfunction	Not applicable to US-APWR
Loss of AC power	1 train EFWS
Loss of normal feedwater	1 train EFWS
Feedwater system pipe break	1 train EFWS
Partial loss of forced coolant flow	1 train RTS
Complete loss of forced coolant flow	1 train RTS
Reactor coolant pump locked rotor	1 train RTS
Reactor coolant pump shaft break	1 train RTS
RCCA bank withdrawal from subcritical	1 train RTS
RCCA bank withdrawal at power	1 train RTS
Dropped RCCA, dropped RCCA bank	1 train RTS
Statically misaligned RCCA	No transient analysis
Single RCCA withdrawal	1 train RTS
Flow controller malfunction	Not applicable to US-APWR
Uncontrolled boron dilution	No transient analysis; Bounded by RCCA withdrawal at power
Improper fuel loading	No transient analysis
RCCA ejection	1 train RTS
Inadvertent ECCS operation at power	Not applicable to US-APWR (low shutoff head)
Increase in reactor coolant inventory (CVCS)	1 train RTS
Inadvertent RCS depressurization	1 train RTS
Failure of a small line carrying primary coolant outside containment	No transient analysis
Steam generator tube rupture	1 train EFWS
Spectrum of LOCA Small breaks Large breaks	 1 train emergency power source 1 train ECCS

Table 15.0-7
EAB and LPZ Accident Dose Criteria

Accident or Case	EAB and LPZ Dose Criteria (rem TEDE)	Analysis Release Duration
LOCA *2	25	30 days for containment and ECCS leakage
Steam generator tube rupture		Affected steam generator: time to isolate; Unaffected generator(s): until cold shutdown is established.
Fuel damage or pre-incident spike	25	
Coincident iodine spike	2.5	
Steam system piping failure		Until cold shutdown is established.
Fuel damage or pre-incident spike	25	
Coincident iodine spike	2.5	
RCP rotor seizure	2.5	Until cold shutdown is established.
Failure of small lines carrying primary coolant outside containment *1	2.5	45 minutes until operator action isolates the break.
Rod ejection accident	6.3	30 days for containment pathway; until cold shutdown is established for secondary pathway.
Fuel handling accident	6.3	2 hours

Notes:

*1 The acceptance criterion except for the failure of small lines carrying primary coolant outside containment is based on RG 1.183. The acceptance criterion for the failure of small lines carrying primary coolant outside containment is based on SRP 15.6.2.

*2 The acceptance criterion for the MCR is 5 rem.

Table 15.0-8
Fraction of Fission Product Inventory in Fuel Rod Gap Used in Non-LOCA Radiological Consequence Evaluations

Nuclide or Nuclide Group	Fraction of Inventory in Gap	
	Non-LOCA	Rod Ejection Accident
I-131	0.08	0.10
Kr-85	0.10	0.10
Other noble gases	0.05	0.10
Other halogens	0.05	0.10
Alkali metals	0.12	0.12
All other nuclides	0.00	0.00

Table 15.0-9
Radionuclide Groups

Radionuclide Group	Elements
Noble gases	Xe, Kr
Halogens	I, Br
Alkali metals	Cs, Rb
Tellurium group	Te, Sb, Se, Ba, Sr
Noble metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium group	Ce, Pu, Np

Table 15.0-10
Reactor Coolant Iodine Concentrations for 1 μCi/g and 60 μCi/g DE I-131

Nuclide	Reactor Coolant Concentration (μCi/g)	
	1 μCi/g DE I-131	60 μCi/g DE I-131
I-131	7.37×10^{-1}	4.42×10^{1}
I-132	3.95×10^{-1}	2.37×10^{1}
I-133	1.27×10^{0}	7.63×10^{1}
I-134	2.71×10^{-1}	1.63×10^{1}
I-135	8.34×10^{-1}	5.00×10^{1}

Table 15.0-11
Iodine Appearance Rates in the Reactor Coolant (Ci/min)

Nuclide	Appearance Rate due to Transient-initiated Spike (Ci/min)		
	Steam System Piping Failure	SGTR	Failure of Small Lines Carrying Primary Coolant Outside Containment
I-131	2.67×10^{2}	2.22×10^{2}	2.79×10^{2}
I-132	1.43×10^{2}	1.19×10^{2}	1.50×10^{2}
I-133	4.60×10^{2}	3.82×10^{2}	4.82×10^{2}
I-134	9.80×10^{1}	8.15×10^{1}	1.03×10^{2}
I-135	3.01×10^{2}	2.51×10^{2}	3.16×10^{2}

Table 15.0-12
Reactor Coolant Noble Gases Concentration for 300 μCi/g DE Xe-133

Nuclide	μCi/g
Kr-85	4.22×10^{1}
Kr-85m	8.15×10^{-1}
Kr-87	5.30×10^{-1}
Kr-88	1.52×10^{0}
Xe-133	1.43×10^{2}
Xe-135	4.69×10^{0}

Table 15.0-13
Offsite χ/Q *1 and Breathing Rate

Location/Time Interval	χ/Q (s/m^3)	Breathing Rate (m^3/s)
EAB		
Limiting 2 hours	5.0 x10^{-4}	3.5 x10^{-4}
LPZ outer boundary		
0 to 8 h	2.1 x10^{-4}	3.5 x10^{-4}
8 to 24 h	1.3 x10^{-4}	1.8 x10^{-4}
24 to 96 h	6.9 x10^{-5}	2.3 x10^{-4}
96 to 720 h	2.8 x10^{-5}	2.3 x10^{-4}

Note:
*1 The χ/Q values for various intake locations for the MCR are described in Appendix 15A.1.5.

Table 15.0-14
Reactor Fission Product Nuclide Inventory and Related Parameters (Sheet 1 of 4)

Nuclide	Half Life	Inventory (Ci) *1	CEDE Dose Conversion Factor *2 (Sv/Bq)	EDE Dose Conversion Factor *2 ($Sv\text{-}m^3$ / Bq- s)
Noble Gases				
Kr-85	10.72y	$1.73x10^{6}$	—	$1.19x10^{-16}$
Kr-85 m	4.48h	$4.83x10^{7}$	—	$7.48x10^{-15}$
Kr-87	76.3m	$9.59x10^{7}$	—	$4.12x10^{-14}$
Kr-88	2.84h	$1.35x10^{8}$	—	$1.02x10^{-13}$
Xe-133	5.245d	$2.99x10^{8}$	—	$1.56x10^{-15}$
Xe-135	9.09h	$9.14x10^{7}$	—	$1.19x10^{-14}$
Iodines				
I-131	8.04d	$1.44x10^{8}$	$8.89x10^{-9}$	$1.82x10^{-14}$
I-132	2.30h	$2.08x10^{8}$	$1.03x10^{-10}$	$1.12x10^{-13}$
I-133	20.8h	$3.00x10^{8}$	$1.58x10^{-9}$	$2.94x10^{-14}$
I-134	52.6m	$3.35x10^{8}$	$3.55x10^{-11}$	$1.30x10^{-13}$
I-135	6.61h	$2.80x10^{8}$	$3.32x10^{-10}$	$8.29x10^{-14}$
Alkali Metals				
Rb-86	18.66d	$3.40x10^{5}$	$1.79x10^{-9}$	$4.81x10^{-15}$
Cs-134	2.062y	$3.39x10^{7}$	$1.25x10^{-8}$	$7.57x10^{-14}$
Cs-136	13.1d	$9.23x10^{6}$	$1.98x10^{-9}$	$1.06x10^{-13}$
Cs-137	30.0y	$1.93x10^{7}$	$8.63x10^{-9}$	$2.73x10^{-14}$

Notes:
*1 These inventories are assumed to be 32.1 MW/MTU of specific power, 28 months as irradiation time for a cycle (these conditions are equivalent to about 55 GWD/MTU.), and 102% of the design core thermal power.

*2 These conversion factors are listed in NUREG/CR-6604 (Ref. 15.0-16). NUREG/CR-6604 is based on Federal Guidance Report 11 and 12.

Table 15.0-14
Reactor Fission Product Nuclide Inventory and Related Parameters (Sheet 2 of 4)

Nuclide	Half Life	Inventory (Ci) *1	CEDE Dose Conversion Factor *2 (Sv/Bq)	EDE Dose Conversion Factor *2 ($Sv\text{-}m^3$ / Bq- s)
Tellurium Group				
Sb-127	3.85d	$1.50x10^{7}$	$1.63x10^{-9}$	$3.33x10^{-14}$
Sb-129	4.32h	$4.54x10^{7}$	$1.74x10^{-10}$	$7.14x10^{-14}$
Te-127	9.35h	$1.48x10^{7}$	$8.60x10^{-11}$	$2.42x10^{-16}$
Te-127m	109d	$1.95x10^{6}$	$5.81x10^{-9}$	$1.47x10^{-16}$
Te-129	69.6m	$4.47x10^{7}$	$2.09x10^{-11}$	$2.75x10^{-15}$
Te-129m	33.6d	$6.69x10^{6}$	$6.48x10^{-9}$	$3.34x10^{-15}$
Te-131m	30h	$2.06x10^{7}$	$1.76x10^{-9}$	$7.46x10^{-14}$
Te-132	78.2h	$2.05x10^{8}$	$2.55x10^{-9}$	$1.03x10^{-14}$
Strontium and Barium				
Sr-89	50.5d	$1.67x10^{8}$	$1.12x10^{-8}$	$7.73x10^{-17}$
Sr-90	29.12y	$1.39x10^{7}$	$3.51x10^{-7}$	$7.53x10^{-18}$
Sr-91	9.5h	$2.21x10^{8}$	$4.55x10^{-10}$	$4.92x10^{-14}$
Sr-92	2.71h	$2.32x10^{8}$	$2.18x10^{-10}$	$6.79x10^{-14}$
Ba-139	82.7m	$2.78x10^{8}$	$4.64x10^{-11}$	$2.17x10^{-15}$
Ba-140	12.74d	$2.66x10^{8}$	$1.01x10^{-9}$	$8.58x10^{-15}$

Notes:
*1 These inventories are assumed to be 32.1 MW/MTU of specific power, 28 months as irradiation time for a cycle (these conditions are equivalent to about 55 GWD/MTU.), and 102% of the design core thermal power.

*2 These conversion factors are listed in NUREG/CR-6604 (Ref. 15.0-16). NUREG/CR-6604 is based on Federal Guidance Report 11 and 12.

Table 15.0-14
Reactor Fission Product Nuclide Inventory and Related Parameters (Sheet 3 of 4)

Nuclide	Half Life	Inventory (Ci) *1	CEDE Dose Conversion Factor*2 (Sv/Bq)	EDE Dose Conversion Factor *2 ($Sv\text{-}m^3$ / Bq- s)
Noble Metals				
Co-58	70.80d	$0.00x10^{0}$	$2.94x10^{-9}$	$4.76x10^{-14}$
Co-60	5.271y	$4.35x10^{5}$	$5.91x10^{-8}$	$1.26x10^{-13}$
Mo-99	66.0h	$2.72x10^{8}$	$1.07x10^{-9}$	$7.28x10^{-15}$
Tc-99m	6.02h	$2.38x10^{8}$	$8.80x10^{-12}$	$5.89x10^{-15}$
Ru-103	39.28d	$2.15x10^{8}$	$2.42x10^{-9}$	$2.25x10^{-14}$
Ru-105	4.44h	$1.41x10^{8}$	$1.23x10^{-10}$	$3.81x10^{-14}$
Ru-106	368.2d	$7.53x10^{7}$	$1.29x10^{-7}$	$1.04x10^{-14}$
Rh-105	35.36h	$1.31x10^{8}$	$2.58x10^{-10}$	$3.72x10^{-15}$
Cerium Group				
Ce-141	32.501d	$2.51x10^{8}$	$2.42x10^{-9}$	$3.43x10^{-15}$
Ce-143	33.0h	$2.45x10^{8}$	$9.16x10^{-10}$	$1.29x10^{-14}$
Ce-144	284.3d	$1.90x10^{8}$	$1.01x10^{-7}$	$2.77x10^{-15}$
Np-239	2.355d	$2.71x10^{9}$	$6.78x10^{-10}$	$7.69x10^{-15}$
Pu-238	87.74y	$7.47x10^{5}$	$7.79x10^{-5}$	$4.88x10^{-18}$
Pu-239	24065y	$5.64x10^{4}$	$8.33x10^{-5}$	$4.24x10^{-18}$
Pu-240	6537y	$8.85x10^{4}$	$8.33x10^{-5}$	$4.75x10^{-18}$
Pu-241	14.4y	$1.96x10^{7}$	$1.34x10^{-6}$	$7.25x10^{-20}$

Notes:
*1 These inventories are assumed to be 32.1 MW/MTU of specific power, 28 months as irradiation time for a cycle (these conditions are equivalent to about 55 GWD/MTU.), and 102% of the design core thermal power.

*2 These conversion factors are listed in NUREG/CR-6604 (Ref. 15.0-16). NUREG/CR-6604 is based on Federal Guidance Report 11 and 12.

Table 15.0-14
Reactor Fission Product Nuclide Inventory and Related Parameters (Sheet 4 of 4)

Nuclide	Half Life	Inventory (Ci) *1	CEDE Dose Conversion Factor *2 (Sv/Bq)	EDE Dose Conversion Factor *2 ($Sv\text{-}m^3$ / Bq- s)
Lanthanides				
Y-90	64.0h	$1.46x10^{7}$	$2.28x10^{-9}$	$1.90x10^{-16}$
Y-91	58.51d	$2.09x10^{8}$	$1.32x10^{-8}$	$2.60x10^{-16}$
Y-92	3.54h	$2.32x10^{8}$	$2.11x10^{-10}$	$1.30x10^{-14}$
Y-93	10.1h	$2.57x10^{8}$	$5.82x10^{-10}$	$4.80x10^{-15}$
Zr-95	63.98d	$2.66x10^{8}$	$6.39x10^{-9}$	$3.60x10^{-14}$
Zr-97	16.90h	$2.66x10^{8}$	$1.17x10^{-9}$	$4.43x10^{-14}$
Nb-95	35.15d	$2.68x10^{8}$	$1.57x10^{-9}$	$3.74x10^{-14}$
La-140	40.272h	$2.70x10^{8}$	$1.31x10^{-9}$	$1.17x10^{-13}$
La-141	3.93h	$2.54x10^{8}$	$1.57x10^{-10}$	$2.39x10^{-15}$
La-142	92.5m	$2.50x10^{8}$	$6.84x10^{-11}$	$1.44x10^{-13}$
Pr-143	13.56d	$2.37x10^{8}$	$2.19x10^{-9}$	$2.10x10^{-17}$
Nd-147	10.98d	$9.94x10^{7}$	$1.85x10^{-9}$	$6.19x10^{-15}$
Am-241	432.2y	$2.64x10^{4}$	$1.20x10^{-4}$	$8.18x10^{-16}$
Cm-242	162.8d	$6.55x10^{6}$	$4.67x10^{-6}$	$5.69x10^{-18}$
Cm-244	18.11y	$7.96x10^{5}$	$6.70x10^{-5}$	$4.91x10^{-18}$

Notes:
*1 These inventories are assumed to be 32.1 MW/MTU of specific power, 28 months as irradiation time for a cycle (these conditions are equivalent to about 55 GWD/MTU.), and 102% of the design core thermal power.

*2 These conversion factors are listed in NUREG/CR-6604 (Ref. 15.0-16). NUREG/CR-6604 is based on Federal Guidance Report 11 and 12.

Table 15.0-15
Fraction of Fission Product Inventory for Fuel Rod Gap and Early In-Vessel Release Phases Used in LOCA Radiological Consequence Evaluations

Nuclide Group	Inventory Fraction in Each Release Phase	
	Gap Release	Early In-Vessel Release
Noble gases	0.05	0.95
Halogens	0.05	0.35
Alkali metals	0.05	0.25
Tellurium metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble metals	0.00	0.0025
Cerium group	0.00	0.0005
Lanthanides	0.00	0.0002

Table 15.0-16
Time of Onset and Duration for Fuel Rod Gap and Early In-Vessel Release Phases Used in LOCA Radiological Consequence Evaluations

Phase	Onset (time after initiation of accident)	Phase Duration
Gap release	30 sec	0.5 hr
Early in-vessel release	0.5 hr	1.3 hr

Table 15.0-17
Summary of Calculated Doses for Events with a Radiological Release

Accident or Case	EAB and LPZ Dose Criteria (rem TEDE)	Calculated EAB Dose (rem TEDE)	Calculated LPZ Dose (rem TEDE)
LOCA [*2]	25	13	13
Steam generator tube rupture			
Fuel damage or pre-incident spike	25	3.6	1.5
Coincident iodine spike	2.5	0.96	0.43
Steam system piping failure			
Fuel damage or pre-incident spike	25	0.19	0.11
Coincident iodine spike	2.5	0.32	0.28
RCP rotor seizure	2.5	0.49	0.70
Failure of small lines carrying primary coolant outside containment [*1]	2.5	1.5	0.60
Rod ejection accident	6.3	5.1	4.5
Fuel handling accident	6.3	3.3	1.4

Notes:

*1 The acceptance criterion except for the failure of small lines carrying primary coolant outside containment is based on RG 1.183. The acceptance criterion for the failure of small lines carrying primary coolant outside containment is based on SRP 15.6.2.

*2 Calculated MCR dose for the LOCA is 4.5 rem. (See Section 15.6.5.)

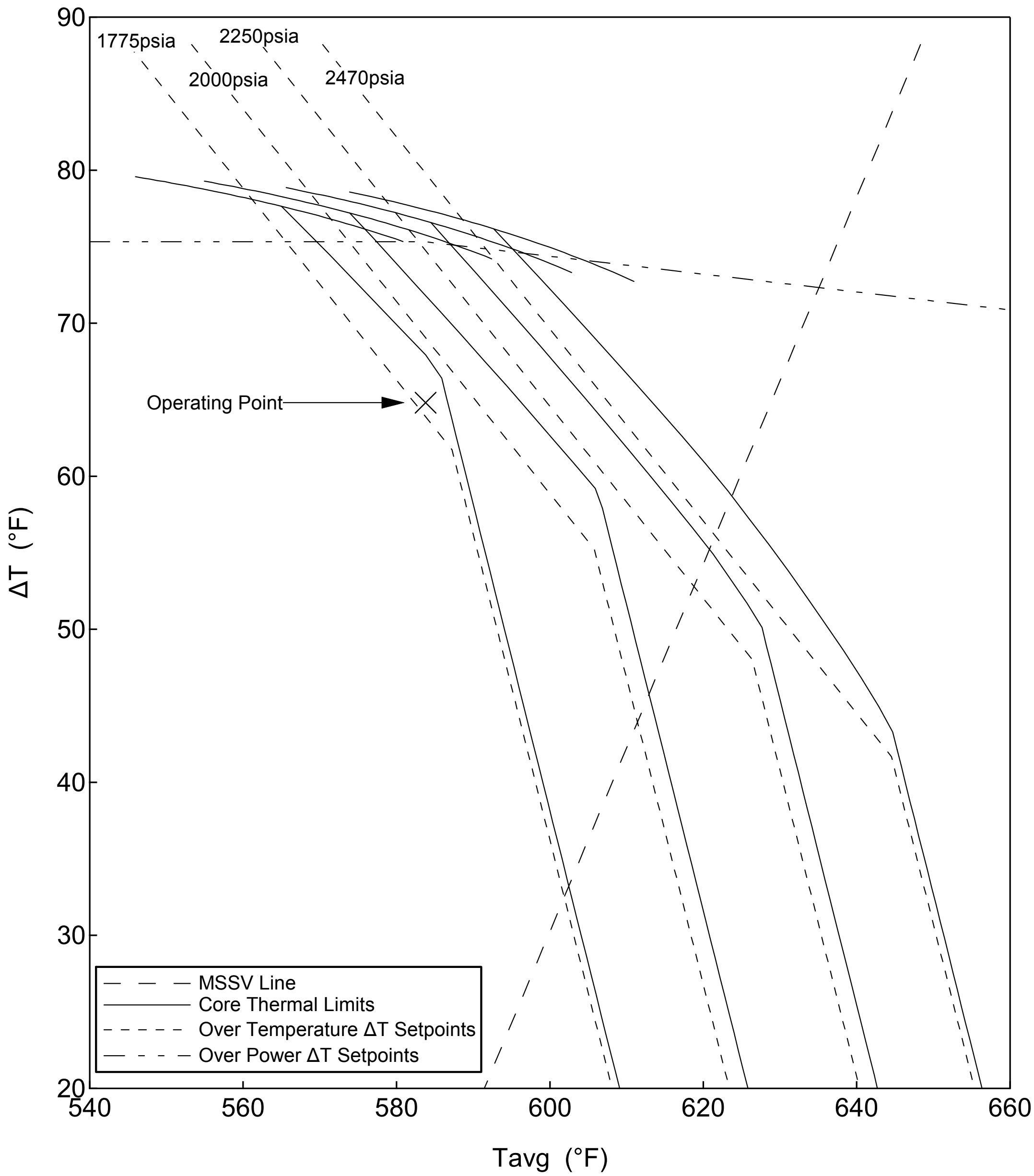


Figure 15.0-1 Over Power and Over Temperature ΔT (Allowed Operating Space)

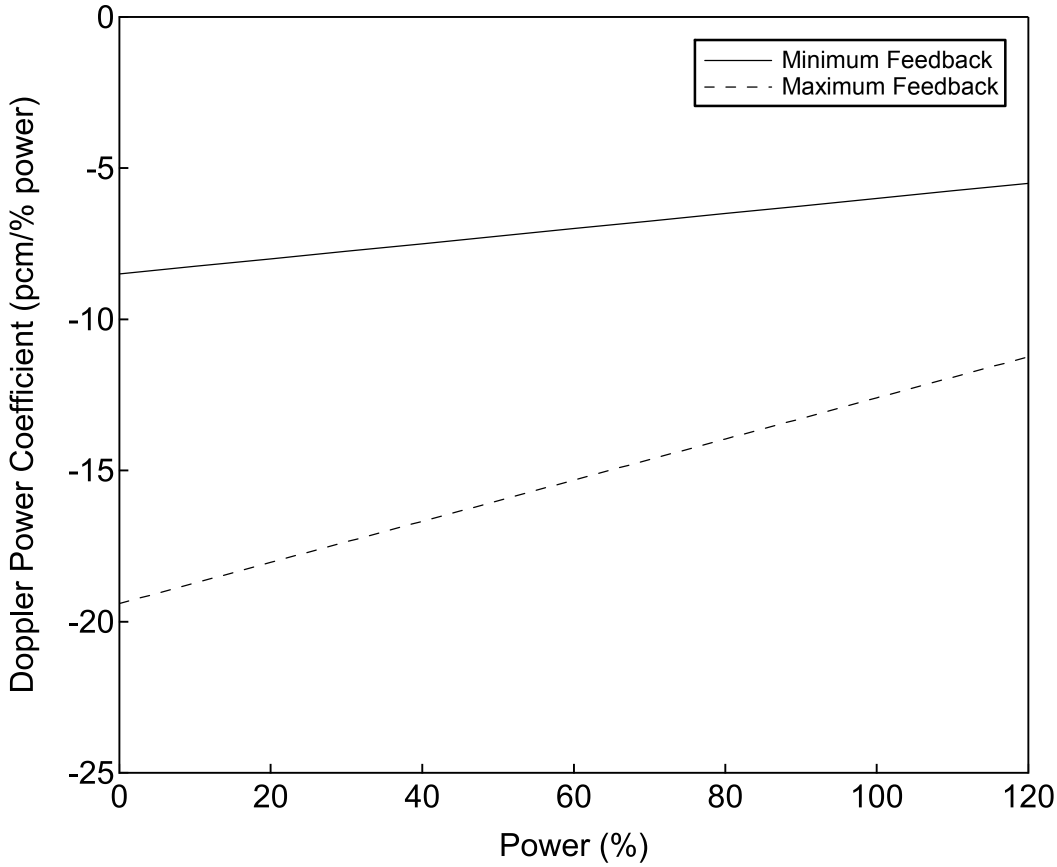


Figure 15.0-2 Doppler Power Coefficient

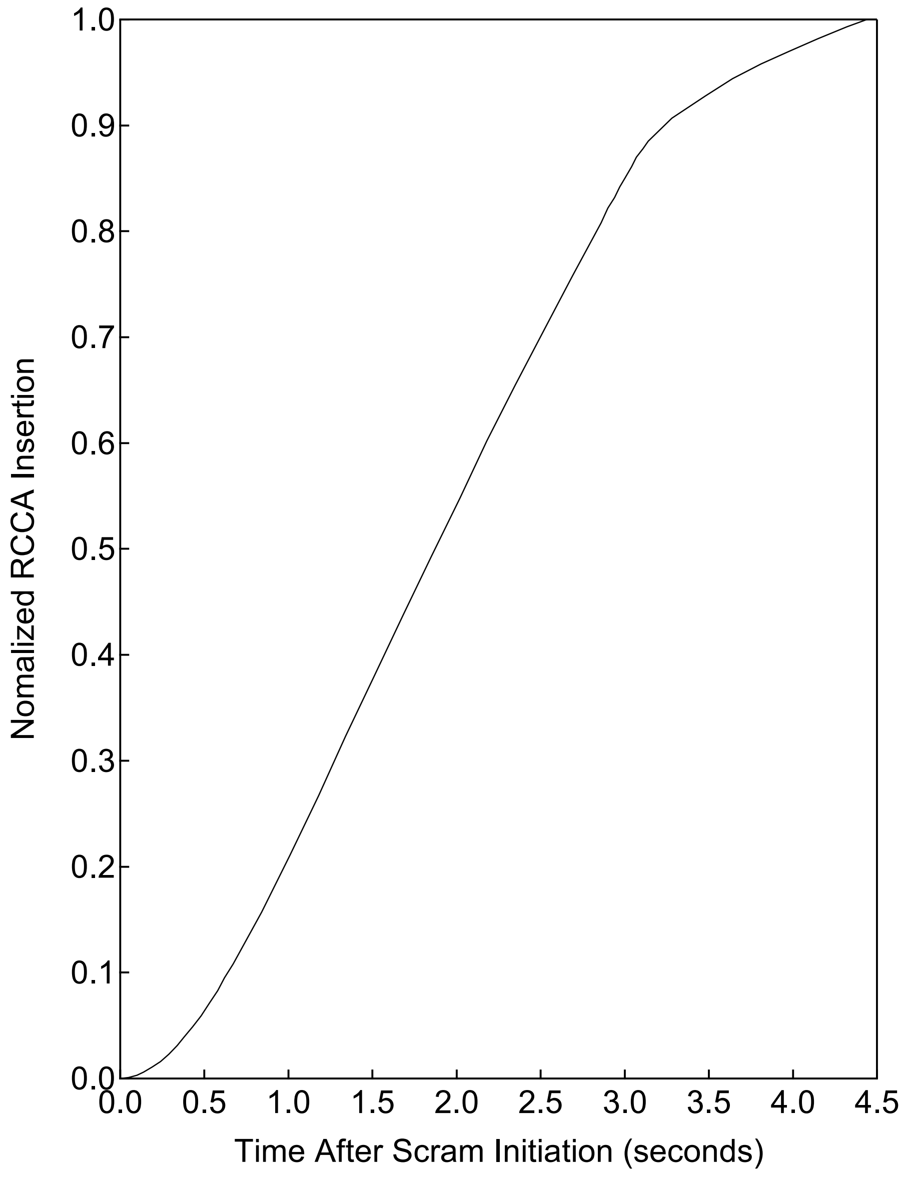


Figure 15.0-3 RCCA Displacement versus Time following Reactor Trip

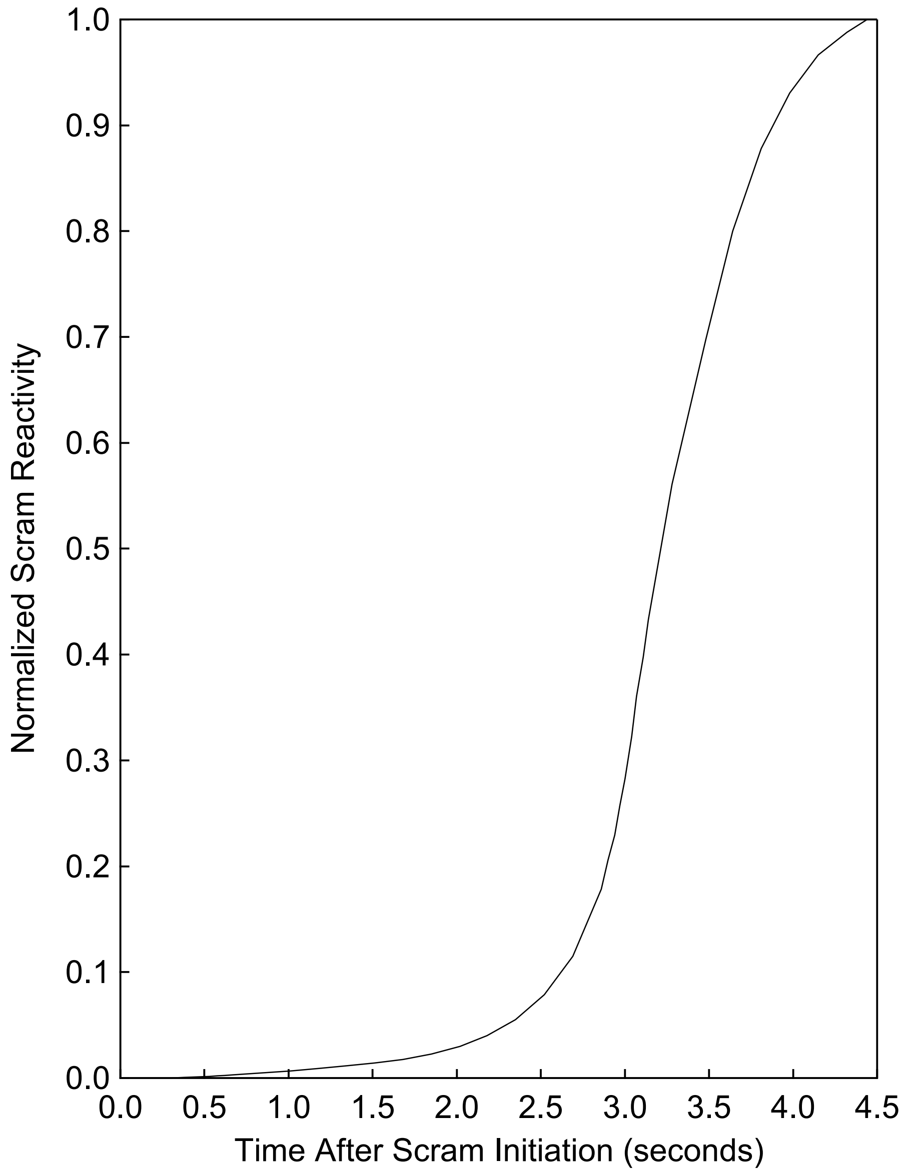


Figure 15.0-4 Negative Reactivity versus Time following Reactor Trip

15.1 Increase in Heat Removal by the Secondary System

This section describes analyses that have been performed for events that could result in an increase in the rate of heat removal by the secondary system, which, in turn, could lead to a temperature decrease in the reactor coolant system.

Analyses of the following events are described in this section:

- Decrease in Feedwater Temperature as a Result of Feedwater System Malfunctions
- Increase in Feedwater Flow as a Result of Feedwater System Malfunctions
- Increase in Steam Flow as a Result of Steam Pressure Regulator Malfunction
- Inadvertent Opening of a Steam Generator Relief or Safety Valve
- Steam System Piping Failure Inside and Outside of Containment

These events are considered anticipated operational occurrences (AOOs) as defined in Section 15.0.0.1, except for the double-ended steam system pipe break which is classified as a postulated accident (PA).

For the US-APWR, the core analysis does not predict fuel failures for any of these events. The radiological consequences for the events in this section are bounded by the radiological consequences of the large double-ended rupture of a main steam line (See Section 15.1.5).

15.1.1 Decrease in Feedwater Temperature as a Result of Feedwater System Malfunctions

15.1.1.1 Identification of Causes and Frequency Classification

A decrease in feedwater temperature causes a reduction in steam generator secondary temperature, resulting in an increase in primary-to-secondary heat transfer. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase. Core conditions that could approach the core DNBR or over power limits are protected by the over temperature and over power ΔT and high power range neutron flux (high setpoint) reactor trips. If there is no protection system action, the reactor reaches a new equilibrium condition at a power level corresponding to the steam generator heat removal rate, characterized by the primary-to-secondary ΔT.

A feedwater temperature decrease can be caused by the functional loss of either the high-pressure or low-pressure feedwater heaters.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.1-1). Event frequency conditions are described in Section 15.0.0.1.

15.1.1.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the decrease in feedwater temperature event is described in the results section.

A decrease in feedwater temperature will result in an increase of the heat transfer from the primary to the secondary side of the steam generator, causing a reduction in the reactor coolant temperature at the reactor vessel inlet. This leads to the introduction of cooler (more dense) water into the core, which adds reactivity as a result of the positive moderator density coefficient (negative moderator temperature coefficient), thereby increasing the reactor power. For transients of this type initiated at power, the reactor will stabilize at a new, higher, power level.

The decrease in average temperature can also cause rod cluster control assemblies (RCCAs) to withdraw if the reactor is operating in automatic rod control mode, as the system attempts to restore the selected core average temperature input into the automatic rod control system. Although, this RCCA motion could contribute to the overall increase in core power, the analysis of the 10% steam flow increase event in Section 15.1.3 demonstrates there is no difference in the results for the maximum negative temperature coefficient cases with and without automatic rod control. Therefore, only the manual rod control case is presented for this event.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

The reactor trip system (RTS) is not actuated and no systems are required to mitigate this event. Therefore, no single failures are identified or assumed in the analysis of this event.

Normal reactor control systems and engineered safety systems are not required to function and are not credited in the evaluation.

The following automatic reactor trip signals are assumed to be available to provide protection from this transient, although they are not credited in the analysis:

- High power range neutron flux (high setpoint)
- Over temperature ΔT
- Over power ΔT

For this analysis, the RTS does not actuate and a turbine trip is not predicted. As a result, a loss of offsite power as a consequence of a turbine trip, as discussed in Section 15.0.0.7, is not applicable for this event. If a turbine trip were to occur, the post-trip response to the loss of offsite power would not be limiting as explained in Section 15.0.0.7.

15.1.1.3 Core and System Performance

15.1.1.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of steam generator temperature and pressure to changes in feedwater and steam flow conditions, steam generator heat transfer, primary temperatures and pressure, reactivity, reactor power, and other parameters for uniform or non-uniform cooldown events initiated by the secondary system such as a decrease in feedwater temperature in all steam generators. MARVEL-M also has the ability to internally calculate DNBRs for transients with full reactor coolant system (RCS) flow and power distributions bounded by the normal design distributions. This simplified model calculates DNBRs based on user-input values of DNBR at known combinations of core inlet temperature, pressure, and power. The MARVEL-M evaluation model is described in Section 15.0.2.2.1. Additional information on the use of the MARVEL-M code for analyzing non-LOCA events can be found in Reference 15.1-2.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

Under no-load conditions, the rate of energy change is reduced as load and feedwater flow decreases, making the no-load case less severe than the full power case. Therefore, the limiting case for hot full power operation is the only case analyzed for this initiating event.

15.1.1.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for a decrease in feedwater temperature event:

- Consistent with use of the RTDP, the initial values of reactor power, reactor coolant average temperature, and RCS pressure are assumed to be the nominal values without uncertainties as defined in Table 15.0-3.

- The moderator density coefficient is assumed to have the maximum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to be the minimum feedback limit shown in Figure 15.0-2. This combination gives the greatest positive reactivity and maximum power increase. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- No credit is taken for the heat capacity of the reactor coolant system and steam generator thick metal in attenuating the resulting plant cooldown.

- To initiate the transient, it is assumed that the function of one high-pressure feedwater heater is instantaneously lost, which causes a reduction in feedwater temperature in all steam generators.

- Conservative axial power profile and radial power distributions are assumed in the analysis as described in Section 15.0.0.2.3.

15.1.1.3.3 Results

For the US-APWR plant design, the limiting reduction in feedwater temperature is 55°F caused by the loss of function of a high-pressure heater. This temperature reduction causes an increase in heat removal equivalent to a plant load increase of 9%.

Table 15.1.1-1 lists the key events and the times at which they occur, relative to the initiation of the postulated feedwater temperature decrease at hot full power transient. Figures 15.1.1-1 through 15.1.1-6 are plots of key system parameters versus time from the core response analysis for this event.

The reactor coolant temperature decreases when cooler feedwater is supplied to the secondary side of a steam generator. This increases the reactor power because of the positive reactivity feedback effect of the moderator density coefficient.

As shown in Figure 15.1.1-1 the reactor power remains well below 118% of nominal (the high power range neutron flux (high setpoint) trip analytical limit) and a new equilibrium condition is reached 75 seconds after the initiation of the event. Since the heat flux will lag behind a reactor neutron power change due to the fuel rod thermal time constant, the peak heat flux shown in Figure 15.1.1-2 is also well below this 118% value. Figure 15.1.1-6 confirms that the transient resulting from the decrease in feedwater temperature does not cause the minimum DNBR to decrease below the 95/95 limit.

The normalized core average heat flux transient is virtually identical to the normalized maximum heat flux used in the DNBR calculations, and is considered representative of both parameters for this event. Similarly, reactor vessel inlet temperature is provided in place of inlet coolant temperature. Because there is significant subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided; these are not key parameters for this event. Steam line pressure is not a key parameter for heat removal events where steam generator pressures are stable or decreasing. Containment parameters and pressurizer safety valve flow are not reported for this event because RCS pressure remains below the pressurizer safety valve set pressure and there are no releases from the RCS or steam generators inside containment.

15.1.1.4 Barrier Performance

The decrease in feedwater temperature event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The results of the Core and System Evaluation case demonstrate that the RCS pressure remains well below 110% of system design pressure. In addition, the main steam pressure cannot challenge the main steam system pressure design limit, as discussed in Section 15.1.1.3.3. Therefore, the integrity of the RCS pressure boundary and the main steam system pressure boundary are maintained.

15.1.1.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.1.1.6 Conclusions

A reduction in the temperature of the feedwater supply to the secondary side of the steam generator will cause cooler water to enter the core, thus increasing reactor power. However, the resulting transient does not cause the minimum DNBR to decrease below the 95/95 limit, and no fuel failures are predicted.

In addition, the RCS pressure and steam system pressure remain well below 110% of their system design pressures, so the integrity of the reactor coolant pressure boundary and the integrity of the main steam system pressure boundary are maintained.

The decrease in feedwater temperature event does not lead to a more serious fault condition.

Table 15.1.1-1
Time Sequence of Events for Decrease in Feedwater Temperature

Event Description	Time [sec]
High-pressure feedwater heater function is lost	0
Approximate time equilibrium condition is reached	75

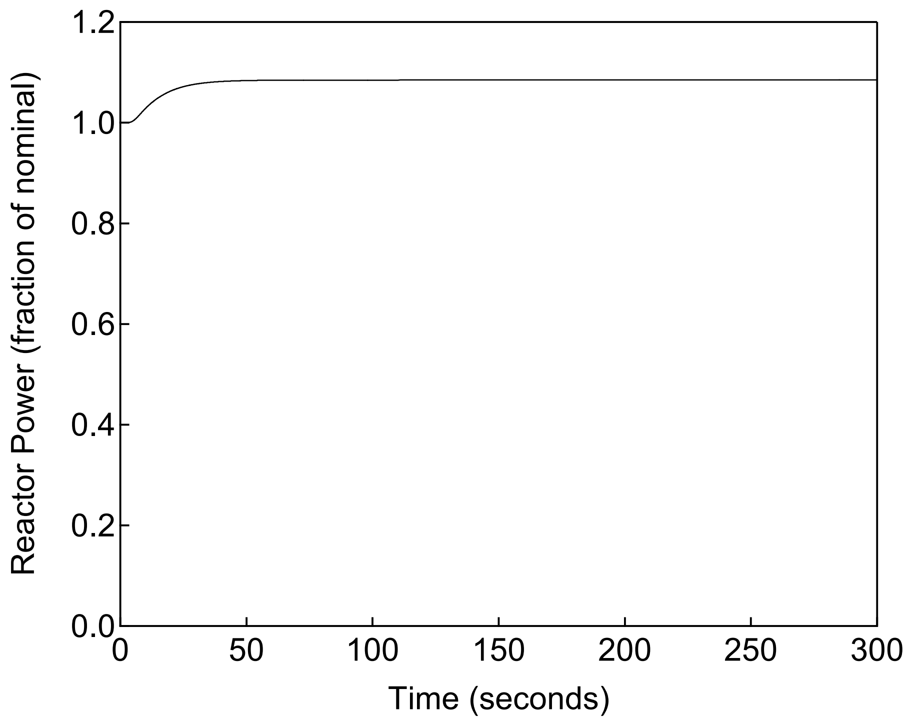


Figure 15.1.1-1 Reactor Power versus Time

Decrease in Feedwater Temperature

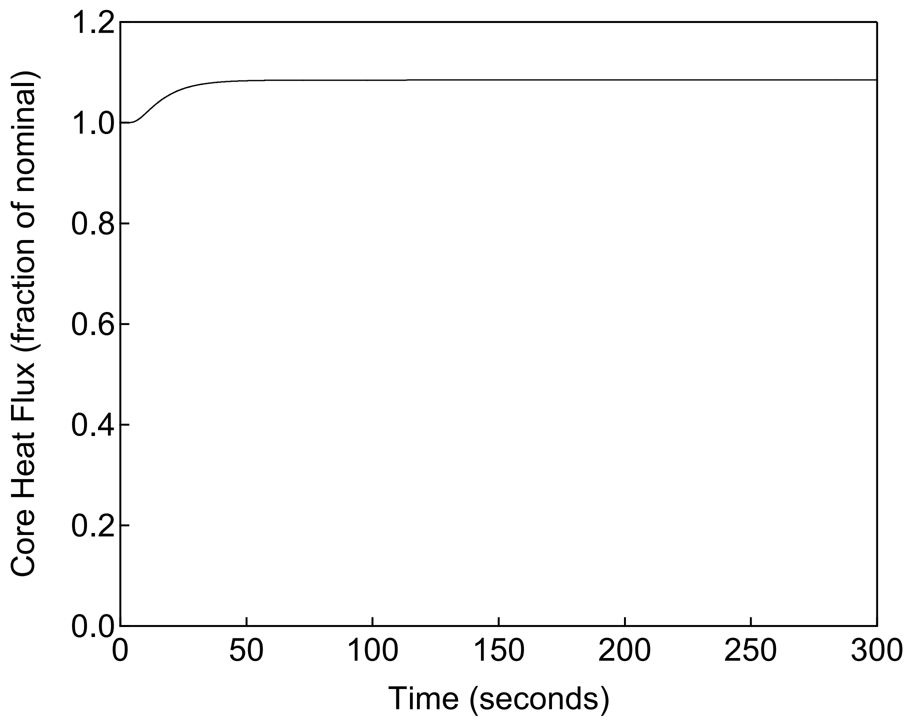


Figure 15.1.1-2 Core Heat Flux versus Time

Decrease in Feedwater Temperature

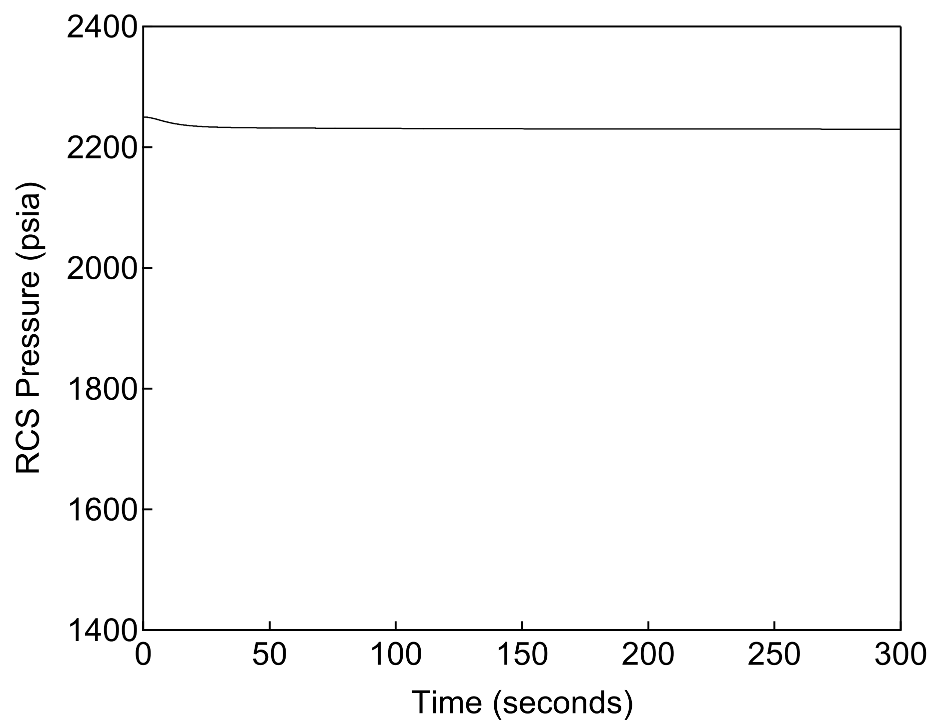


Figure 15.1.1-3 RCS Pressure versus Time

Decrease in Feedwater Temperature

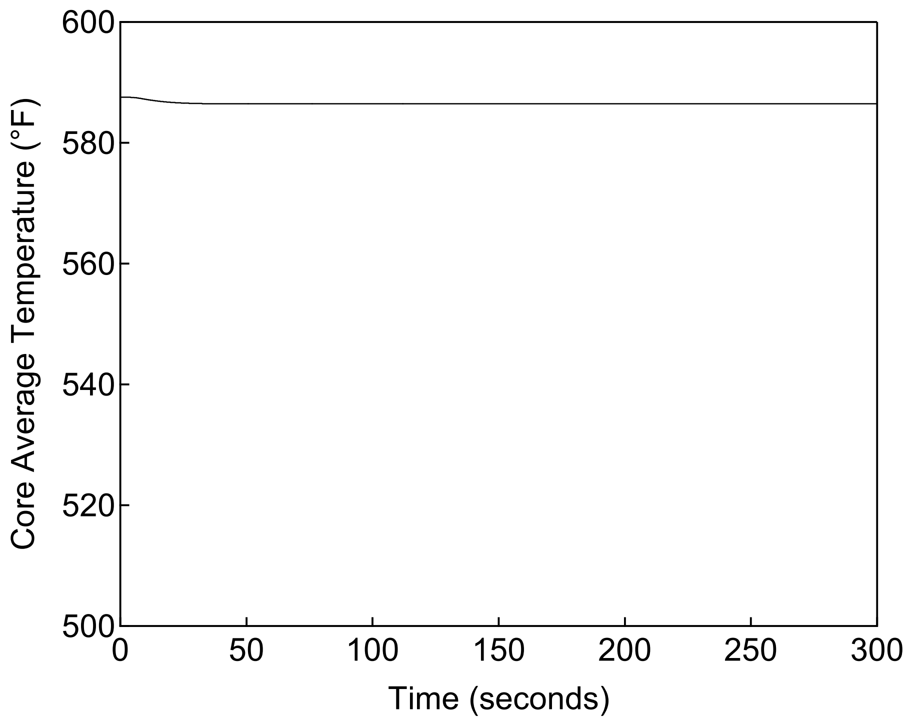


Figure 15.1.1-4 Core Average Temperature versus Time

Decrease in Feedwater Temperature

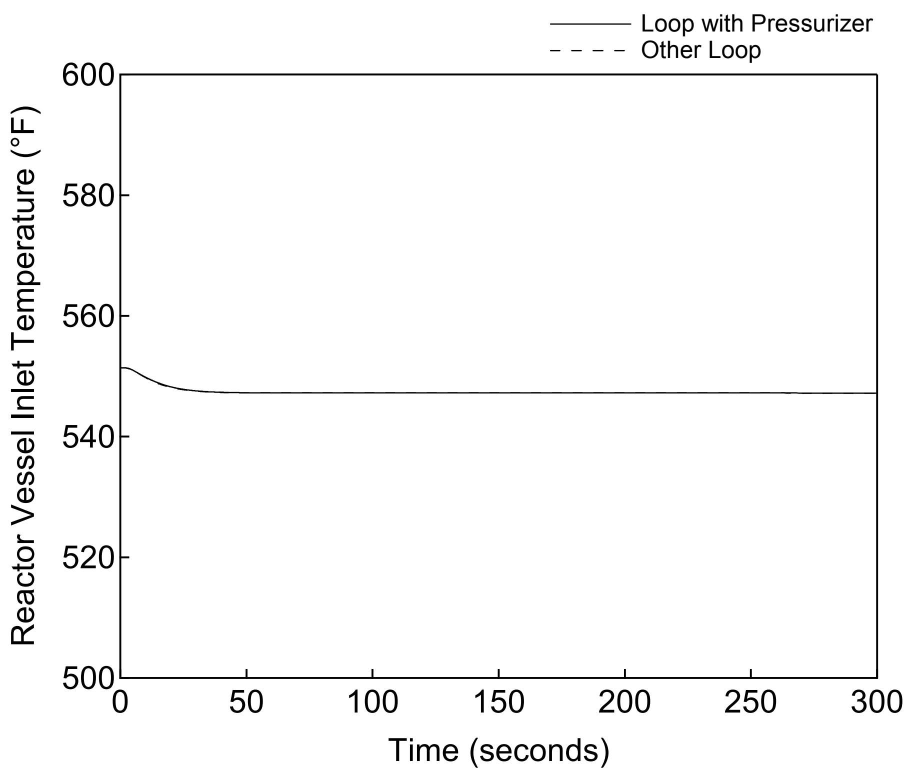


Figure 15.1.1-5 **Reactor Vessel Inlet Temperature versus Time**

Decrease in Feedwater Temperature

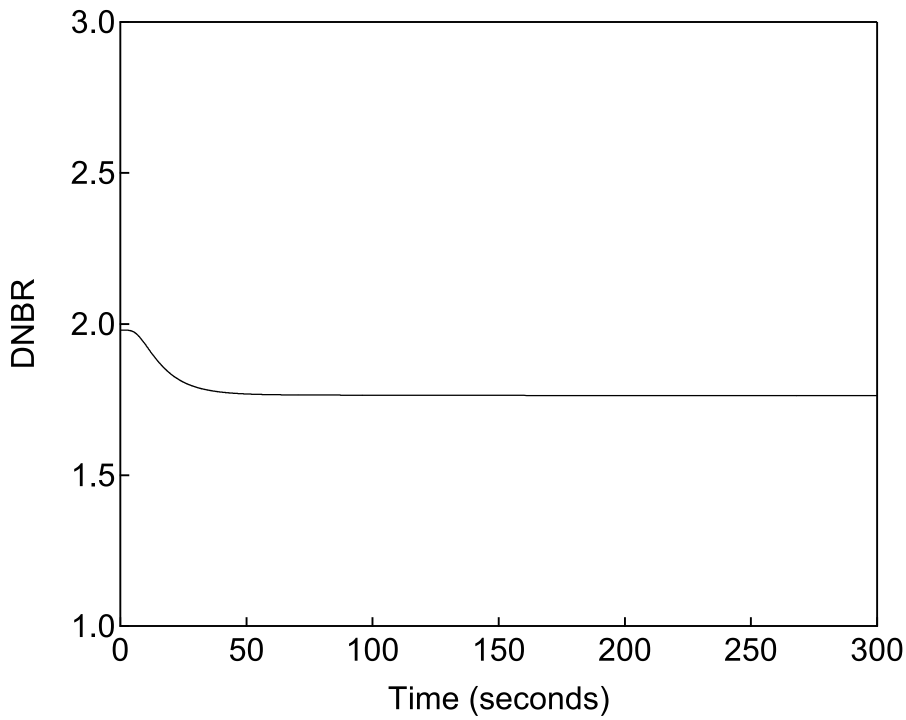


Figure 15.1.1-6 DNBR versus Time

Decrease in Feedwater Temperature

15.1.2 Increase in Feedwater Flow as a Result of Feedwater System Malfunctions

15.1.2.1 Identification of Causes and Frequency Classification

An increase in the feedwater flow rate to the secondary side of the steam generator will increase the heat transfer from the primary to the secondary side of the steam generator. This will cause a reduction in the reactor coolant temperature at the reactor vessel inlet. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase.

This transient is caused by a main feedwater regulation valve that is opened fully due to an operator error or a malfunction of the feedwater control system during rated power or part load operation. The feedwater control system is designed such that a single control system failure will affect only one main feedwater regulation valve, and hence, one steam generator.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.1-1). Event frequency conditions are described in Section 15.0.0.1.

MHI conservatively adopts an additional acceptance criterion to not allow steam generator overfill.

15.1.2.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the increase in feedwater flow event is described in the results section.

An increased water supply rate to the secondary side of the steam generator will increase the heat transfer from the primary to the secondary side of the steam generator, causing a reduction in the reactor coolant temperature at the reactor vessel inlet. This leads to the introduction of cooler (more dense) water into the core, which adds reactivity as a result of the positive moderator density coefficient (negative moderator temperature coefficient), thereby increasing the reactor power. For transients of this type initiated at power, the reactor will stabilize at a new, higher, power level.

The decrease in average temperature can also cause rod cluster control assemblies (RCCAs) to withdraw if the reactor is operating in automatic rod control mode, as the system attempts to restore the selected core average temperature input into the automatic rod control system. Although, this RCCA motion could contribute to the overall increase in core power, the analysis of the 10% steam flow increase event in Section 15.1.3 demonstrates there is no difference in the results for the maximum negative temperature coefficient cases with and without automatic rod control. Therefore, only the manual rod control case is presented for this event.

The temperature decrease, and associated density change, leads to a decrease in reactor coolant system (RCS) pressure. The combination of higher core power and lower RCS pressure can lead to a lower departure from nucleate boiling ratio (DNBR).

The main feedwater regulation valves are designed to have a maximum capacity, which is selected so that transients caused by a malfunction that results in them becoming fully open will have a limited effect. In addition, the feedwater control system is designed so that no single failure will cause two or more valves to open fully at the same time.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause a reactor coolant pump (RCP) coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

During normal operation, the steam generator level, main steam flow, and other indications in the main control room are monitored regularly. If the steam generator level rises abnormally, the high-high steam generator water level signal alarms in the main control room.

The high-high steam generator water level signal automatically trips the turbine, trips the feedwater pumps, closes all the main feedwater regulation valves, main feedwater bypass regulation valves, steam generator water filling control valves, and main feedwater isolation valves. The feedwater isolation functions are part of the excessive cooldown protection logic.

Normal reactor control systems and engineered safety systems are not required to function and are not credited in the evaluation.

The following signals are assumed to be available to trip the reactor and therefore provide protection from this transient:

- High power range neutron flux (high setpoint)
- Over temperature ΔT
- Over power ΔT
- High-high steam generator water level

15.1.2.3 Core and System Performance

15.1.2.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of steam generator temperature and pressure to changes in feedwater and steam flow conditions, steam generator heat transfer, primary temperatures and pressure, reactivity, reactor power, and other parameters for uniform or non-uniform cooldown events initiated by the secondary system such as an increase in feedwater flow to one steam generator. A non-perfect mixing model is used for the reactor vessel inlet plenum for the purpose of conservatively predicting reactivity for the non-uniform core inlet conditions caused by this event. MARVEL-M also has the ability to internally calculate DNBRs for transients with full RCS flow and power distributions bounded by the normal design distributions. This simplified model calculates DNBRs based on user-input values of DNBR at known combinations of core inlet temperature, pressure, and power. The MARVEL-M evaluation model is described in Section 15.0.2.2.1. Additional information on the use of the MARVEL-M code for analyzing non-LOCA events can be found in Reference 15.1-2.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

For the analyzed increase in feedwater flow from zero power conditions, the maximum reactivity addition rate and minimum DNBR are less severe than for the Uncontrolled Control Rod Assembly Withdrawal from Subcritical event that is described in Section 15.4.1. The detailed results of the zero power feedwater flow increase transient are not presented here because they are bounded by the results presented in Section 15.4.1.

The limiting case is for full power operation at end-of-cycle conditions. The full power evaluation includes analysis performed only for manual rod control (See Section 15.1.2.2).

15.1.2.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for an increase in feedwater flow event:

- Consistent with use of the RTDP, the initial values of reactor power, reactor coolant average temperature, and RCS pressure are assumed to be the nominal values without uncertainties as defined in Table 15.0-3.

- The moderator density coefficient is assumed to have the maximum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to be the minimum feedback limit shown in Figure 15.0-2. This combination gives the greatest positive reactivity and maximum power increase. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- No credit is taken for the heat capacity of the reactor coolant system and steam generator thick metal in attenuating the resulting plant cooldown.
- To initiate the transient, it is assumed that one main feedwater regulation valve is instantaneously opened to its maximum flow rate, which causes one steam generator to be supplied with water at 300% of the rated loop flow.
- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. RCCA insertion characteristics assumed in analysis is described in Section 15.0.0.2.5.
- Conservative axial power profile and radial power distribution are assumed in the analysis as described in Section 15.0.0.2.3.
- The model simulates the high-high steam generator water level signal and the function of the protection and safety monitoring system to trip the reactor and isolate feedwater. Table 15.0-4 summarizes the reactor trip setpoint and signal delay time used in the analysis, and Table 15.0-5 defines the additional delay for isolation of feedwater flow.

15.1.2.3.3 Results

Only the hot full power end-of-cycle (HFP EOC) manual rod control case is presented for this event. (See Section 15.1.2.3.1) Table 15.1.2-1 lists the key events and the times at which they occur, relative to the initiation of the postulated feedwater flow increase at full power transient. Figures 15.1.2-1 through 15.1.2-7 are plots of key system parameters versus time from the core response analysis for this event.

The reactor coolant temperature decreases when excessive water is supplied to the secondary side of a steam generator. This increases the reactor power because of the positive reactivity feedback effect of the moderator density coefficient. The high-high steam generator water level signal trips the reactor and turbine automatically. Additionally, feedwater is isolated by this signal.

As shown in Figure 15.1.2-1 the reactor neutron power remains well below 118% of nominal (the high power range neutron flux (high setpoint) trip analytical limit). Since the heat flux will lag behind a reactor neutron power change due to the fuel rod thermal time constant, the peak heat flux shown in Figure 15.1.2-2 is also well below this 118% value. Figure 15.1.2-7 confirms that the transient resulting from the decrease in feedwater temperature does not cause the minimum DNBR to decrease below the 95/95 limit.

Figure 15.1.2-6 demonstrates that the MHI internal criterion for ensuring that the steam generator does not fill with water has been met.

The normalized core average heat flux transient is virtually identical to the normalized maximum heat flux used in the DNBR calculations, and is considered representative of both parameters for this event. A plot for reactor vessel inlet temperature (showing all loops) is provided in place of inlet coolant temperature to illustrate the non-uniform inlet temperatures prior to mixing in the reactor vessel inlet. Because there is significant subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided; these are not key parameters for this event. Steam line pressure is not a key parameter for heat removal events where steam generator pressures are stable or decreasing. Containment parameters and pressurizer safety valve flow are not reported for this event because RCS pressure remains below the pressurizer safety valve set pressure and there are no releases from the RCS or steam generators inside containment.

15.1.2.4 Barrier Performance

The increase in feedwater flow event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The results of the Core and System Evaluation case demonstrate that the RCS pressure remains well below 110% of system design pressure. In addition, the main steam pressure cannot challenge the main steam system pressure design limit, as discussed in Section 15.1.2.3.3. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.1.2.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.1.2.6 Conclusions

An increased feedwater supply to the secondary side of the steam generator will cause an introduction of cooler water into the core, thus increasing the reactor power. However, the resulting transient, even under the most severe conditions at the end-of-cycle, does not cause the minimum DNBR to decrease below the 95/95 limit, and no fuel failures are predicted.

In addition, the RCS pressure and steam pressure remain well below 110% of their system design pressures, so the integrity of the reactor coolant pressure boundary and the integrity of the main steam system pressure boundary are maintained.

The analyses also demonstrate that the MHI internal criterion for ensuring that the steam generator does not fill with water has been met.

The increase in feedwater flow event does not lead to a more serious fault condition.

Table 15.1.2-1
Time Sequence of Events for Increase in Feedwater Flow

Event Description	Time [sec]
One main feedwater regulation valve fails fully open	0.0
High-high steam generator water level analytical limit reached	24.0
Reactor trip initiated (rod motion begins)	25.8
Minimum DNBR occurs	26.1
Main feedwater isolation valves close	32.0

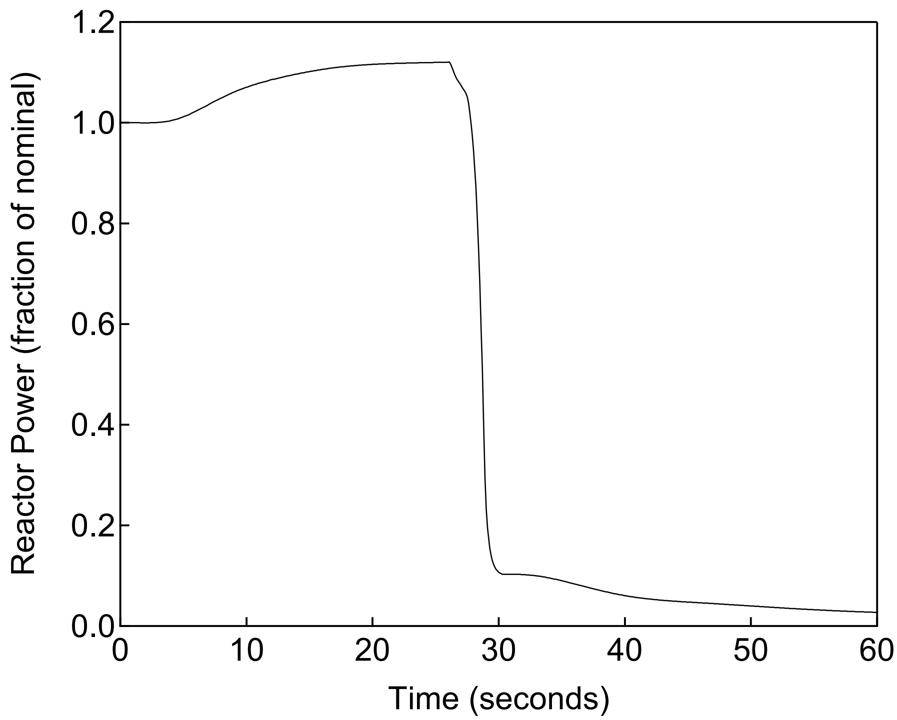


Figure 15.1.2-1 Reactor Power versus Time

Increase in Feedwater Flow

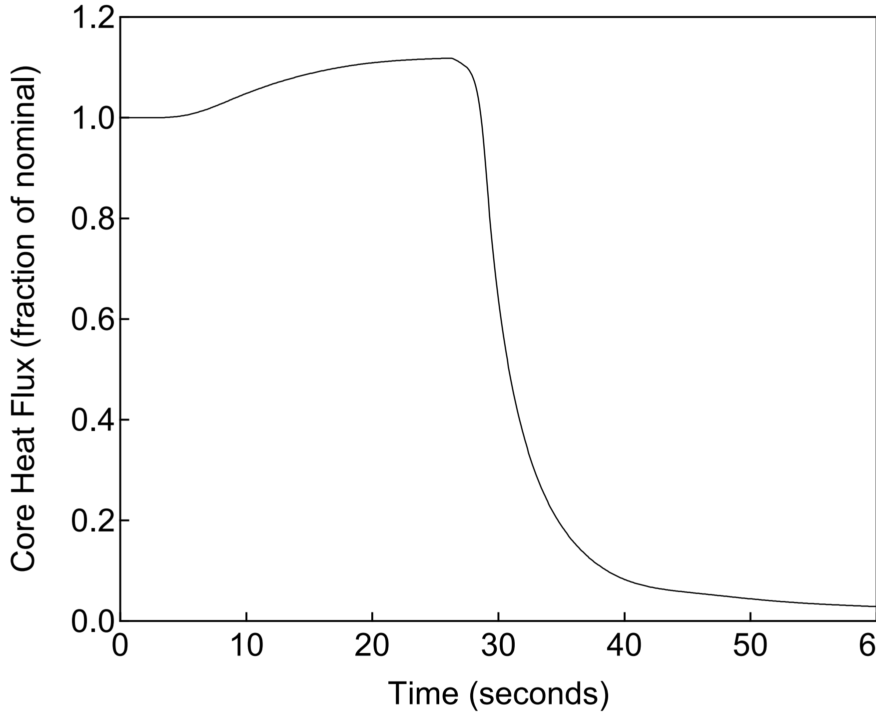


Figure 15.1.2-2 Core Heat Flux versus Time

Increase in Feedwater Flow

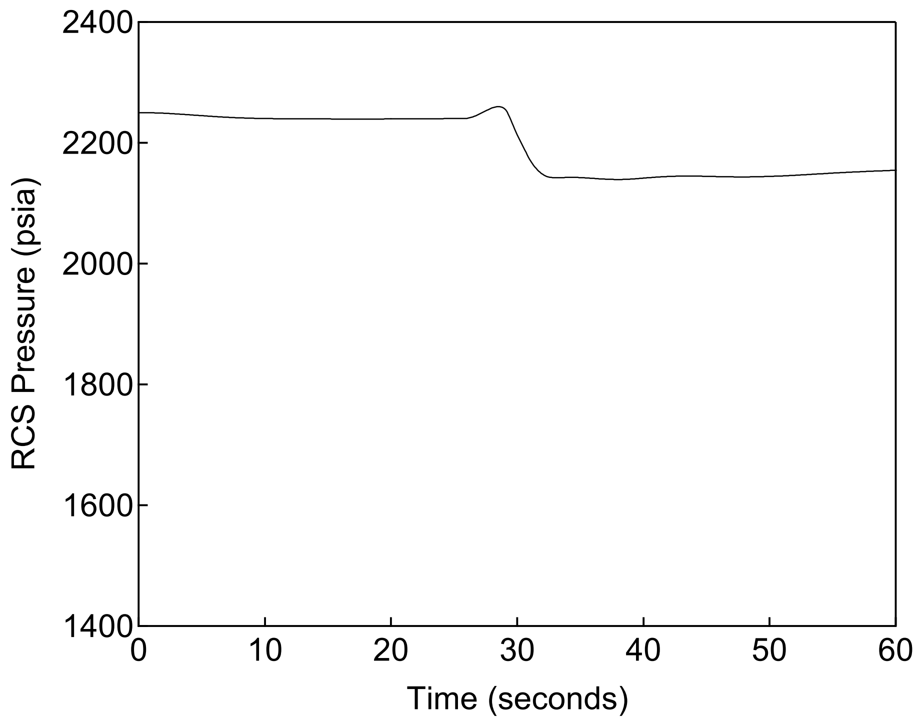


Figure 15.1.2-3 RCS Pressure versus Time

Increase in Feedwater Flow

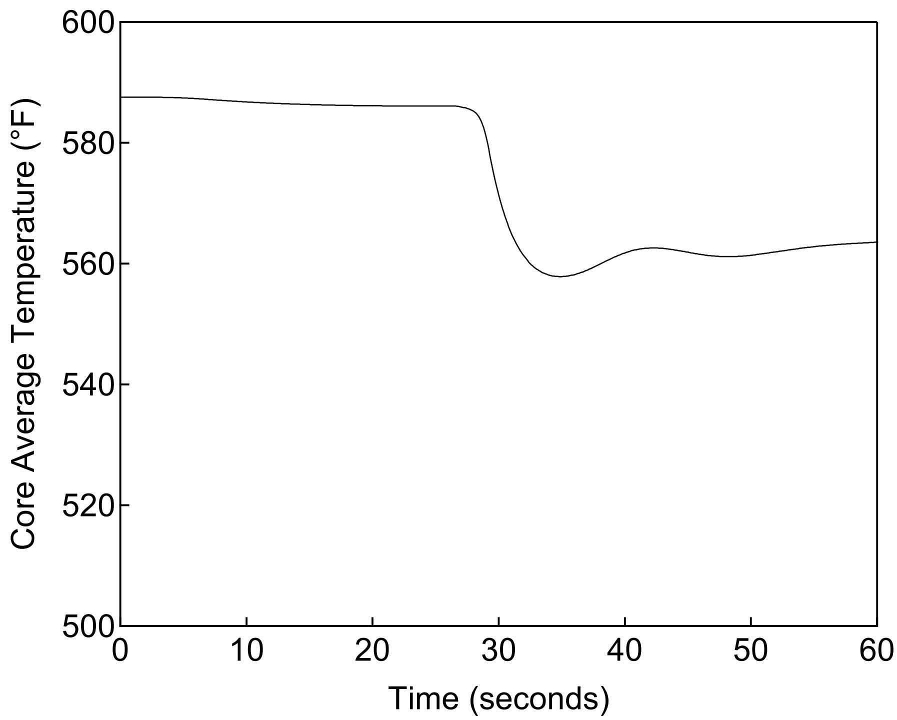


Figure 15.1.2-4 Core Average Temperature versus Time

Increase in Feedwater Flow

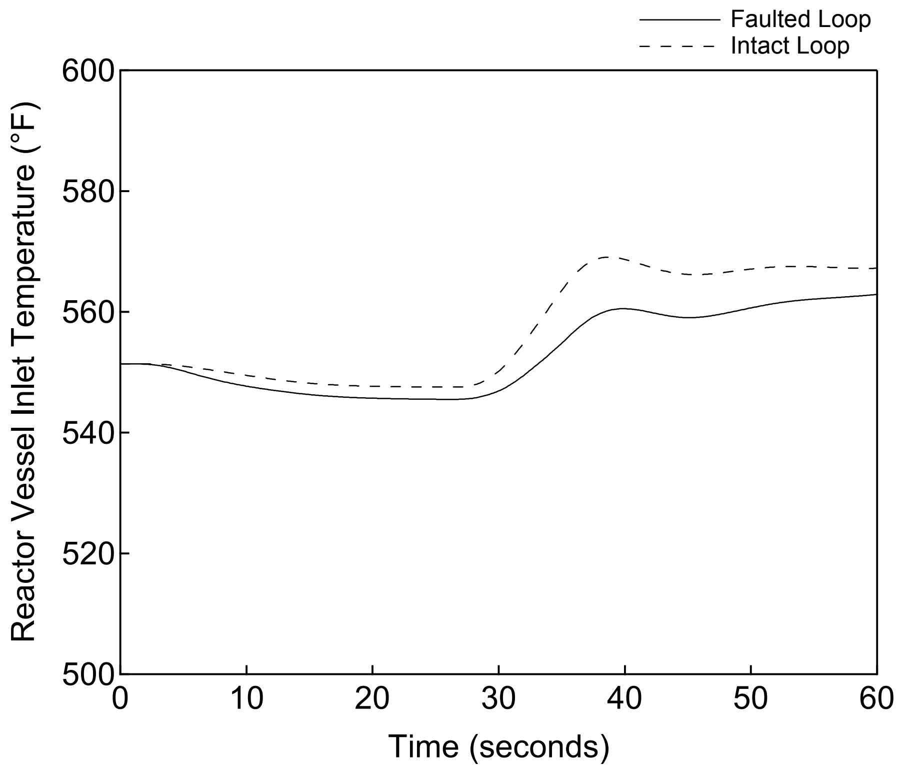


Figure 15.1.2-5 Reactor Vessel Inlet Temperature versus Time

Increase in Feedwater Flow

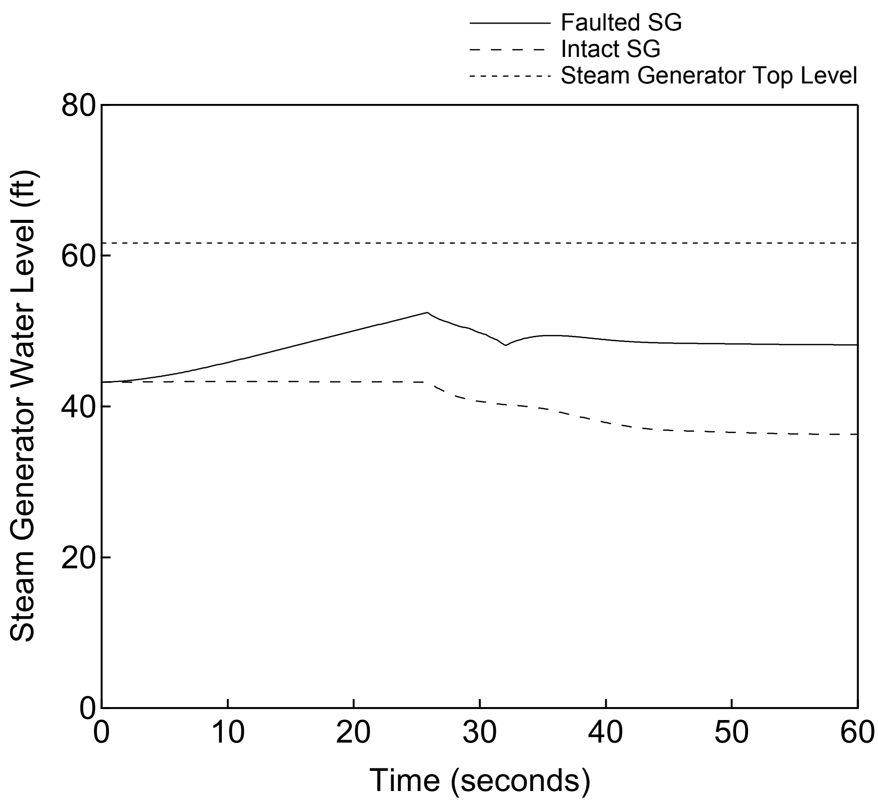


Figure 15.1.2-6 Steam Generator Water Level versus Time

Increase in Feedwater Flow

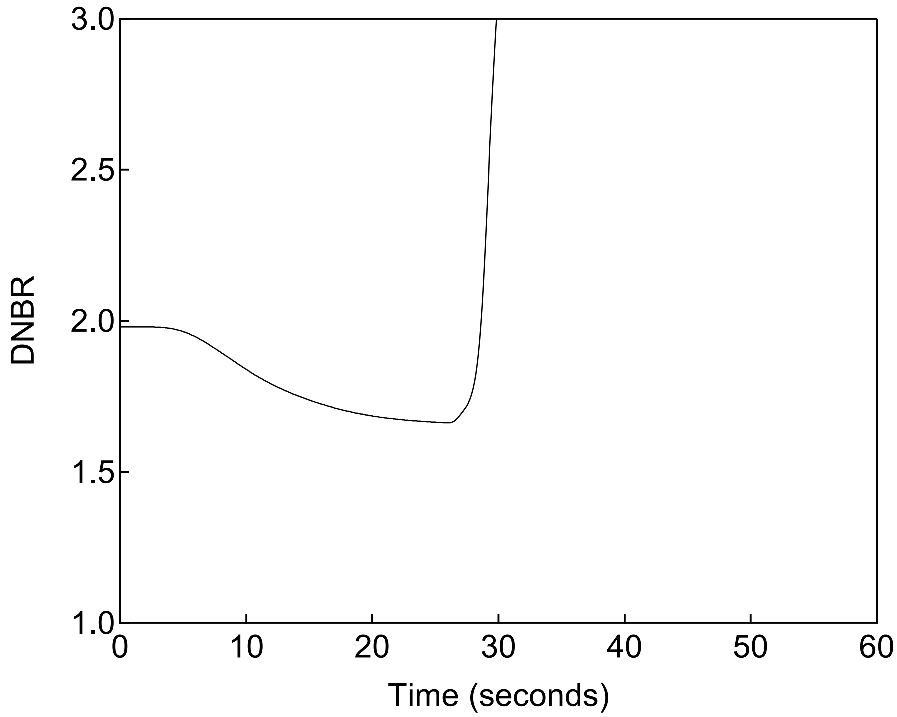


Figure 15.1.2-7 DNBR versus Time

Increase in Feedwater Flow

15.1.3 Increase in Steam Flow as a Result of Steam Pressure Regulator Malfunction

15.1.3.1 Identification of Causes and Frequency Classification

A rapid increase in steam flow can cause a temporary mismatch between the power produced by the reactor core and the power demanded by the steam generators. This situation can reduce the temperature of the coolant re-entering the reactor vessel, which, in turn, can lead to an increase in reactor power.

The reactor control system is designed to follow a 10% step load change or a 5% per minute ramp load increase. The analysis of the event documented in this section confirms that core, system, and boundary performance is acceptable following the more severe of these two conditions – the 10% step load change – without reactor trip system (RTS) action. Transients involving more rapid steam demand increases are analyzed in Sections 15.1.4 and 15.1.5.

A rapid increase in steam flow can be caused by (1) an administrative or operator error resulting in an excessive load increase during power operations or (2) an operator error or equipment malfunction that causes a turbine bypass valve, main turbine control valve, main steam relief valve, or main steam depressurization valve to inadvertently fully open.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.1-1). Event frequency conditions are described in Section 15.0.0.1.

15.1.3.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the increase in steam flow event is described in the results section.

The event is initiated with a 10% step increase in steam demand. This rapid increase in steam flow will cause a temporary mismatch between the power produced by the reactor core and the power demanded by the steam generators. This can reduce the temperature of the coolant re-entering the reactor vessel, which, in turn, can lead to an increase in reactor power. The reactor reaches a new steady state condition at the new power level. The analysis intentionally does not credit any protection system actions, but systems that could make the transient more severe, such as the use of automatic rod control, are addressed.

The turbine bypass valves, the main steam relief valves, and the main steam depressurization valves are designed to have a maximum capacity to limit the effect of transients caused by a malfunction that fully opens one of them. In addition, the control system is designed so that no single failure will cause two or more valves to fully open at the same time.

For this analysis, the RTS is not assumed to actuate, and no systems are required to mitigate this event. Therefore, no single failures are identified or assumed in the analysis for this event.

Normal reactor control systems and engineered safety systems are not required to function and are not credited in the evaluation. Although this analysis is performed assuming no reactor trips, the following automatic reactor trip signals are assumed to be available to provide protection from this event:

- High power range neutron flux (high setpoint)
- Over temperature ΔT
- Over power ΔT

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

For this analysis, the RTS is not actuated. As a result, a loss of offsite power as a consequence of a turbine trip, as discussed in Section 15.0.0.7, is not applicable for this event. If a turbine trip were to occur, the post-trip response to the loss of offsite power would not be limiting as explained in Section 15.0.0.7.

15.1.3.3 Core and System Performance

15.1.3.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of steam generator temperature and pressure to changes in feedwater and steam flow conditions, steam generator heat transfer, primary temperatures and pressure, reactivity, reactor power, and other parameters for uniform or non-uniform cooldown events initiated by the secondary system such as a 10% step increase in steam demand. MARVEL-M also has the ability to internally calculate DNBRs for transients with full reactor coolant system (RCS) flow and power distributions bounded by the normal design distributions. This simplified model calculates DNBRs based on user-input values of DNBR at known combinations of core inlet temperature, pressure, and power. The MARVEL-M evaluation model is described in Section 15.0.2.2.1. Additional information on the use of the MARVEL-M code for analyzing non-LOCA events can be found in Reference 15.1-2.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

The analysis is performed with both automatic and manual rod control. In addition, cases are performed for both minimum moderator reactivity feedback (typical of beginning-of-cycle conditions) and maximum moderator reactivity feedback (typical of end-of-cycle conditions). The minimum Doppler coefficient is used for all cases to minimize power feedback. Thus, four cases are considered:

- Manual rod control, minimum moderator reactivity feedback
- Manual rod control, maximum moderator reactivity feedback
- Automatic rod control, minimum moderator reactivity feedback
- Automatic rod control, maximum moderator reactivity feedback

15.1.3.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for an increase in steam flow event:

- Consistent with use of the RTDP, the initial values of reactor power, reactor coolant average temperature, and RCS pressure are assumed to be the nominal values without uncertainties as defined in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value for the beginning-of-cycle case and the maximum value for the end-of-cycle case as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to be the minimum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative axial power profile and radial power distribution are assumed in the analysis as described in Section 15.0.0.2.3.

- A 10% step change in steam flow takes place when the reactor is operating at rated power.

15.1.3.3.3 Results

Table 15.1.3-1 provides a sequence of events for all four cases. Because no reactor trips occur, only the time to establish equilibrium conditions is reported.

Figures 15.1.3-1 through 15.1.3-6 are plots of system parameters versus time for Case A (manual rod control, minimum moderator reactivity feedback).

Figures 15.1.3-7 through 15.1.3-12 are plots of system parameters versus time for Case B (manual rod control, maximum moderator reactivity feedback).

Figures 15.1.3-13 through 15.1.3-18 are plots of system parameters versus time for Case C (automatic rod control, minimum moderator reactivity feedback)

Figures 15.1.3-19 through 15.1.3-24 are plots of system parameters versus time for Case D (automatic rod control, maximum moderator reactivity feedback)

In all cases, the reactor reaches a stable condition at a slightly higher power level and a lower reactor coolant average temperature. The reactor neutron power remains well below 118% of nominal (the high power range neutron flux (high setpoint) trip analytical limit). The peak heat flux, which will stabilize at a higher value, is also well below this 118% value.

The DNBR remains well above the 95/95 limit for all cases. Thus, the fuel cladding temperature would not increase significantly during this transient. The RCS pressure tends to decrease or remain relatively constant for all cases and does not challenge the RCS design pressure. In addition, this cooldown transient does not challenge the main steam system pressure design limit. No figure is provided since steam line pressure is not a key parameter for heat removal events where steam generator pressures are stable or decreasing.

In all cases, the reactor reaches a stable condition that meets the acceptance criteria, even with no protective action. The reactor power can be reduced from this post-transient level following normal operating procedures. It is possible that, in some cases, the reactor could trip due to combinations of uncertainties in instrumentation and trip points. If this occurs the plant reaches a stable shutdown condition.

The normalized core average heat flux transient is virtually identical to the normalized maximum heat flux used in the DNBR calculations, and is considered representative of both parameters for this event. A plot for reactor vessel inlet temperature (showing all loops) is provided for each case in place of inlet coolant temperature to illustrate the non-uniform inlet temperatures prior to mixing in the reactor vessel inlet. Because there is significant subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided; these are not key parameters for this event. Containment parameters and pressurizer safety valve flow are not reported for this event because RCS pressure remains below the pressurizer safety valve set pressure and there are no releases from the RCS or steam generators inside containment.

15.1.3.4 Barrier Performance

The increase in steam flow event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The results of the Core and System Evaluation case demonstrate that the RCS pressure remains well below 110% of system design pressure. In addition, the main steam pressure cannot challenge the main steam system pressure design limit, as discussed in Section 15.1.3.3.3. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.1.3.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.1.3.6 Conclusions

A rapid increase in steam demand will cause an introduction of cooler water into the core, thus increasing the reactor power. However, the resulting transient, even under the most severe combination of conditions and no protective action, does not cause the minimum DNBR to decrease to below the 95/95 limit, and no fuel failures are predicted.

In addition, the RCS pressure and steam pressure remain well below 110% of their system design pressures, so the integrity of the reactor coolant pressure boundary and the integrity of the main steam system pressure boundary are maintained.

The increase in steam flow event does not lead to a more serious fault condition.

Table 15.1.3-1
Time Sequence of Events for Increase in Steam Flow

Event Description	Time [sec]
10% step load increase	0
Approximate time equilibrium conditions are reached	Case A: 175 Case B: 60 Case C: 100 Case D: 35

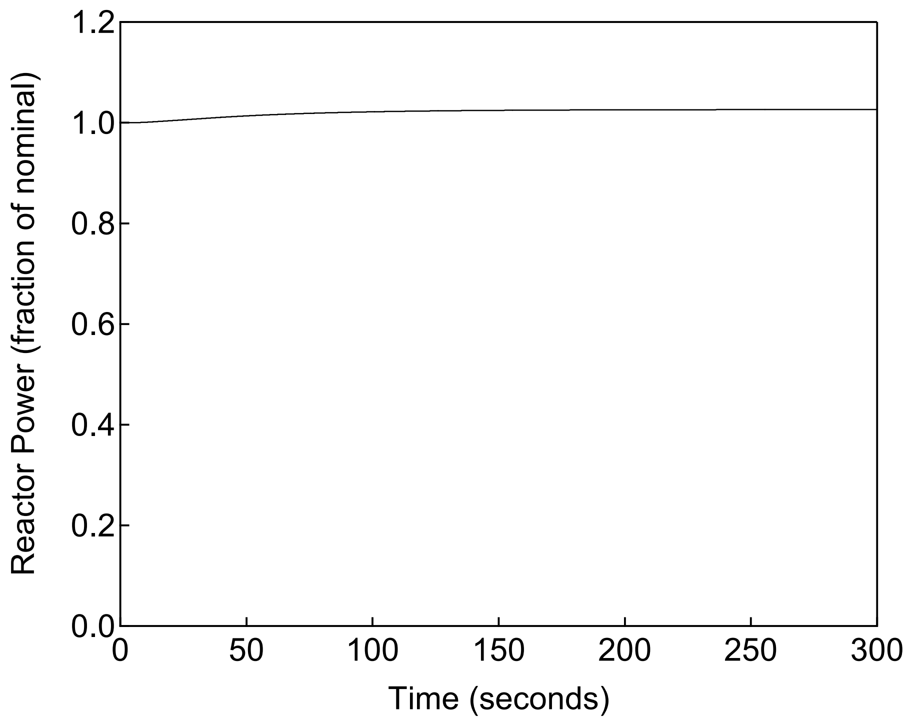


Figure 15.1.3-1 Reactor Power versus Time

Increase in Steam Flow
- Case A: Manual Rod Control, Minimum Moderator Feedback

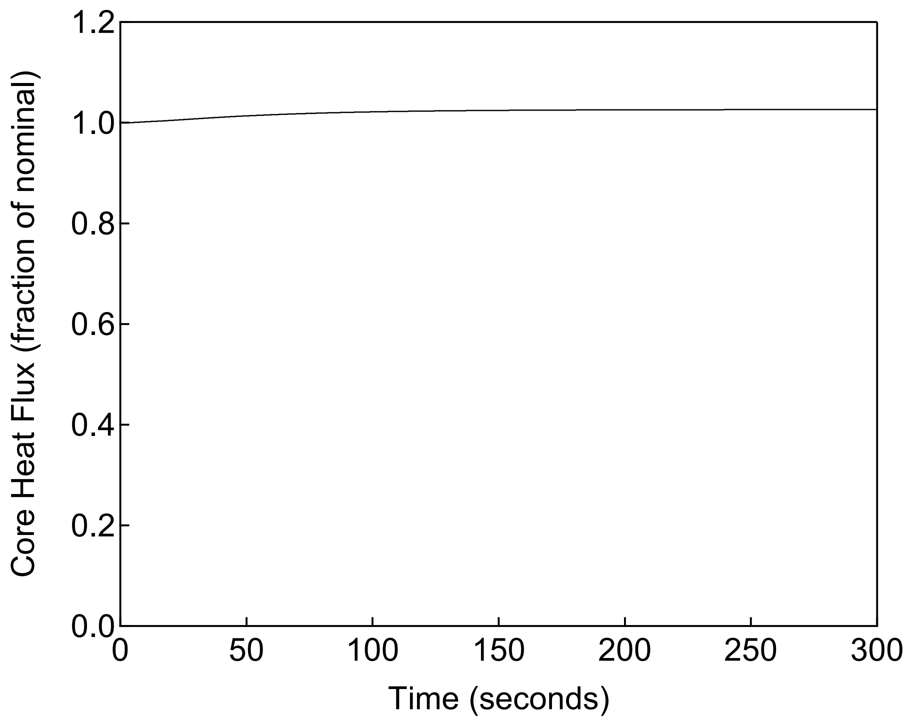


Figure 15.1.3-2 Core Heat Flux versus Time

Increase in Steam Flow
- Case A: Manual Rod Control, Minimum Moderator Feedback

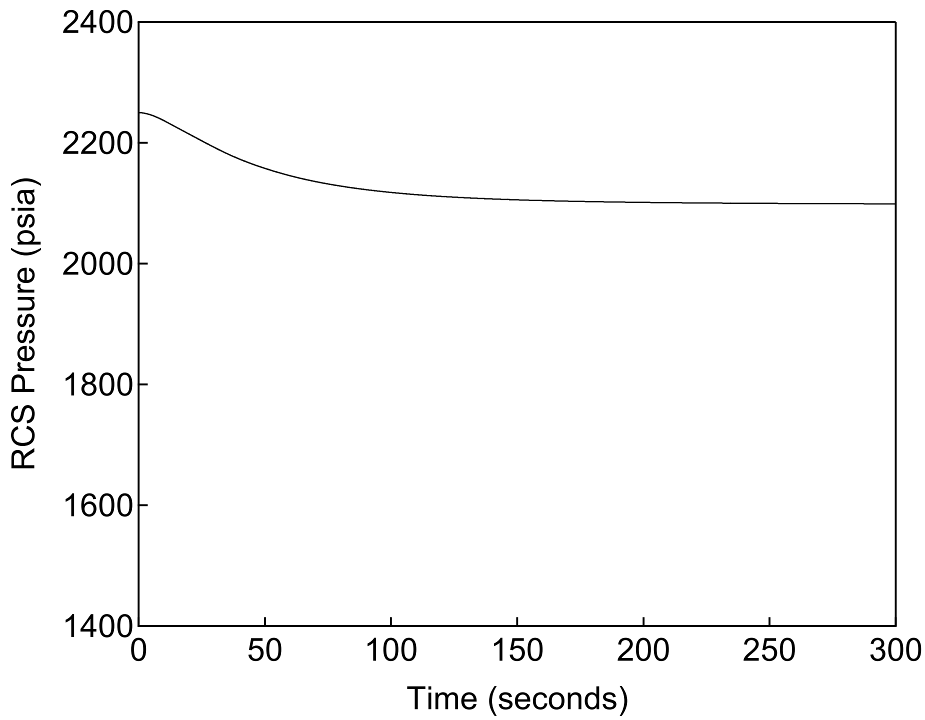


Figure 15.1.3-3 RCS Pressure versus Time

Increase in Steam Flow
- Case A: Manual Rod Control, Minimum Moderator Feedback

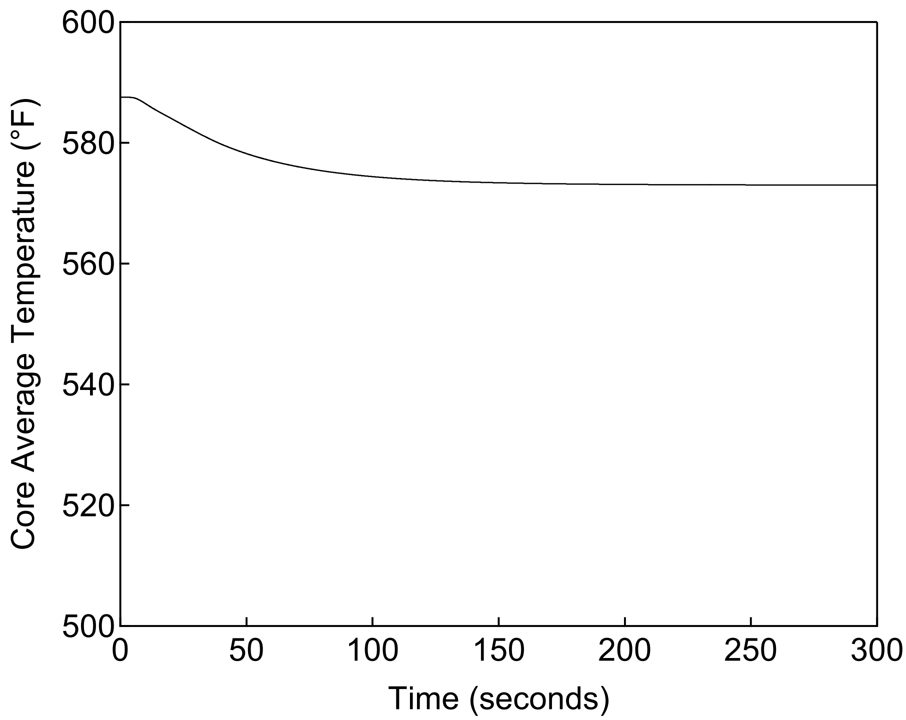


Figure 15.1.3-4 Core Average Temperature versus Time

Increase in Steam Flow
- Case A: Manual Rod Control, Minimum Moderator Feedback

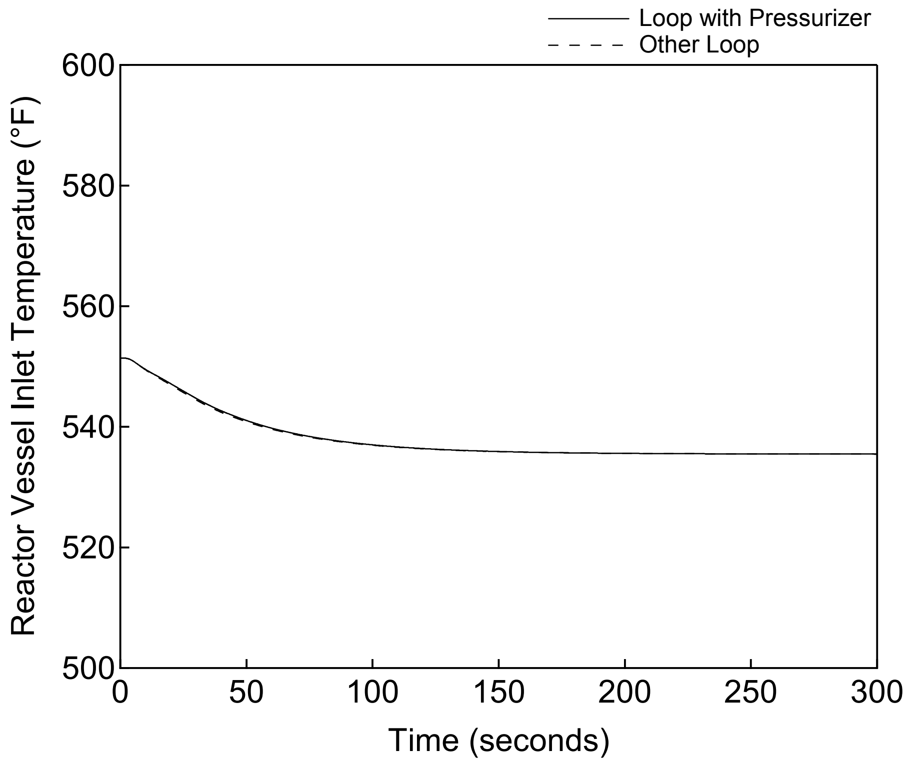


Figure 15.1.3-5 Reactor Vessel Inlet Temperature versus Time

Increase in Steam Flow
- Case A: Manual Rod Control, Minimum Moderator Feedback

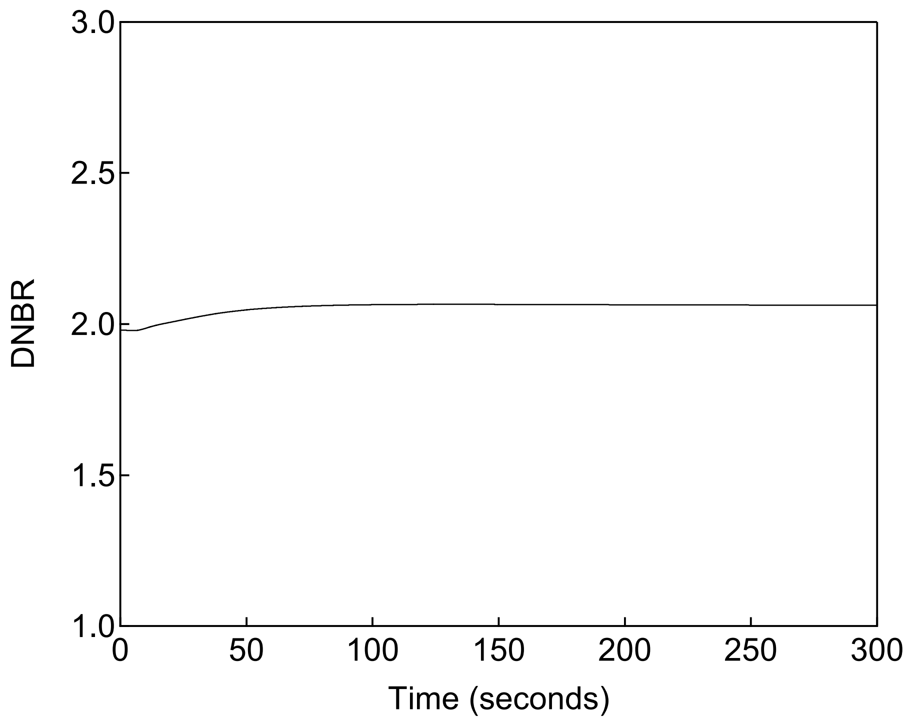


Figure 15.1.3-6 DNBR versus Time

Increase in Steam Flow
- Case A: Manual Rod Control, Minimum Moderator Feedback

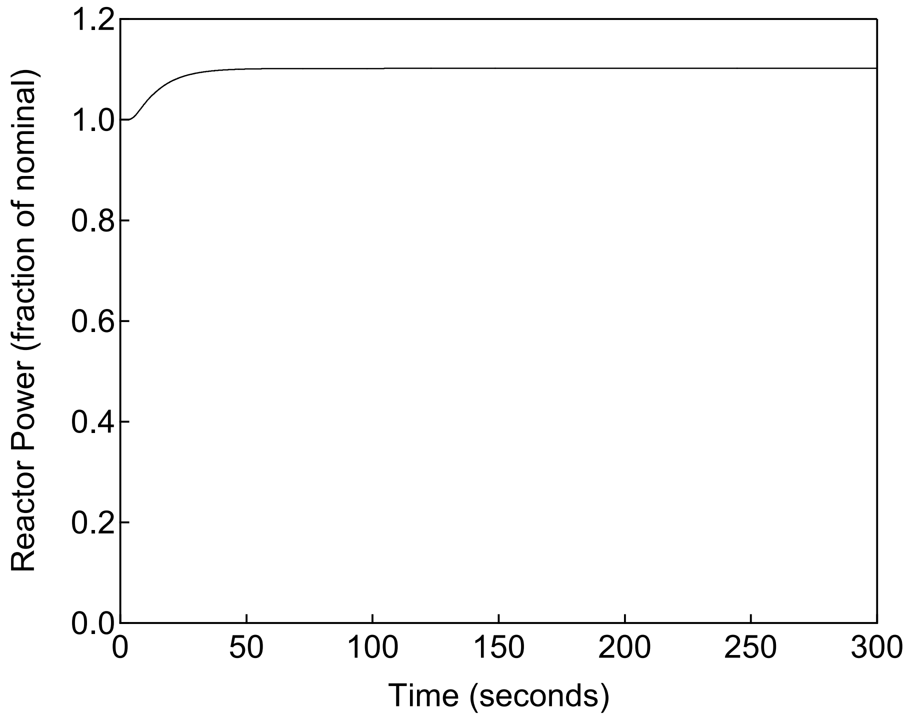


Figure 15.1.3-7 Reactor Power versus Time

Increase in Steam Flow
- Case B: Manual Rod Control, Maximum Moderator Feedback

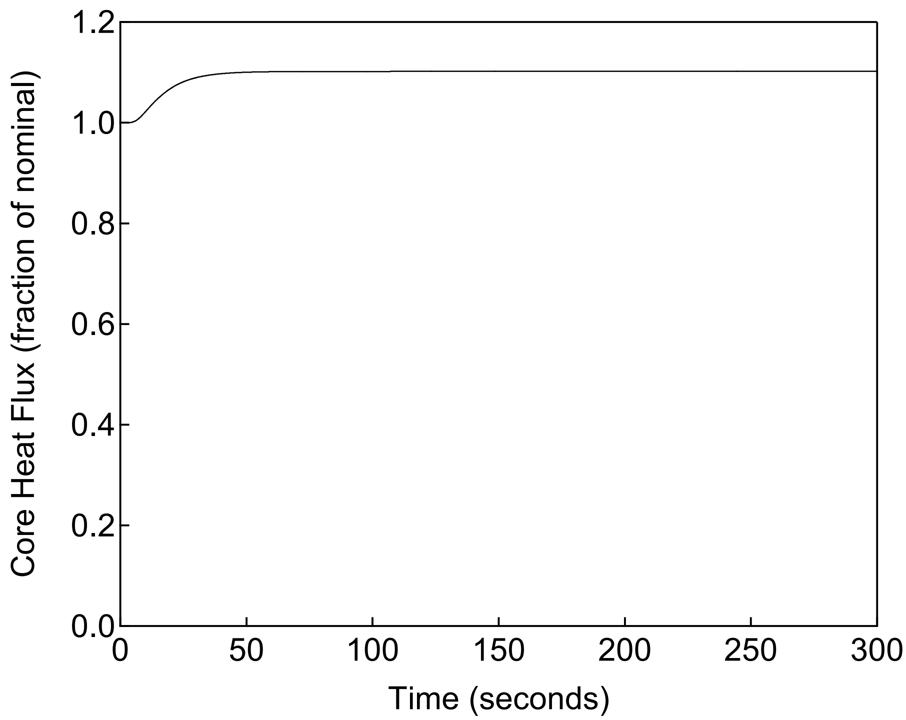


Figure 15.1.3-8 Core Heat Flux versus Time

Increase in Steam Flow
- Case B: Manual Rod Control, Maximum Moderator Feedback

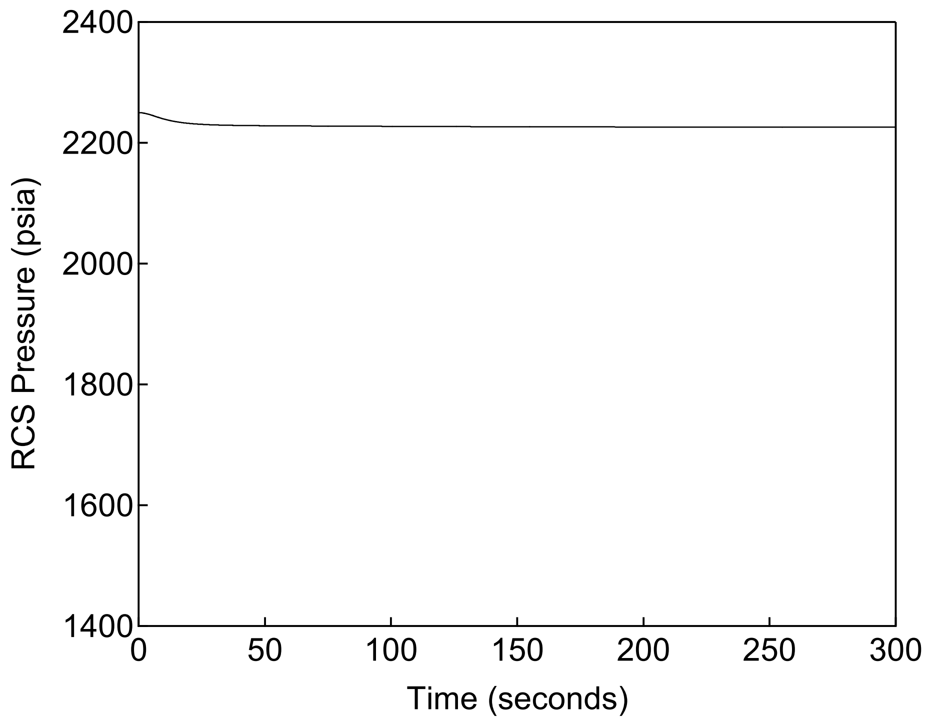


Figure 15.1.3-9 RCS Pressure versus Time

Increase in Steam Flow
- Case B: Manual Rod Control, Maximum Moderator Feedback

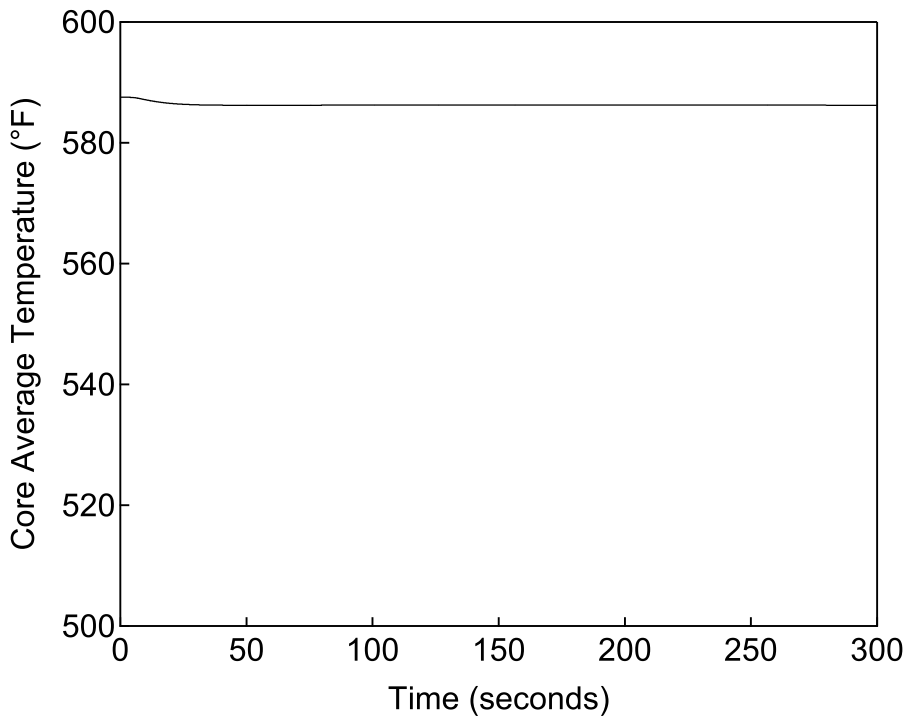


Figure 15.1.3-10 Core Average Temperature versus Time

Increase in Steam Flow
- Case B: Manual Rod Control, Maximum Moderator Feedback

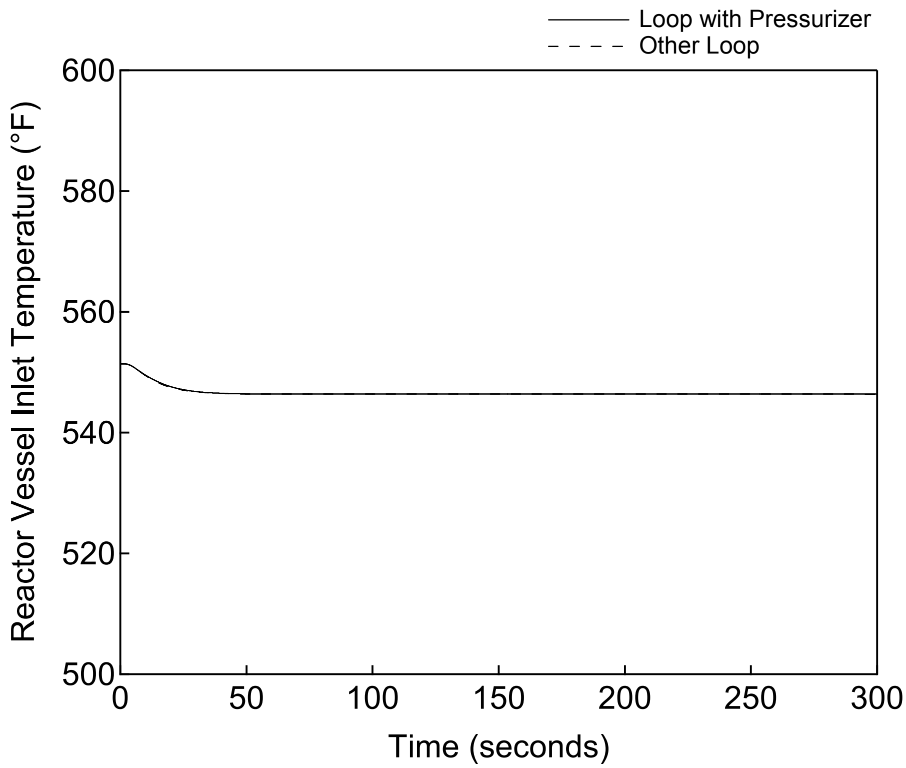


Figure 15.1.3-11 Reactor Vessel Inlet Temperature versus Time

Increase in Steam Flow
- Case B: Manual Rod Control, Maximum Moderator Feedback

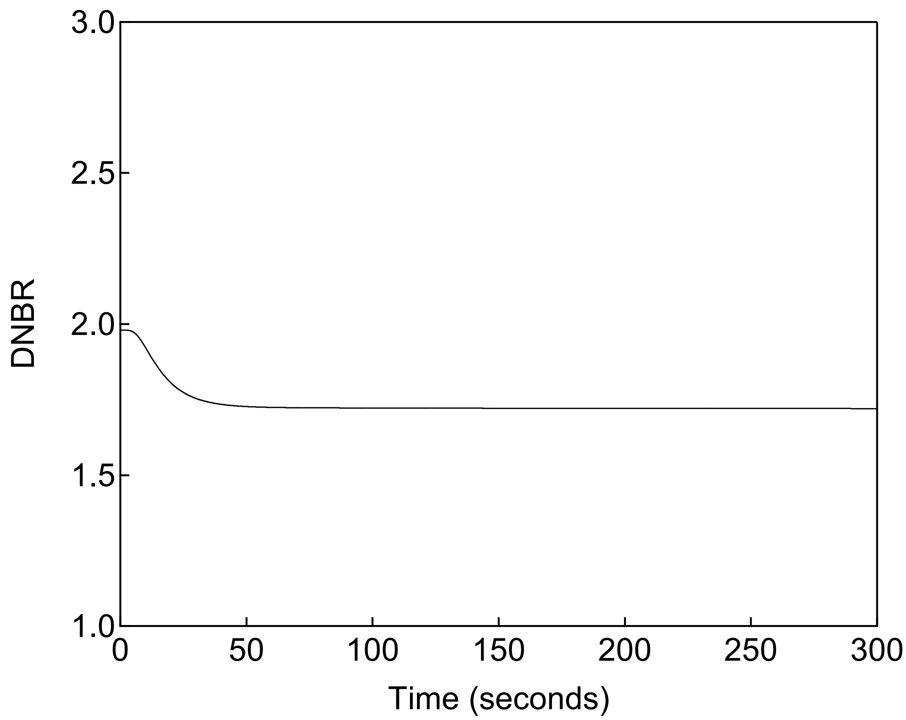


Figure 15.1.3-12 DNBR versus Time

Increase in Steam Flow
- Case B: Manual Rod Control, Maximum Moderator Feedback

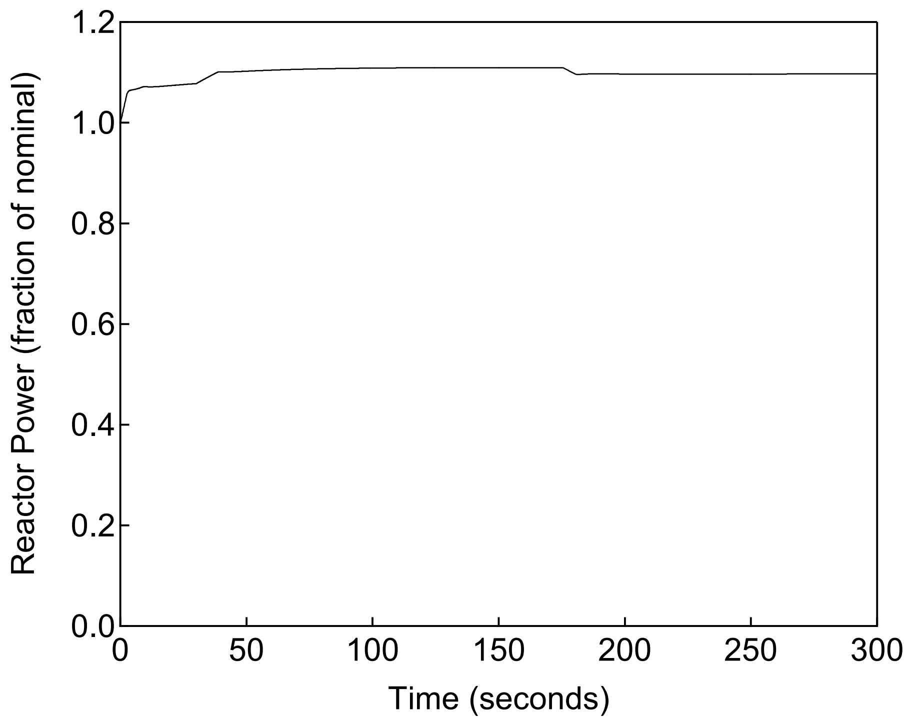


Figure 15.1.3-13 Reactor Power versus Time

Increase in Steam Flow as
- Case C: Automatic Rod Control, Minimum Moderator Feedback

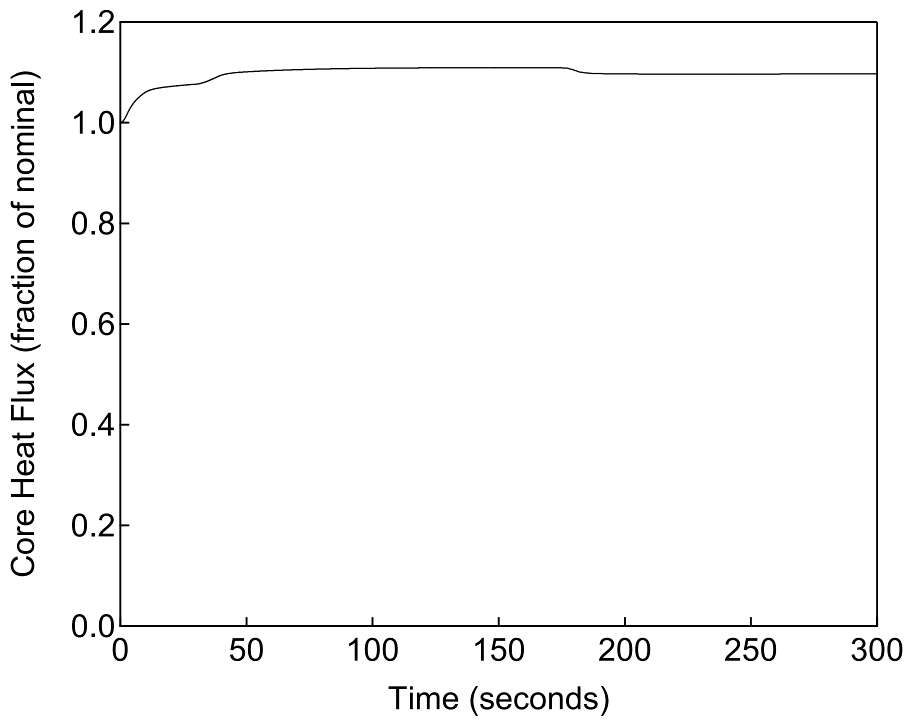


Figure 15.1.3-14 Core Heat Flux versus Time

Increase in Steam Flow
- Case C: Automatic Rod Control, Minimum Moderator Feedback

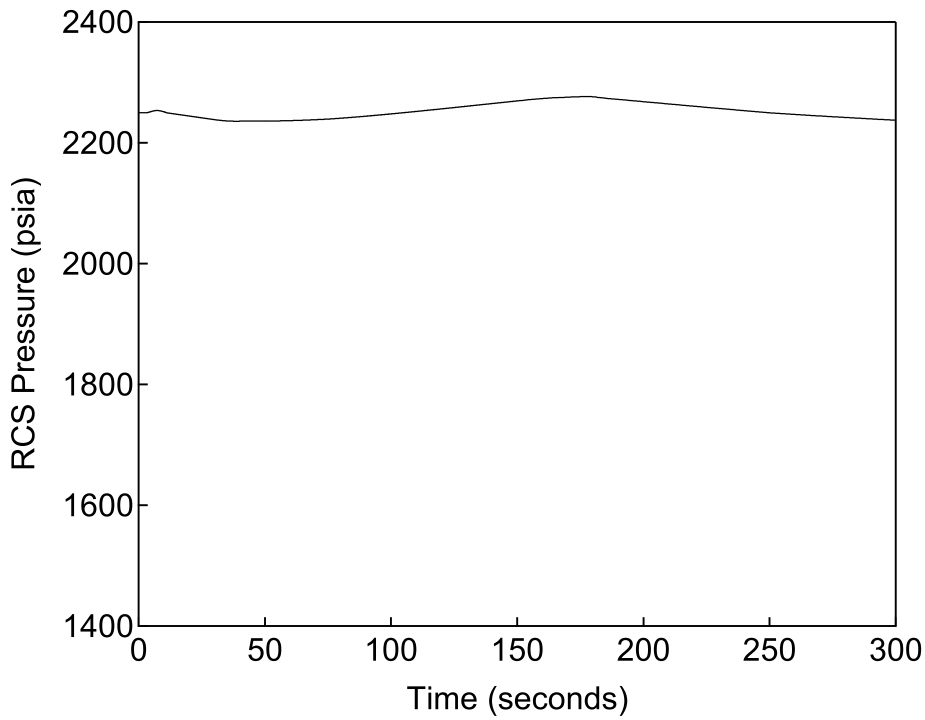


Figure 15.1.3-15 RCS Pressure versus Time

Increase in Steam Flow
- Case C: Automatic Rod Control, Minimum Moderator Feedback

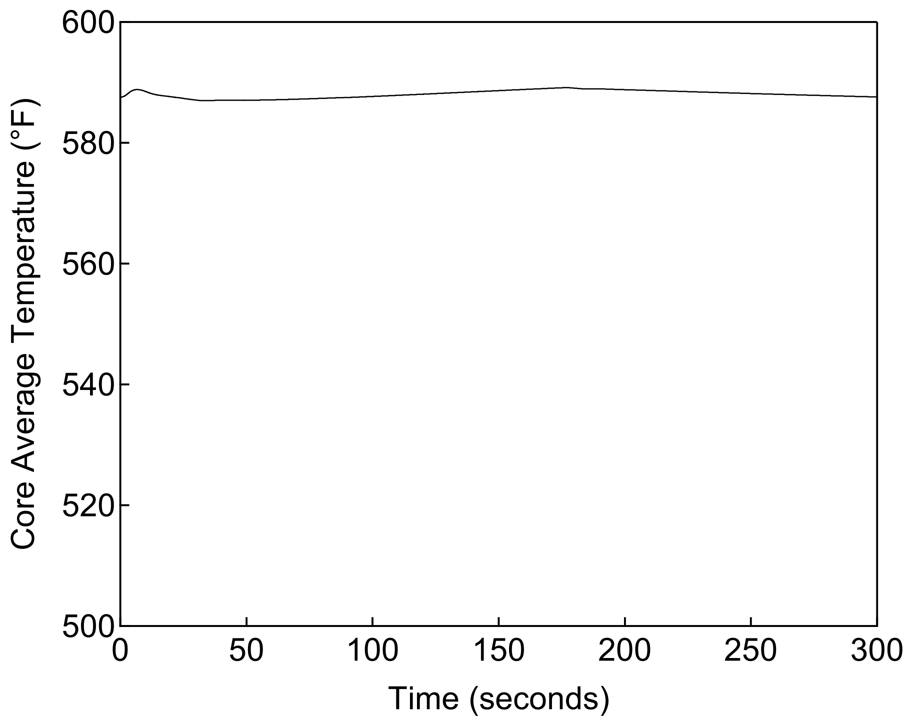


Figure 15.1.3-16 Core Average Temperature versus Time

Increase in Steam Flow
- Case C: Automatic Rod Control, Minimum Moderator Feedback

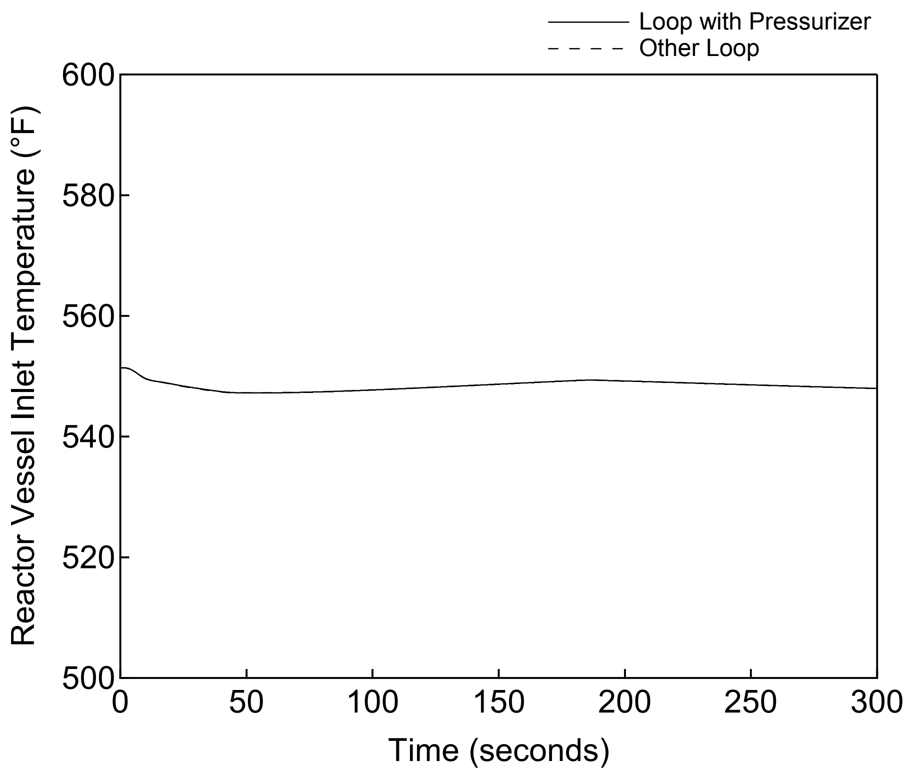


Figure 15.1.3-17 **Reactor Vessel Inlet Temperature versus Time**

Increase in Steam Flow
- Case C: Automatic Rod Control, Minimum Moderator Feedback

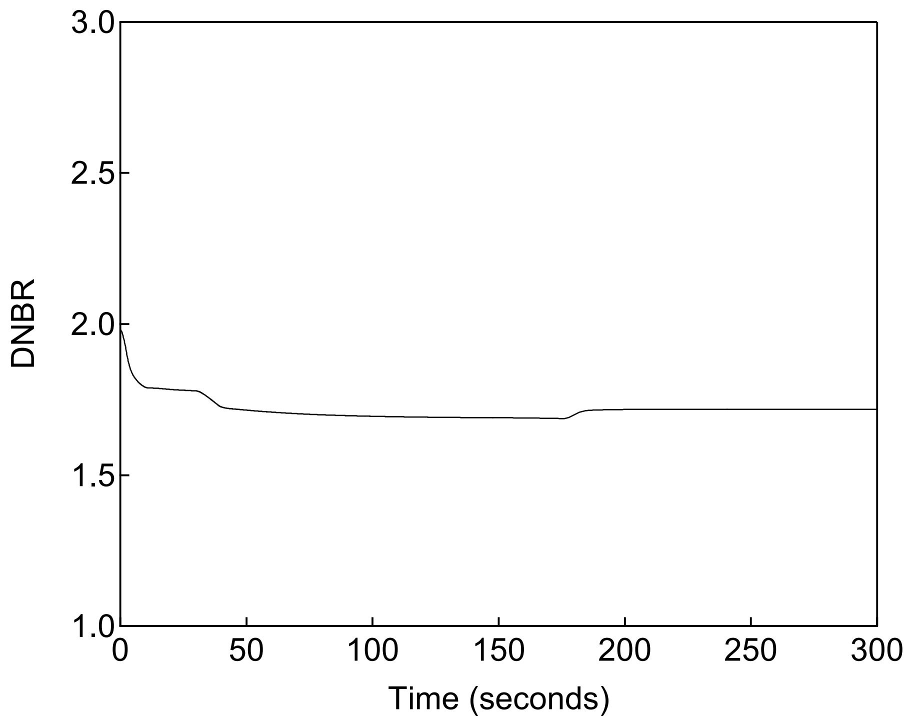


Figure 15.1.3-18 DNBR versus Time

Increase in Steam Flow
- Case C: Automatic Rod Control, Minimum Moderator Feedback

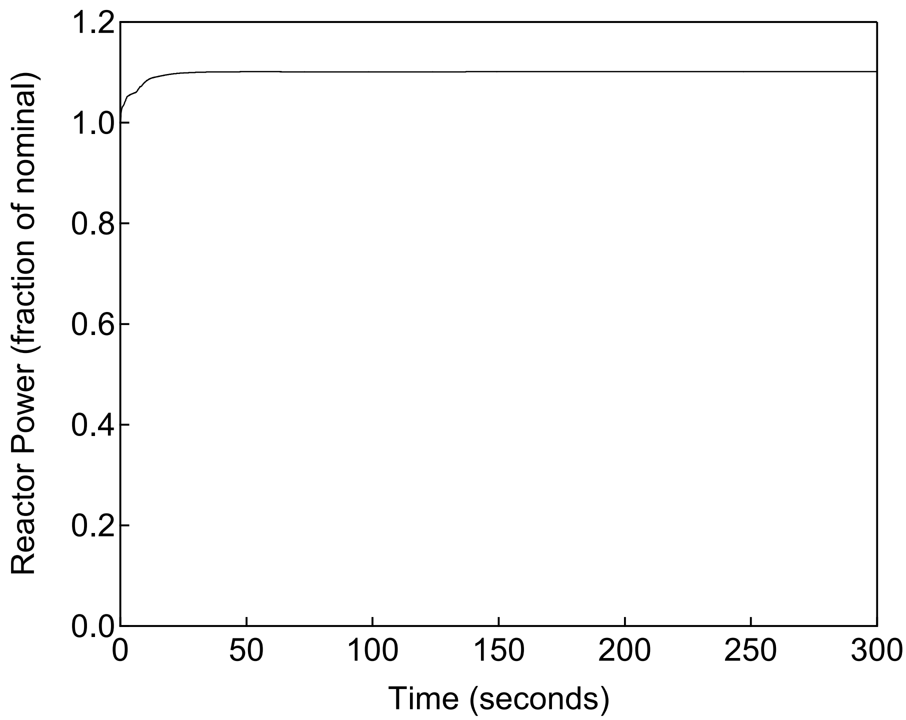


Figure 15.1.3-19 Reactor Power versus Time

Increase in Steam Flow
- Case D: Automatic Rod Control, Maximum Moderator Feedback

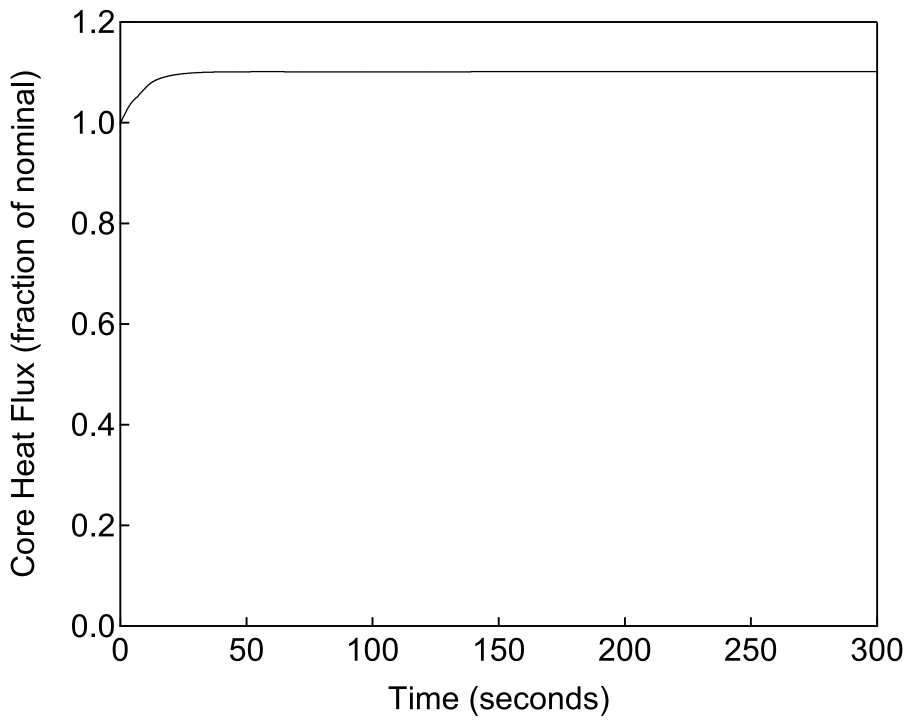


Figure 15.1.3-20 Core Heat Flux versus Time

Increase in Steam Flow
- Case D: Automatic Rod Control, Maximum Moderator Feedback

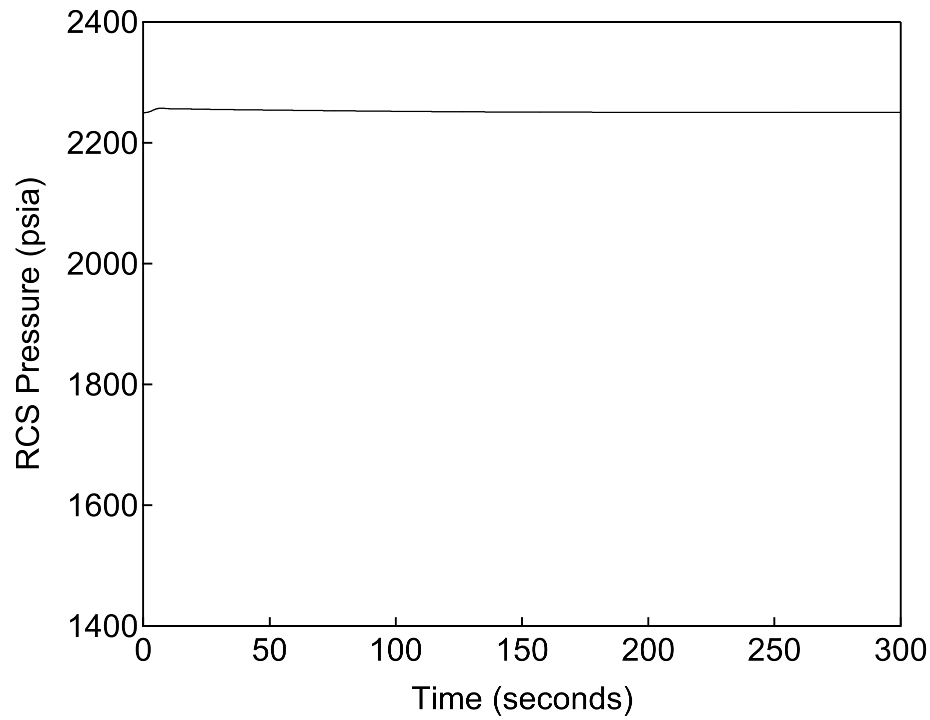


Figure 15.1.3-21 RCS Pressure versus Time

Increase in Steam Flow
- Case D: Automatic Rod Control, Maximum Moderator Feedback

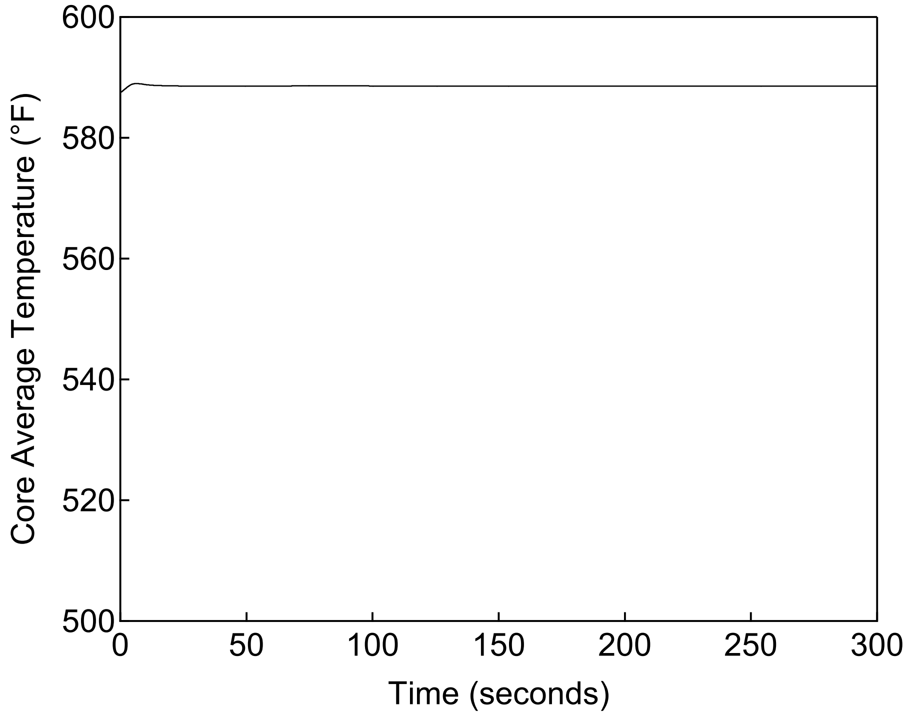


Figure 15.1.3-22 Core Average Temperature versus Time

Increase in Steam Flow
- Case D: Automatic Rod Control, Maximum Moderator Feedback

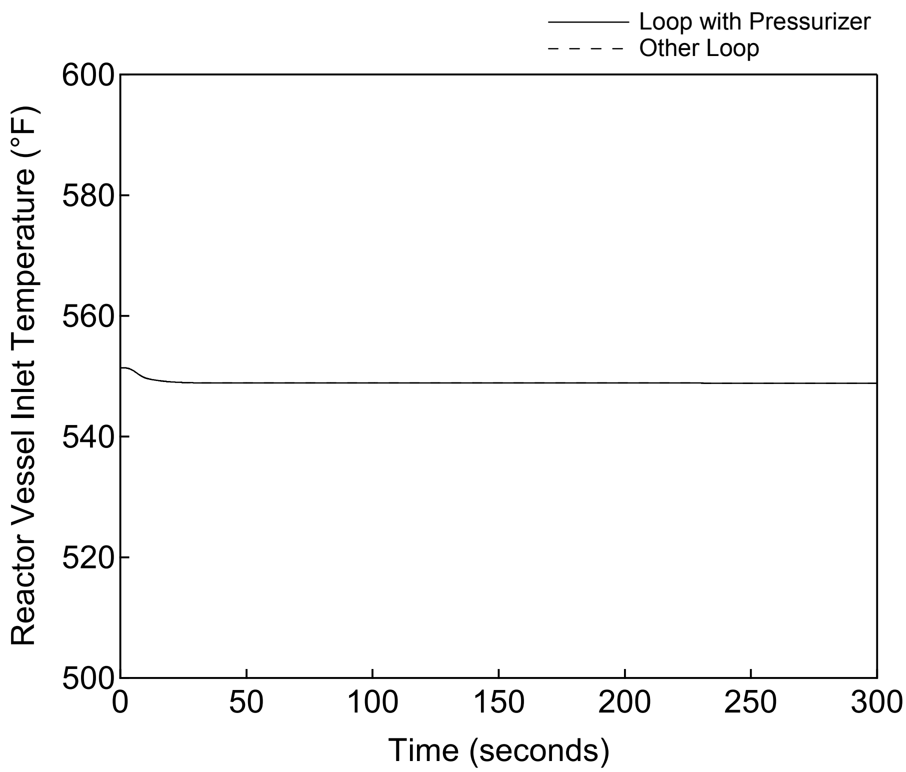


Figure 15.1.3-23 Reactor Vessel Inlet Temperature versus Time

Increase in Steam Flow
- Case D: Automatic Rod Control, Maximum Moderator Feedback

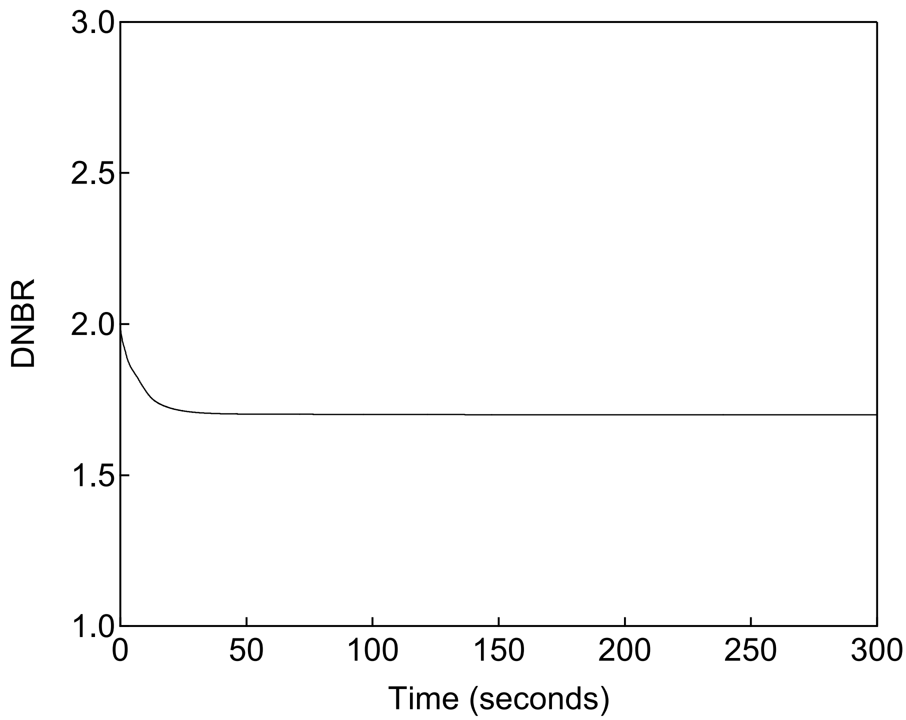


Figure 15.1.3-24 DNBR versus Time

Increase in Steam Flow
- Case D: Automatic Rod Control, Maximum Moderator Feedback

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Identification of Causes and Frequency Classification

The inadvertent opening of a main steam relief valve, main steam depressurization valve, main steam safety valve, or turbine bypass valve can cause a rapid increase in steam flow and a depressurization of the secondary system. The steam release removes energy from the reactor coolant system (RCS), which causes a reduction in the reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase. A return to criticality and power is possible if the event is initiated from hot standby.

This event is caused by a malfunction that inadvertently opens a single main steam relief, depressurization, safety valve or turbine bypass valve, resulting in the uncontrolled release of steam from the main steam system.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.1-1). Event frequency conditions are described in Section 15.0.0.1.

The analysis of the more severe event, the postulated failure of a main steam line accident, is documented in Section 15.1.5.

15.1.4.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the inadvertent opening of a steam generator relief or safety valve event is described in the results section.

The event is initiated by a malfunction that causes one of the main steam relief, depressurization or safety valves or one of the turbine bypass valves to fail in the fully open position, releasing steam from the main steam system.

Inadvertent opening of a main steam relief, depressurization, safety, or turbine bypass valve can cause a rapid increase in steam flow and a depressurization of the secondary system. The steam release removes energy from the reactor coolant system, which causes a reduction in the reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase.

The rate of steam release decreases as the steam pressure decreases during the course of the transient.

If the event occurs from full power, the increase in flow would be shared between the steam generators. This case would be bounded by the plant response to the 10% step increase in steam flow analyzed in Section 15.1.3. Therefore, the event discussed in this section is analyzed for hot standby initial conditions. For the US-APWR, the main steam relief, depressurization and safety valves are upstream of the main steam isolation valves and main steam check valves. This results in steam release from one steam generator and a non-uniform cooldown. If the failed valve is a turbine bypass valve, which is downstream of the main steam check valves, the steam release would be shared by all the steam generators and a uniform cooldown would result. The energy removed from the reactor coolant system by this event is sufficient to cause the RCS pressure to initiate the emergency core cooling system (ECCS) on low pressurizer pressure. However, the RCS pressure does not decrease below the accumulator charge pressure; therefore, the accumulators are not credited in the analysis. The most reactive rod cluster control assembly (RCCA) is assumed to be fully withdrawn, consistent with the guidance in the SRP, and its effect on both reactivity and core radial power distribution is considered. A single bounding case corresponding to the largest single main steam relief, main steam safety, or turbine bypass valve is defined assuming the largest single valve is upstream of the main steam check valve. This non-uniform cooldown results in the most positive reactivity and would not be terminated by the closure of the main steam isolation valves.

The main steam relief valves, main steam depressurization valves, main steam safety valves, and turbine bypass valves are designed with a maximum capacity that limits the severity of transients caused by a full-open failure. In addition, the control systems for these valves are designed so that no single failure will cause two or more valves to open fully at the same time.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

The analysis from the hot standby condition will not result in a reactor or turbine trip. As a result, a loss of offsite power as a consequence of a turbine trip, as discussed in Section 15.0.0.7, is not applicable for this case. For the less limiting case initiated from at-power conditions, if a turbine trip were to occur, the post-trip response to the loss of offsite power would not be limiting as explained in Section 15.0.0.7. A larger core cooldown rate results from assuming full RCS flow for the entire duration of this event.

The limiting single failure for this event is the failure of one train of the ECCS. If this occurs, the remaining trains provide the functions credited in this analysis.

Normal reactor control systems are not required to function and are not credited in the evaluation. Certain systems, such as main feedwater control, are assumed to operate if their operation makes the event more severe.

If the event is caused by a failed-open main steam relief valve, the transient can be terminated by closing the upstream main steam relief valve block valve. If the event is caused by a failed-open turbine bypass valve, the transient can be terminated by closing the main steam isolation valves. However, these operator actions are not credited in the evaluation of this event.

The following automatic reactor trip signals are assumed to be available (but not necessarily credited in the analysis) to provide protection from this transient:

- High power range neutron flux
- Over temperature ΔT
- Over power ΔT
- Low pressurizer pressure
- ECCS actuation

The following signals could actuate the ECCS, which injects borated water into the reactor vessel via the safety injection pumps:

- Low pressurizer pressure
- Low main steam line pressure (any one loop)

An ECCS actuation signal provides feedwater isolation by automatically tripping the main feedwater pumps and fully closing all control valves and feedwater isolation valves in the feedwater system to prevent feedwater from excessively cooling the RCS. The signal also starts the emergency feedwater (EFW). In addition to the reactor trips listed above, the following engineered safety feature functions are assumed to be available to mitigate the accident:

- Steam line isolation
- EFW isolation
- ECCS
- Main feedwater isolation

The automatic reactor coolant pump (RCP) trip will actuate on an ECCS actuation signal generated from low pressurizer pressure or low main steam line pressure. However, the RCP trip is ignored in this analysis to maximize the RCS cooldown and associated reactivity and return-to-power response.

15.1.4.3 Core and System Performance

15.1.4.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of primary and secondary temperatures and pressures, pressure-dependent relief or safety valve flow rate, reactivity, and boron concentration, as well as the automatic actuation and operation of engineered safety feature functions such as safety injection, emergency feedwater initiation and isolation, and main feedwater isolation. In addition, a non-perfect mixing model is used for the reactor vessel inlet plenum for the purpose of conservatively predicting reactivity for the non-uniform core inlet conditions. Additional information on the use of the MARVEL-M code for analyzing non-LOCA events including secondary steam releases can be found in Reference 15.1-2. This evaluation model is described in Section 15.0.2.2.1.

A steady-state analysis is performed at the peak transient power (as determined by MARVEL-M) to calculate the minimum DNBR using VIPRE-01M. The ANC code is used to calculate the limiting power distribution assuming the most reactive rod is fully withdrawn; the limiting location is in the core quadrant associated with the broken loop.

The ANC analysis also confirms the conservatism of the reactivity and reactor power transients as calculated by the MARVEL-M code. The ANC power distribution and core inlet temperature distribution are used to perform a hot channel DNBR analysis using VIPRE-01M and the W-3 correlation. The W-3 correlation (See Sections 4.4.2.2.1 and 4.4.4.1) and its associated 95/95 limit are used because the RCS pressures are below the applicable pressure range for the WRB-2 DNBR correlation. Additional details regarding this methodology are provided in Reference 15.1-2.

The analysis is performed for the reactor initially in a hot standby condition. The basis for this is described below.

If a main steam relief or safety valve were to inadvertently open when the reactor was at power, the reactor power would increase corresponding to the resulting steam load. If a reactor trip occurred, the post-trip condition would approach a hot standby condition similar to what is assumed as the starting point of the analysis documented in this section. However, in the case initiated from power, the initial average coolant temperature would be higher than the no-load temperature and the reactor coolant system and core would contain more stored energy as well as post-trip decay heat. This energy would have to be removed before the reactor reached the temperature assumed in the case analyzed from the hot standby initial condition. Therefore, the peak reactivity condition would be reached at a later time during a transient initiated from power than for the transient initiated from hot standby. The delayed reactivity addition will occur when the steam flow has decayed, further reducing the effect of the cooldown on the transient initiated at power. Also, because the initial steam generator water inventory is greatest at no load, the magnitude and duration of the reactor coolant system cooldown is greater for the transient initiating from hot standby than for a transient initiated from power operation.

15.1.4.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for an inadvertent opening of a steam generator relief or safety valve event:

- The analysis is performed with maximum moderator reactivity feedback (typical of end-of-cycle conditions), when the reactor system cooldown has the largest effect on core reactivity. The relationship between moderator defect (%Δk/k) and moderator density (g/cc) at various boron concentrations used for this event assuming the most reactive RCCA fully withdrawn is provided in Figure 15.1.4-1. The Doppler defect shown in Figure 15.1.4-2 is used for this event to maximize the return to power.

- The initial values of the reactor coolant average temperature 557°F and RCS pressure are assumed to be 2250 psia, respectively, which correspond to hot standby conditions.

- The maximum capacity main steam relief valve, main steam depressurization valve, main steam safety valve, or turbine bypass valve is assumed to instantaneously open. The steam release from this valve is assumed to be

485 lb/s at 1200 psia. The valve is assumed to be upstream of the main steam check valve, resulting in the unterminated blowdown of a single steam generator.

- The shutdown margin is assumed to be 1.6% Δk/k, corresponding to the most limiting condition in the cycle, with the most reactive RCCA in a fully withdrawn position.

- The boron concentration in the refueling water storage pit (RWSP) is assumed to be 4000 ppm. This value is lower than or equal to the minimum allowable Technical Specification value. The accumulators are not actuated for this accident.

- A dry steam blowdown (steam quality = 1.0) is assumed. This assumption maximizes the energy released from the failed-open valve. The Moody curve for f(L/D) = 0 (Ref. 15.1-4) is used for calculating the steam flow from the failed-open valve.

- EFW is assumed to be initiated at time t = 0 and deliver flow at rated capacity for the purpose of maximizing the cooldown. EFW is automatically isolated from the affected steam generator on a low main steam line pressure signal. This assumption increases the primary-to-secondary heat transfer.

- Only two pumps operate to inject borated water from the RWSP into the reactor vessel downcomer. This treatment is consistent with the most severe single active failure, assumed to be one train of the ECCS, and allows for future operational flexibility.

- The core and systems performance analysis conservatively ignores decay heat to provide the maximum RCS cooldown during the transient.

- The nominal primary-to-secondary heat transfer coefficient is used to maximize heat transfer to the secondary. In addition, the reverse heat transfer coefficient is set to zero, so that heat cannot be transferred from the secondary to the primary side if the steam generator temperature is warmer than the primary coolant in the steam generator tubes.

- No credit is taken for the heat capacity of the reactor coolant system or steam generator thick metal in attenuating the resulting plant cooldown. This assumption helps to maximize the heat transfer from the primary to the secondary.

- The time required for borated water to reach the core is determined by taking into consideration: (1) the period from the time the ECCS actuation signal is generated to the time the safety injection pumps reach full speed and (2) transport time for the injected water to pass through the reactor coolant piping. These delays and purge volumes are directly modeled in the MARVEL-M code.

- Offsite power is assumed available for the evaluation starting from hot standby. This assumption is conservative because offsite power allows the reactor coolant pumps to continue operating, maintaining the reactor coolant flow, which maximizes the core cooldown rate.

- The pressurizer is modeled on one of the intact loops.
- Conservative axial power profile and radial power distribution are assumed in the analysis as described in Section 15.0.0.2.3.

15.1.4.3.3 Results

Table 15.1.4-1 lists the key events and the times at which they occur, relative to the initiation of the full-open failure of a main steam relief or safety valve.

Figures 15.1.4-3 through 15.1.4-13 are plots of key system parameters versus time from the Core and System Performance Evaluation for this event.

The reactor briefly becomes critical as shown in Figures 15.1.3-3, 15.1.3-4, and 15.1.3-5 due to the cooldown associated with this event. However, two (of the four) safety injection pumps automatically start and shut down the nuclear reaction by injecting borated water from the RWSP into the reactor vessel downcomer.

The analysis shows that the DNBR remains well above the 95/95 limit for the W-3 correlation. Thus the fuel cladding temperature would not increase significantly during this transient.

The normalized core average heat flux transient is virtually identical to the normalized maximum heat flux used in the DNBR calculations, and is considered representative of both parameters for this event. A plot for reactor vessel inlet temperature (showing all loops) is provided in place of inlet coolant temperature to illustrate the non-uniform inlet temperatures prior to mixing in the reactor vessel inlet. A DNBR state point analysis was performed only at the peak power point, so a DNBR plot is not provided. Because there is significant subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided; these are not key parameters for this event. A plot of steam generator pressures is provided in place of steam line pressure to show the non-uniform and independent response of the steam generators for this event. Containment parameters and pressurizer safety valve flow are not reported for this event because RCS pressure remains below the pressurizer safety valve set pressure and there are no releases from the RCS or steam generators inside containment.

15.1.4.4 Barrier Performance

The inadvertent opening of a steam generator relief or safety valve event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The results of the Core and System Evaluation case demonstrate that the RCS pressure and main steam system pressure remain well below 110% of their respective system design pressures. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.1.4.5 Radiological Consequences

Any radiological releases from this transient are bounded by those from the postulated steam system piping failures inside and outside of containment event, which is described in Section 15.1.5. Radiological doses described in Section 15.1.5 do not exceed the guideline values of 10 CFR 50.34.

15.1.4.6 Conclusions

Inadvertent opening of the largest single main steam relief, depressurization, safety, or turbine bypass valve will cause an introduction of cooler water into the core, thus increasing core reactivity. From hot standby conditions (with the most reactive single control rod postulated to be fully withdrawn), this event can lead to criticality and a brief return to low power commensurate with the steam load caused by the steam release from the valve. A low pressurizer pressure signal initiates the ECCS, which terminates the transient by injecting borated water into the core. The resulting transient, even under the most severe combination of conditions, does not cause the minimum DNBR to decrease below the 95/95 limit.

In addition, the RCS pressure and steam pressure remain well 110% of their system design pressures, so the integrity of the reactor coolant boundary and the integrity of the main steam system pressure boundary are maintained.

The inadvertent opening of a steam generator relief or safety valve event does not lead to a more serious fault condition. Additionally, the radiological consequences of this event are substantially less than that of the postulated steam system piping failures analyzed in Section 15.1.5.

Table 15.1.4-1
Time Sequence of Events for Inadvertent Opening of a Steam Generator Relief or Safety Valve

Event Description	Time [sec]
Inadvertent opening of one main steam relief or safety valve	0
Pressurizer empties	135
Safety injection actuation (low pressurizer pressure ECCS actuation analytical limit reached)	169
Boron reaches core	240

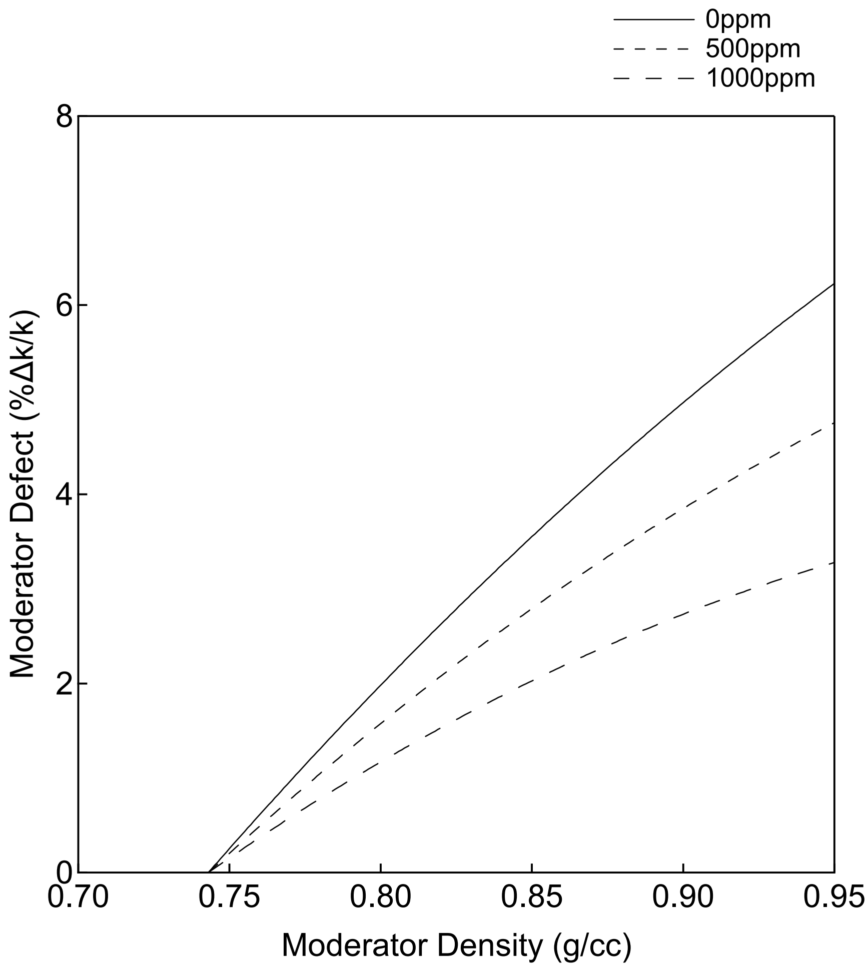


Figure 15.1.4-1 Moderator Defect versus Moderator Density at Various Boron Concentrations for Inadvertent Opening of a Steam Generator Relief or Safety Valve and Steam System Piping Failure

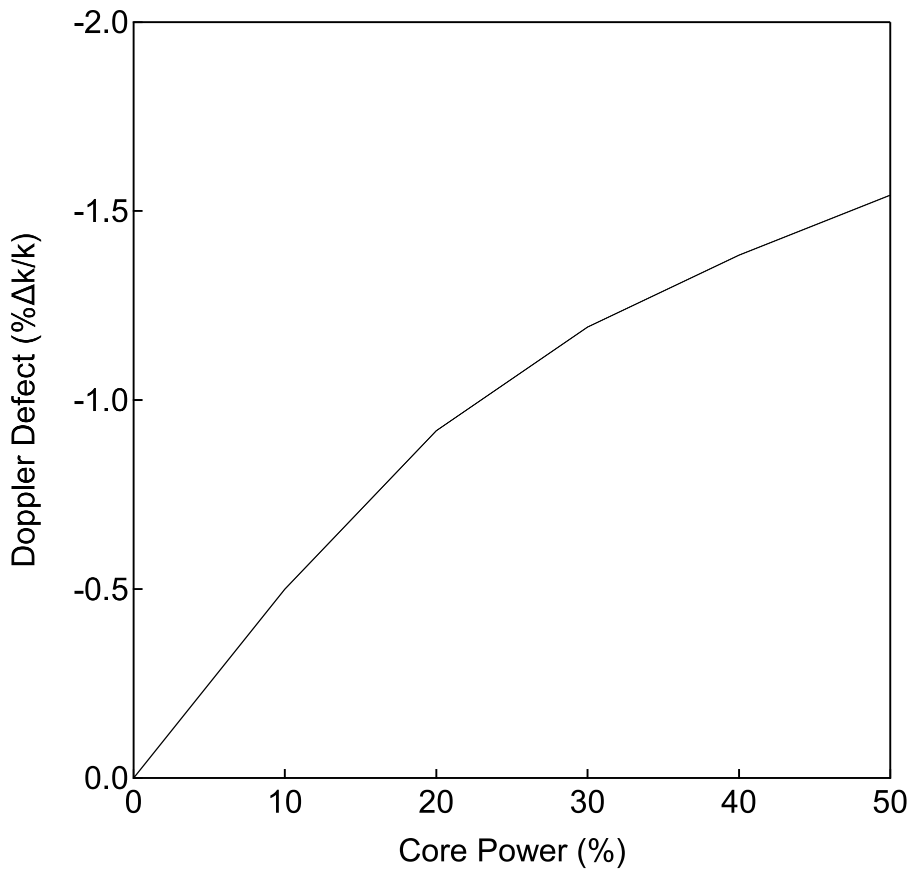


Figure 15.1.4-2 Doppler Defect versus Core Power for Inadvertent Opening of a Steam Generator Relief or Safety Valve and Steam System Piping Failure

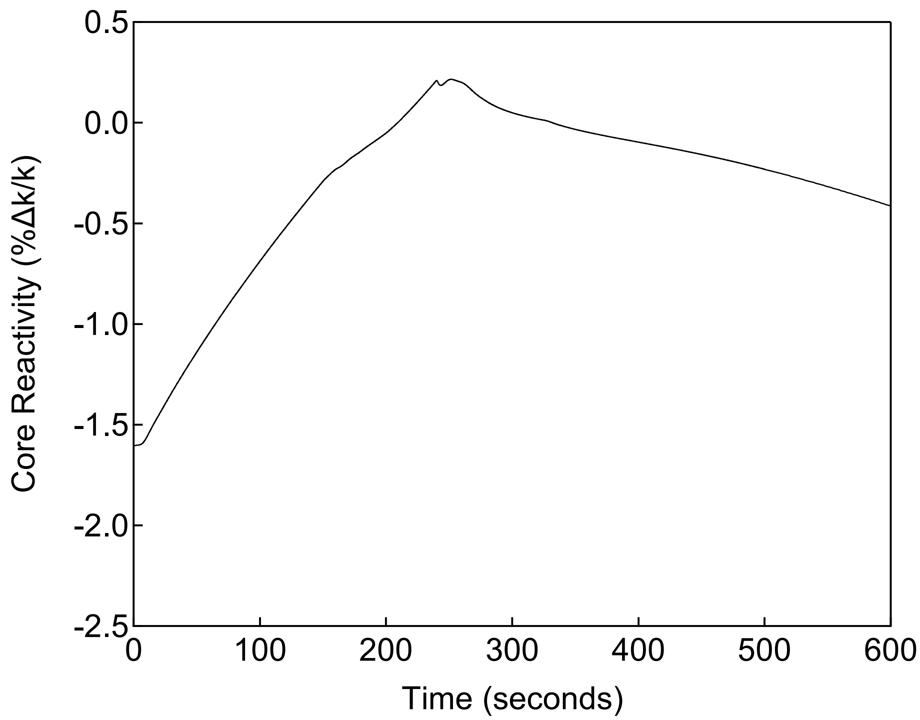


Figure 15.1.4-3 Core Reactivity versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

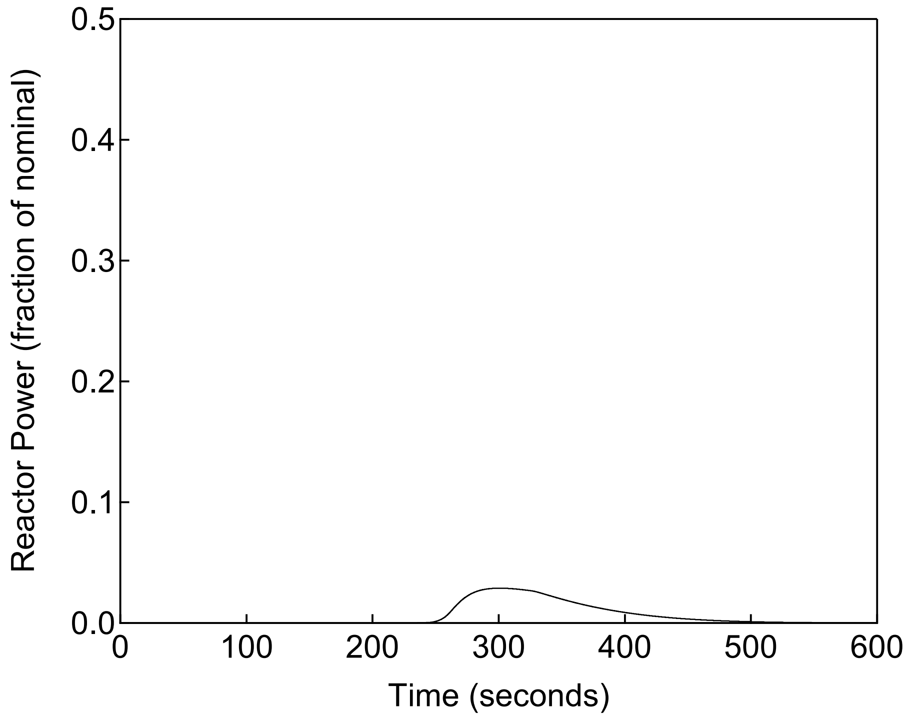


Figure 15.1.4-4 **Reactor Power versus Time**

Inadvertent Opening of a Steam Generator Relief or Safety Valve

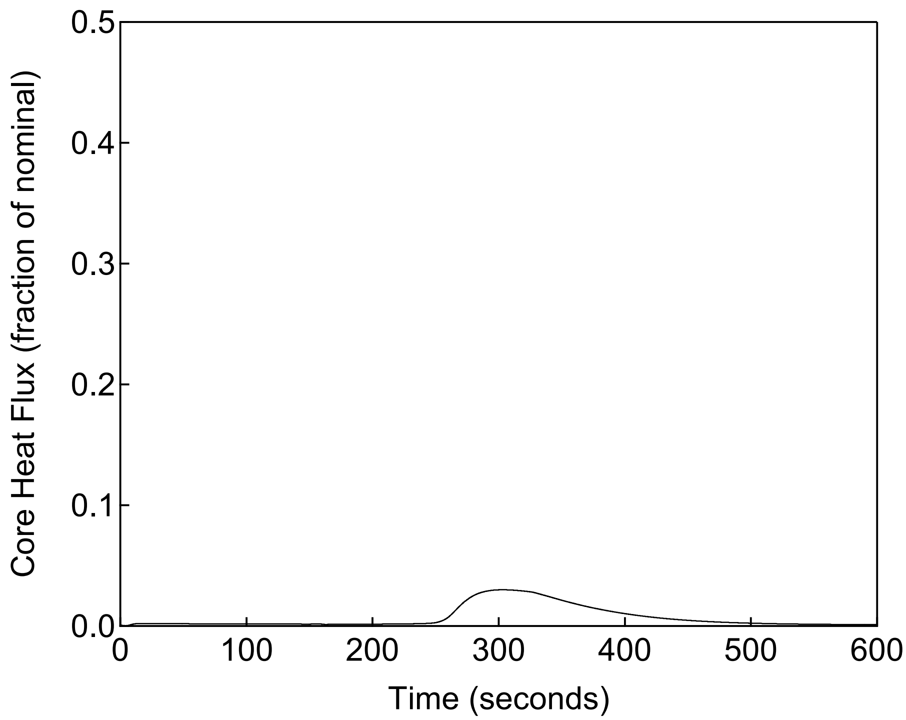


Figure 15.1.4-5 Core Heat Flux versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

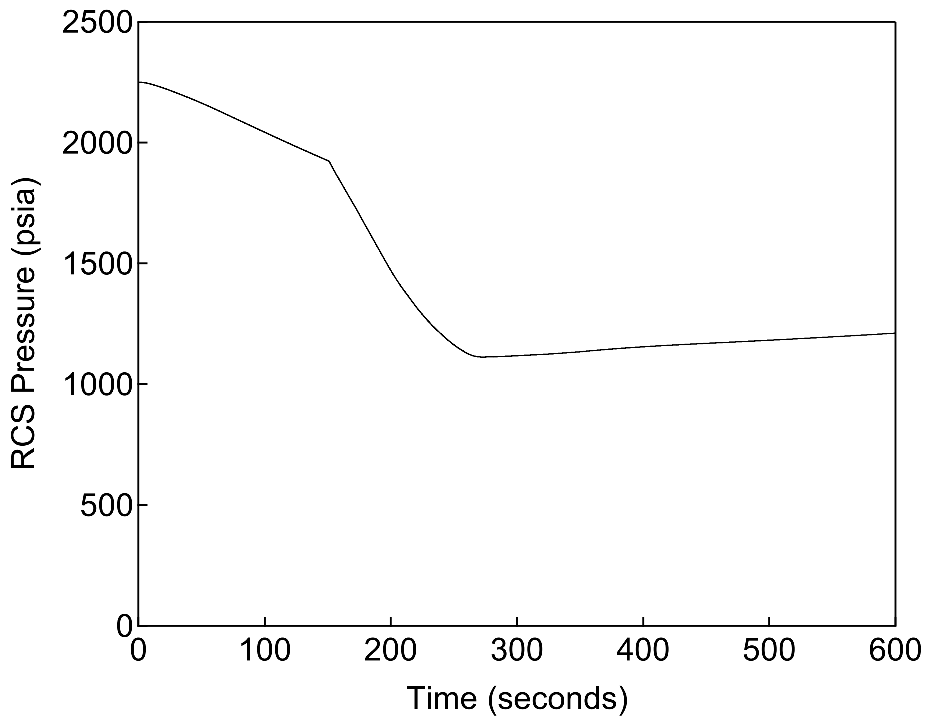


Figure 15.1.4-6 RCS Pressure versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

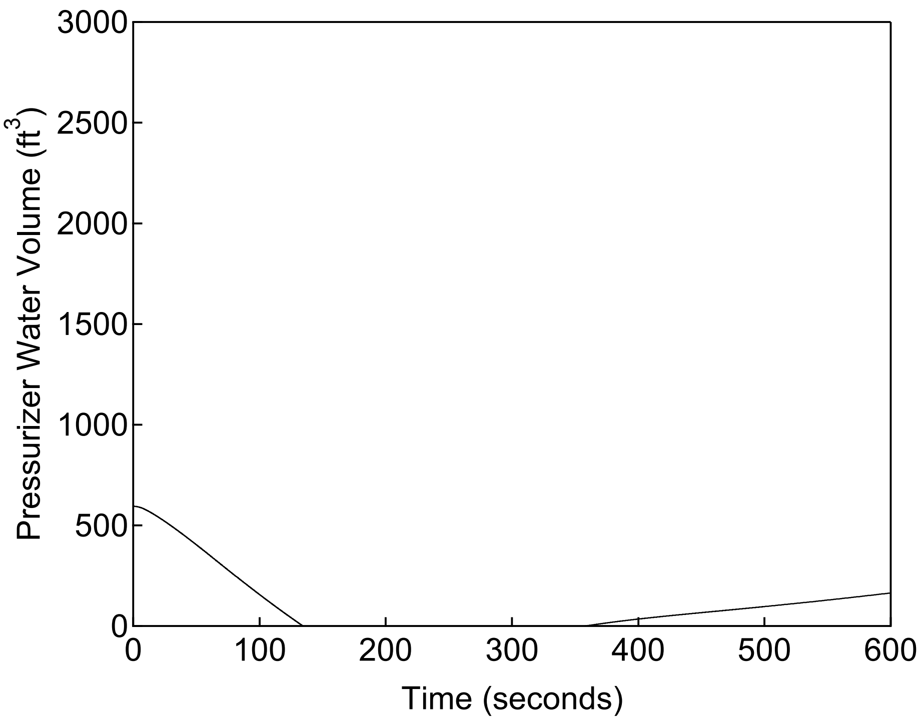


Figure 15.1.4-7 Pressurizer Water Volume versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

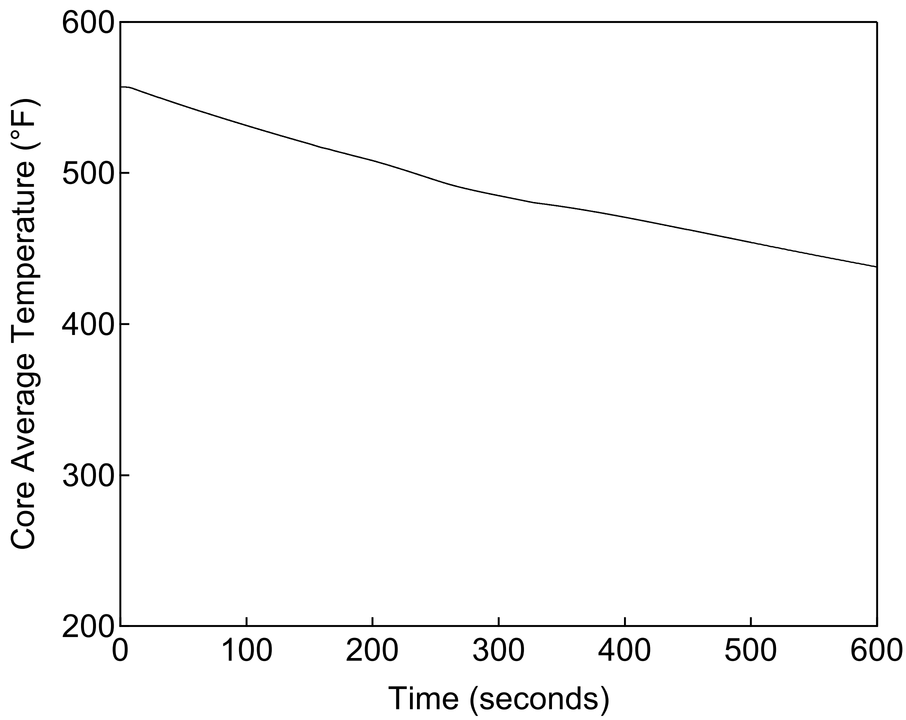


Figure 15.1.4-8 Core Average Temperature versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

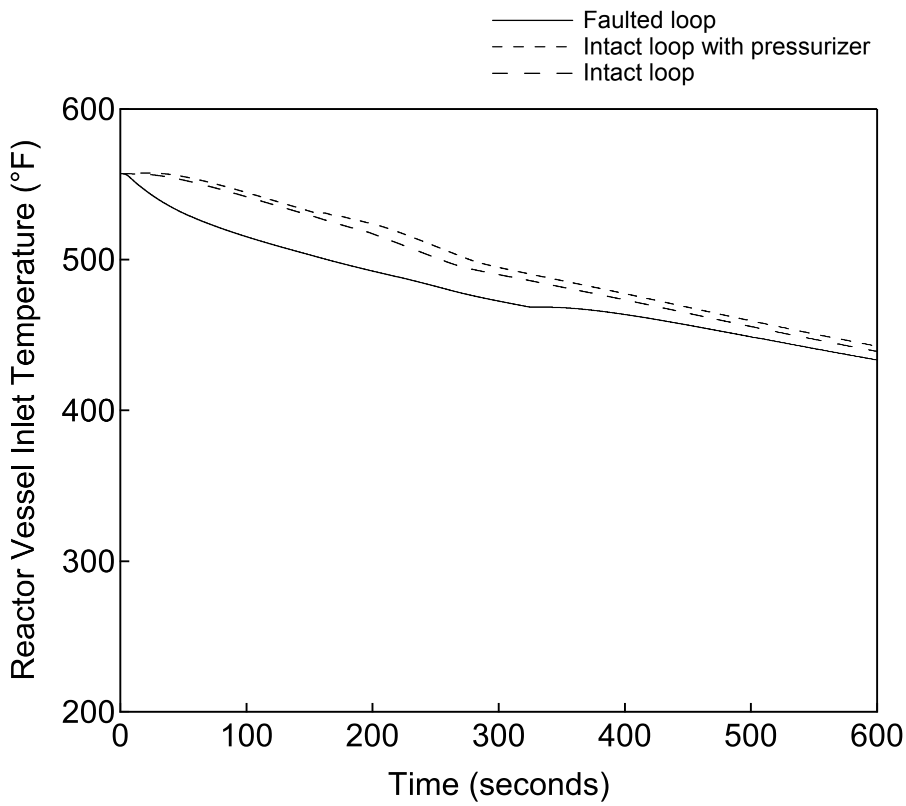


Figure 15.1.4-9 Reactor Vessel Inlet Temperature versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

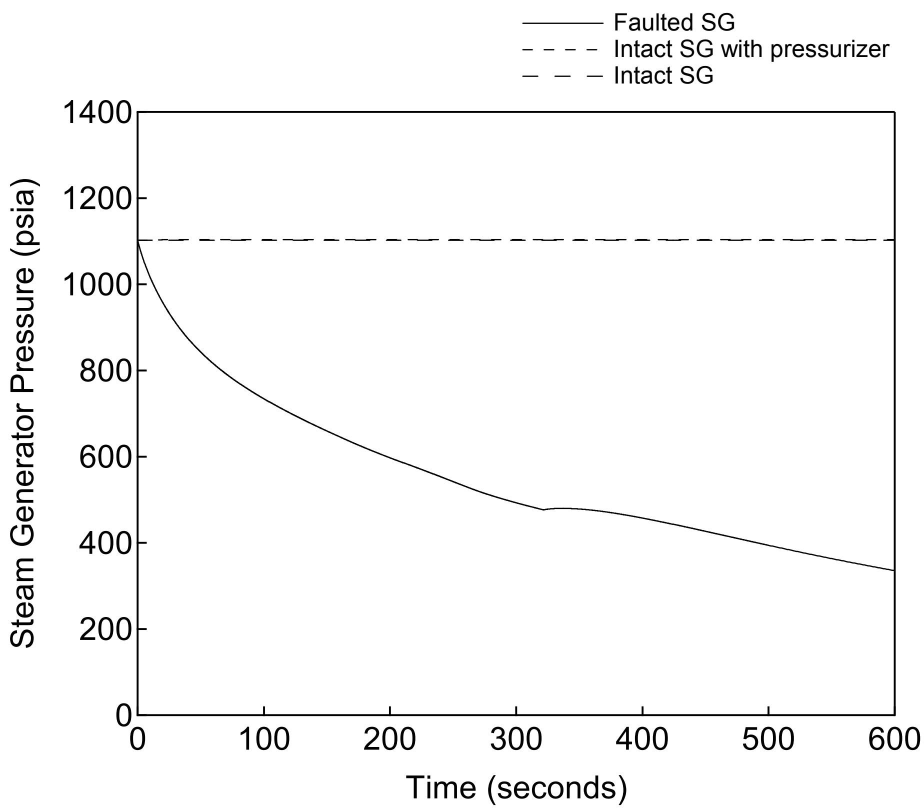


Figure 15.1.4-10 Steam Generator Pressure versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

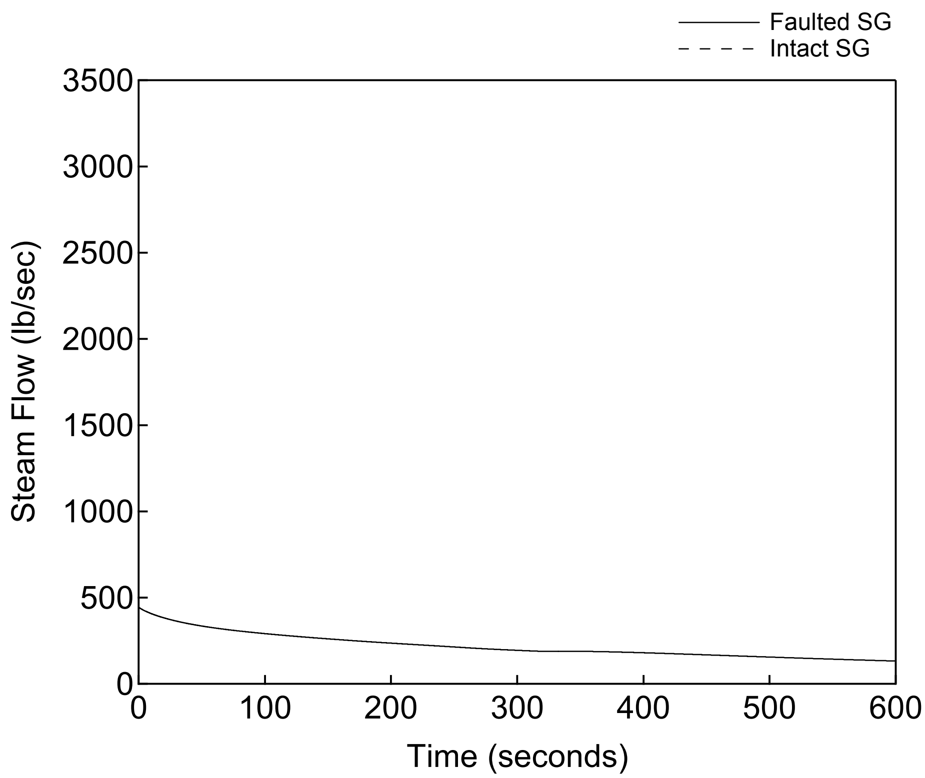


Figure 15.1.4-11 Steam Flow Rate versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

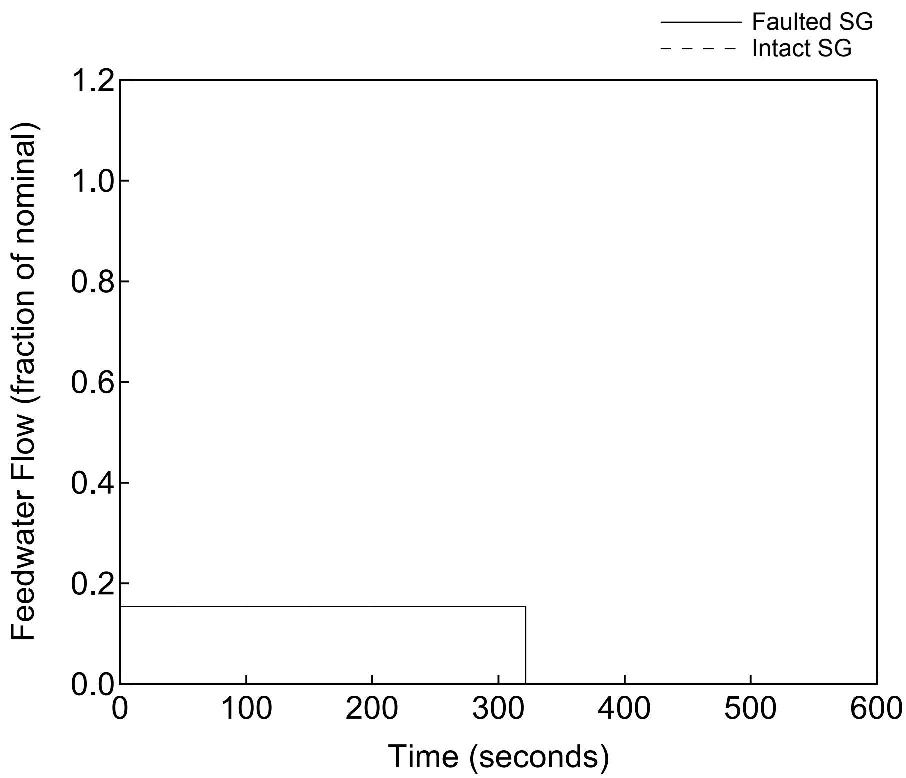


Figure 15.1.4-12 Feedwater Flow Rate versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

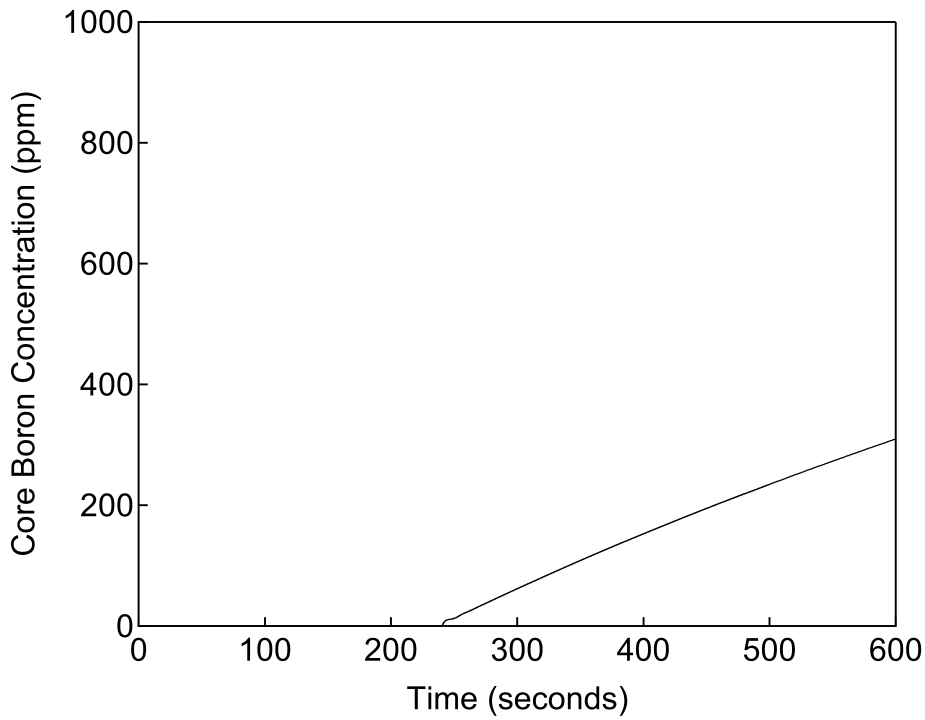


Figure 15.1.4-13 Core Boron Concentration versus Time

Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.5 Steam System Piping Failures Inside and Outside of Containment

15.1.5.1 Identification of Causes and Frequency Classification

The increase in steam generation rate caused by the postulated steam system piping failure removes heat from the reactor coolant system (RCS), which, in turn, lowers the temperature and pressure of the RCS. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase.

This section provides the nuclear steam supply system (NSSS) response analysis for postulated main steam system piping between the steam generator outlet nozzle and the turbine, both inside and outside containment, resulting in steam discharge from the break. The US-APWR design includes both a main steam isolation valve and a check valve in each steam line immediately outside the containment upstream of the main steam header and lines to the turbine. Depending on the combination of break location and single failure assumed, the steam flow may be non-uniform (from only one steam generator) or uniform (all steam generators contribute to the break flow), and may either be automatically isolated or result in an uncontrolled blowdown from one steam generator. Because the emergency core cooling system (ECCS) is actuated for this event, the availability of offsite power is also addressed.

The approach used in the analysis is to define a bounding case that envelopes the various assumptions so that each combination does not require a separate analysis.

This section addresses a spectrum of steam system piping failure sizes and locations from both power operation and hot zero power initial conditions. If the break occurs inside the containment volume, high containment pressure signals are available to actuate ECCS and containment heat removal systems. These signals and containment systems are not used in the core response analysis presented in this section.

The full double-ended failure of a main steam system pipe is classified as a postulated accident (PA) event. Historically, the double-ended steam line break was classified as a Condition IV event as defined by ANSI -N18.2 (Ref. -15.1-1). The Standard Review Plan for the Main Steam System Piping Failure allows fuel failures for this event, subject to including the failed fuel source term in the radiological consequence analysis. The MHI core response analysis conservatively uses a criterion of no DNB (DNBR less than the 95/95 limit) for the limiting steam line break, to preclude addressing DNB propagation in the low pressure environment of the fuel.

Failure of a minor steam system pipe is classified as an anticipated operational occurrence (AOO). Historically, these smaller breaks were defined by ANSI N18.2 as Condition III events. The transient from a failure of a minor steam system pipe is bounded by the analysis described in this section. Event frequency conditions are described in Section 15.0.0.1.

15.1.5.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the analyzed cases for the steam system pipe rupture event is described in the results section.

For the transient initiated from hot zero power (HZP) conditions, the single highest-worth rod cluster control assembly (RCCA) is assumed to be in a fully withdrawn position. The fully withdrawn RCCA is not counted as a single failure. In the presence of a large negative moderator temperature coefficient, the core returns critical and the power peaks at a value depending on break size. RCS pressure decreases below the shutoff head of the ECCS, resulting in the addition of borated water to the RCS. The core is ultimately shut down by a combination of the high concentration boric acid water delivered by the ECCS and the termination of the cooldown when the steam generator inventory is depleted. The RCS pressure does not decrease below the accumulator charge pressure; therefore, the accumulators are not credited in the analysis. The HZP case uses bounding assumptions that assure that the post-trip portion of main steam line break events initiated from power is bounded by the HZP analysis.

The limiting single failure for the event initiated from hot standby conditions is the failure of one ECCS train. Two of the remaining trains are assumed to operate to provide the safety injection functions credited in this analysis. A single failure of one train of either the reactor trip system (RTS) or engineered safety features (ESF) actuation system will not affect the analysis since any one of the other trains will provide the protective functions credited in the analysis.

The limiting single failure for the event initiated from power is the failure of one train of the RTS. If this occurs, the other three trains remain functional. Any one of the remaining trains is adequate to provide the protection functions credited in this analysis.

Besides the limiting single failures defined above, certain other single failures or combinations of failures in mitigating systems have been assumed as part of the definition of the analyzed cases for the purpose of defining bounding or limiting cases to reduce the number or complexity of the analyzed cases. Prior to steam line isolation, the affected loop steam generator is assumed to blow down with its flow defined by the 1.4 ft^2 flow restrictor integral to the steam generator outlet nozzle. During that time, the "intact" steam generators are assumed to blow down through the opposite end of the double-ended rupture with their flows defined by their respective 1.4 ft^2 flow restrictors integral to the steam generator outlet nozzle. A realistic response would not result in blowdown of the "intact" steam generators due to the main steam check valve in the affected steam line. However, this conservative assumption bounds both the case where the affected loop check valve is assumed to fail and the case of a break downstream of the main steam isolation valves and a failure of one main steam isolation valve. In summary, the following are examples of single failures that have been addressed as part of the definition of the bounding case:

- Failure of one ECCS mechanical train (reduced safety injection flow)
- Failure of one main steam check valve for a break upstream of the check valve on one steam line (single steam generator blowdown after steam line isolation)

- Failure of one main steam isolation valve for a break [in non-seismic piping] downstream of the main steam isolation valves (single steam generator blowdown after steam line isolation)
- Failure of one reactor trip system train (no impact)
- Failure of one ESF actuation train (no impact)

The following paragraphs describe the progression of the rupture of a main steam line initiated from hot standby.

The major steam pipe rupture results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. Although an integral flow restrictor is installed in the steam generator exit nozzle to mitigate the steam flow, a double-ended steam line break causes a large steam flow from the faulted steam generator to induce a rapid cooldown of the steam generator secondary side. The energy removal from the reactor coolant system causes a reduction in coolant temperature and pressure. The colder fluid in the loop with the faulted steam generator is mixed with the flow from the other intact loops in the reactor vessel inlet plenum. The core inlet temperature distribution and the cooldown of the core water are non-uniform due to the imperfect mixing of the loop flows in the reactor vessel inlet.

In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. The effect is the largest at the end-of-cycle. If the event occurs at nominal operating conditions, a core power increase results. If the event occurs at hot zero power condition with 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 is a potential problem mainly because of the existing high radial power peaking factors, assuming the most reactive RCCA to be stuck in its fully withdrawn position.

When the steam pressure in the faulted steam generator falls below the low main steam line pressure setpoint (in any loop), the ECCS is actuated and the main steam isolation valves are closed. The ECCS signal also actuates emergency feedwater (EFW) and feedwater isolation to isolate the steam generators from each other.

The core is ultimately shut down by a combination of the high concentration boric acid water delivered by the ECCS and the termination of the cooldown when the steam generator inventory is depleted.

<u>Systems Operation Assumptions</u>

The following automatic reactor trip signals are assumed to be available to provide protection from this transient (but are not necessarily credited in the analysis):

- ECCS actuation
- Over power ΔT
- Over temperature ΔT
- Low pressurizer pressure
- High power range neutron flux

The following signals could actuate the ECCS, which injects borated water into the reactor vessel via the safety injection pumps:

- Low pressurizer pressure
- Low main steam line pressure (any one loop)
- High containment pressure

An ECCS actuation signal provides feedwater isolation by automatically tripping the main feedwater pumps and fully closing all control valves and feedwater isolation valves in the feedwater system. The signal also starts the EFW. In addition to the reactor trips listed above, the following engineered safety feature functions are assumed to be available to mitigate the accident:

- Main steam line isolation
- EFW isolation
- ECCS
- Main feedwater isolation

The automatic reactor coolant pump (RCP) trip will actuate on an ECCS actuation signal generated from low pressurizer pressure, low main steam line pressure, or high containment pressure. The core response for the limiting steam system piping failure event is analyzed with and without offsite power as described in Section 15.1.5.3 below (Cases A and B). The RCP trip is ignored for the case with offsite power available to maximize the RCS cooldown and associated reactivity and return-to-power response. For the case assuming loss of offsite power, the RCP trip is assumed to occur on the ECCS actuation signal as designed. No operator action is required to trip the RCPs for this event.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

To prevent excessive cooldown of the reactor coolant, the main feedwater regulation valves are fully closed by the reactor trip coincident with a low reactor coolant average temperature (P-4) signal. Also, the ECCS actuation signal automatically trips the main feedwater pumps, and fully closes all the control valves and main feedwater isolation valves.

Main steam check valves are provided downstream of the main steam isolation valves to prevent blowdown of the steam generators by reverse flow through the postulated piping failure in the event the break is upstream of a main steam check valve. The main steam isolation valves, which provide positive flow isolation in the normal direction of flow, are fully closed by the following signals:

- Low main steam line pressure
- High main steam line pressure negative rate
- High-high containment pressure

The main steam isolation valves are assumed to close within 5 seconds after receiving a main steam line isolation signal, in accordance with Table 15.0-5. After this time, the three intact loops are isolated and one steam generator continues to blow down with its flow defined by the 1.4 ft^2 flow restrictor integral to the steam generator outlet nozzle.

Only two safety injection trains are assumed to operate to inject borated water into the reactor vessel.

The time required for borated water to reach the core is determined by taking into consideration: (1) the period from the time the ECCS actuation signal is generated to the time the safety injection pumps reach full speed and (2) transport time for the injected water to pass through the reactor coolant piping. The time for the safety injection pumps to reach full speed includes time for the emergency gas turbine generators to start for the case where offsite power is not available. ECCS signal delays, backup power start delays, and safety injection piping and purge volumes are modeled by the MARVEL-M code.

The long-term shutdown can be provided by using the chemical and volume control system to borate the reactor coolant system. This is not credited in the analysis.

Cases assuming offsite power available and unavailable (Cases A & B) are analyzed for the limiting double-ended break from hot standby conditions as described in the following sections.

Only safety-related equipment is credited in the analysis to mitigate the consequences of this event.

15.1.5.3 Core and System Performance

15.1.5.3.1 Evaluation Model

The MARVEL-M, ANC, and VIPRE-01M codes are used for this steam system piping failure analysis. The MARVEL-M, ANC, and VIPRE-01M codes are described in Section 15.0.2.2. Reference 15.1-2 contains additional detailed information about the evaluation models used for this transient.

(1) System Analysis by the MARVEL-M Code

The MARVEL-M code is used to analyze the plant transient following steam piping ruptures. The break flow rate from the steam generators is calculated using the Moody correlation. The released steam is conservatively assumed saturated and dry without moisture carry-over, since steam release without carry-over causes the maximum energy release and cooldown.

The overall primary-to-secondary heat transfer coefficient in the steam generators is modeled in the code by the four major thermal resistance components: the primary convection film resistance, the tube metal resistance, the fouling resistance, and the secondary side boiling heat transfer resistance, taking account of the dependency on the relevant operating conditions, such as temperature, pressure and flow. The model is applicable over the wide range of the operating conditions characteristic of the steam generator during a steam pipe break event.

The RCS model in the code can analyze the non-uniform reactor system transient response to the event. The steam system model in the code can simulate steam flow redistribution from the steam generators (described in Ref. 15.1-2, Section 2.1.3.4). The flow mixing in the reactor vessel is modeled in the code. The mixing factor for the reactor vessel inlet plenum is defined conservatively to reflect imperfect mixing (Ref. 15.1-2, Section 2.1.3.2).

A weighting factor for calculating core reactivity can be also input to take account of the azimuthal tilt of the core coolant properties.

The ESF actuation system and the ESF sub-systems necessary for the steam line break analysis are modeled in the MARVEL-M code.

The steam system piping failure inside the containment may cause a containment pressure increase due to the steam release. Containment volume response and the ECCS containment signals and isolation functions are not modeled in the MARVEL-M code and are not credited in the core response analysis.

(2) DNBR calculation

In the hot zero power condition, the VIPRE-01M code calculates the minimum DNBR. These DNBR calculations are steady state calculations at pre-selected state point conditions, using the MARVEL-M calculated values of core average heat flux, RCS pressure, inlet core flow rate and the core inlet temperatures, for a certain number of state points around the time the highest core average heat flux is reached. Additionally, the core inlet coolant enthalpy distribution coupled with core power distribution, which is calculated by the core design code ANC considering a steady-state condition assuming a stuck rod, is also input to the VIPRE-01M code. The history files used in the more standard MARVEL-M / VIPRE-01M sequences are not used for the steam piping failure. A suitable bundle DNB correlation is used at the low RCS pressure conditions characteristic of this accident. Because the RCS pressures are below the applicable pressure range for the WRB-2 DNBR correlation, the W-3 correlation (See Sections 4.4.2.2.1 and 4.4.4.1) and its associated 95/95 limit are used.

For the hot full power condition, the MARVEL-M code calculates the minimum DNBR using its internal DNBR data tables, with core average heat flux, RCS pressure, and core inlet temperature in the same manner as is used for the RCCA Bank Withdrawal at Power described in Section 15.4.2. The internal DNBR table used is evaluated by using RTDP and applicable WRB-2 rod bundle DNB correlation. This methodology is acceptable, since the core operating condition is within the range of the pre-evaluated DNBR table in MARVEL-M and because the minimum DNBR occurs within a short time after the reactor trip is initiated. The one rod stuck assumption is considered in defining the shutdown reactivity, but is not meaningful for the period up to reaching the minimum DNBR in the at-power transients of this kind. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

(3) General

A double-ended guillotine break of a main steam line upstream of the main steam isolation valve and main steam check valve is assumed. The largest break will result in both the largest return to power as well as the largest radial peaking factor. The analysis is performed with maximum reactivity feedback (typical of end-of-cycle conditions), when the reactor system cooldown has the largest effect on core reactivity.

Cases are considered where the reactor is assumed to be initially in hot standby and at rated power. Three cases are evaluated:

- Case A – Double-ended break from hot standby with offsite power (reactor coolant pumps continue to operate)
- Case B – Double-ended break from hot standby without offsite power (all reactor coolant pumps begin coast down at the time of ECCS actuation)
- Case C – Spectrum of breaks from power with offsite power (reactor coolant pumps continue to operate)

The hot standby case is generally more severe than the post-trip situation following a steam pipe failure initiated from rated power because it generates a event where the core can reach criticality and return to power at an earlier time that for a event that starts during power operation. The basis for this is described below.

If the rupture of a main steam line were to occur when the reactor was at power, the reactor power would increase corresponding to the resulting steam load. If a reactor trip occurred, the post-trip condition would approach a hot standby condition similar to what is assumed in the analysis documented in this section. However, in the case initiated from power, the initial average coolant temperature would be higher than the no-load, and the reactor coolant system and core would contain more stored energy, as well as post-trip decay heat. This energy would have to be removed before the reactor reached the no-load temperature assumed in the case analyzed from the hot standby condition. Therefore, the peak reactivity condition would be reached at a later time than for the transient initiated from hot standby. The delayed reactivity addition will occur when the steam flow has decayed, further reducing the effect of the cooldown on the transient initiated at power. Also, because the steam generator water inventory is greatest at no load, the magnitude and duration of the reactor coolant system cooldown is greater for the transient initiating from hot standby than for a transient initiated from power operation.

The hot standby transient analysis considers cases with and without offsite power available. Because the plant design includes an automatic reactor coolant pump trip on an ECCS actuation signal, the RCPs would actually coast down whether or not offsite power was available. For the case with offsite power available, the reactor coolant pumps are assumed to run for the duration of the transient and provide nominal RCS flow. This is conservative in that full flow enhances primary-to-secondary heat transfer, maximizing the adverse effect of reactivity, power, and non-uniform inlet temperature on the core. This bounds the expected response where the reactor coolant pumps trip as designed on ECCS actuation. The case without offsite power includes a coastdown of the reactor coolant pumps at the time of the ECCS actuation signal, consistent with the design of the automatic reactor coolant pump trip logic. The loss of offsite power case also results in a later start time for the safety injection pumps due to the additional time delay associated with the startup and loading of the emergency gas turbine generators.

15.1.5.3.2 Input Parameters and Initial Conditions

(1) Cases A and B – Double-Ended Breaks from Hot Standby

Because the RCS pressures are below the applicable pressure range for the WRB-2 DNBR correlation, the W-3 correlation and its associated 95/95 limit are used to calculate the DNBR at the peak heat flux point using the VIPRE-01M code as described in Section 15.1.5.3.1.

The following input parameters and initial conditions are used in the MARVEL-M analysis:

- The analysis is performed with maximum moderator reactivity feedback (typical of end-of-cycle conditions), when the reactor system cooldown has the largest effect on core reactivity. The relationship between moderator defect (%Δk/k) and moderator density (g/cc) at various boron concentrations used for this event assuming the most reactive RCCA fully withdrawn is provided in Figure 15.1.4-1. The Doppler defect shown in Figure 15.1.4-2 is used for this event to maximize the return to power.

- The initial values of reactor coolant average temperature and RCS pressure are assumed to be 557°F and 2250 psia, respectively, which correspond to hot standby conditions.

- The reactivity shutdown margin is assumed to be 1.6% Δk/k corresponding to the most restrictive time in the cycle, with the most reactive RCCA in the fully withdrawn position.

- The boron concentration in the refueling water storage pit (RWSP) is assumed to be 4000 ppm, corresponding to the minimum allowable Technical Specification boron concentration value.

- A dry steam blowdown (steam quality = 1.0) is assumed. This assumption maximizes the energy released form the break. The Moody curve for f(L/D) = 0 (Ref. 15.1-4) is used for calculating the steam flow from the break.

- EFW is assumed to be initiated at time t = 0 and deliver flow at rated capacity for the purpose of maximizing the cooldown. EFW is automatically isolated from the affected steam generator on a low main steam line pressure signal. This assumption increases the primary-to-secondary heat transfer.

- Only two pumps operate to inject borated water from the RWSP into the reactor vessel downcomer. This treatment is consistent with the most severe single active failure, assumed to be one train of the ECCS, and allows for future operational flexibility.

- The core and systems performance analysis conservatively ignores decay heat to provide the maximum RCS cooldown during the transient.

- The nominal primary-to-secondary heat transfer coefficient is used to maximize heat transfer to the secondary. In addition, the reverse heat transfer coefficient is set to zero, so that heat cannot be transferred from the secondary to the primary side if the steam generator temperature is warmer than the primary coolant in the steam generator tubes.

- The time required for borated water to reach the core is determined by taking into consideration: (1) the period between the time the ECCS actuation signal is generated and the time the safety injection pumps reach full speed and (2) the time for the injected water to pass through the reactor coolant piping. These delays and purge volumes are directly modeled in the MARVEL-M code.

- No credit is taken for the heat capacity of the reactor coolant system or steam generator thick metal in attenuating the resulting plant cooldown. This assumption helps to maximize the heat transfer from the primary to the secondary.

- The pressurizer is modeled on one of the intact loops.

- Conservative axial power profile and radial power distribution are assumed in the analysis as described in Section 15.0.0.2.3.

- For Case A with offsite power available, the reactor coolant pumps are assumed to run for the duration of the transient and provide nominal RCS flow. This is conservative in that full flow enhances primary-to-secondary heat transfer, maximizing the adverse effect of reactivity, power, and non-uniform inlet temperature on the core. This assumption bounds the case where the reactor coolant pumps trip as designed on ECCS actuation.

- For Case B without offsite power, the reactor coolant pumps are assumed to trip as designed on ECCS actuation.

(2) Case C – Spectrum of Breaks from Power with Offsite Power

DNBR is calculated the same way as for the RCCA Bank Withdrawal at Power described in Section 15.4.2 (using internal data tables based on RTDP calculations using the WRB-2 DNBR correlation). Each analysis is terminated shortly after reactor trip. The post-trip core response is bounded by Cases A and B.

Similar to the breaks analyzed from hot standby, the break flow is shared by all the steam generators until steam line isolation occurs. After that time, one steam generator is assumed to continue to blow down. Each analysis is terminated shortly following the reactor trip for the purpose of confirming that the DNBR does not exceed the 95/95 limit.

- Consistent with the use of RTDP, the initial values of reactor power, reactor coolant average temperature, and RCS pressure are assumed to be the nominal values without uncertainties as defined in Table 15.0-3.

- Because this analysis is to evaluate the phase of the transient from the time the break occurs to stable post-trip conditions, the moderator density coefficient and Doppler defect used in Cases A and B are not used. As shown in Table 15.0-1, the moderator density coefficient is assumed to be 0.51 (Δk/k)/(g/cc) and the Doppler power coefficient is assumed to be the minimum feedback curve in Figure 15.0-2.

- Offsite power is assumed to be available, resulting in the maximum positive reactivity and power increases. The analysis from at-power conditions results in a turbine trip. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause an RCP coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, reactor trip system signal processing delays) are used in the analysis. RCCA insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- The reactor is assumed to be automatically tripped by either the ECCS actuation signal due to low main steam line pressure (larger breaks) or the over power ΔT signal. Table 15.0-4 summarizes the trip setpoint and signal delay times used in the analysis.

15.1.5.3.3 Results

(1) Case A - Hot Standby, Double-Ended Rupture with Offsite Power

Figures 15.1.5-1 through 15.1.5-12 provide plots of system parameters versus time from the core response analysis for the double-ended steam line failure from hot standby with offsite power available. The corresponding sequence of events is provided in Table 15.1.5-1. As described in Section 15.1.5.2, the piping failure analyzed is a case that bounds a combination of break locations and single failures. The piping failure is assumed to result in the blowdown of all steam generators at their maximum flow prior to steam line isolation, followed by the continued blowdown of one steam generator after steam line isolation. If the core is at critical hot zero power conditions when the piping failure occurs, the low main steam line pressure signal will trip the reactor, leading to a transient much like the case presented here.

Immediately following the piping failure, a low main steam line pressure (any one steam generator) signal will occur on one or more of the loops, resulting in steam line isolation and reactor trip. The low main steam line pressure signal also causes an ECCS actuation signal, which in turn, starts the emergency feedwater pumps, isolates main feedwater, and starts the safety injection pumps.

As shown in Figure 15.1.5-1, the reactor becomes critical with the control rods inserted (assuming the single most reactive rod in the fully withdrawn position), and the reactor returns to power. The cooldown continues at a decreasing rate due to decreasing steam pressure, until the affected steam generator inventory is depleted and emergency feedwater to it is isolated. As can be seen from Figures 15.1.5-7 and 15.1.5-8, the intact loop with the pressurizer surge line connection responds differently from the other two intact loops due to the addition of warmer pressurizer outsurge to the hot leg. The steam generator pressure in the pressurizer loop remains higher than the other intact loops after steam line isolation due to this effect and the assumption of no reverse heat transfer from the steam generators to the RCS. When the RCS pressure decreases below the shutoff pressure of the safety injection pumps, borated water begins to flow to the RCS, as indicated by the boron concentration transient shown in Figure 15.1.5-12. A single failure of one safety injection train is assumed in the analysis. The limiting point in the transient occurs when the reactor power and core heat flux peak, resulting from the combination of decreasing steam flow and increasing core boron concentration.

Only one steam generator blows down completely following a steam pipe failure transient. EFW to the affected steam generator is isolated automatically on an uncompensated steam generator pressure signal as shown in Figure 15.1.5-11. As shown in Figure 15.1.5-9, the blowdown is terminated when the affected steam generator mass is depleted, terminating the rapid cooldown. After the faulted steam generator mass is depleted, the pressurizer level recovers and the differences in loop inlet temperature decrease due to mixing in the reactor vessel as shown in Figures 15.1.5-5 and 15.1.5-7. The other three steam generators remain available for removal of decay heat from the primary coolant after the initial transient is over.

The analysis shows that the minimum DNBR remains above the 95/95 limit for the W-3 correlation. Thus the fuel cladding temperature would not increase significantly during this transient.

Since offsite power is available in Case A, full reactor coolant flow exists.

Long term decay heat can be removed by controlled steam relief from the intact steam generators and later, by the residual heat removal system as described in Section 15.0.0.8.

(2) Case B - Hot Standby, Double-Ended Rupture without Offsite Power

Figures 15.1.5-13 through 15.1.5-25 provide plots of system parameters versus time from the core response analysis for the double-ended steam line failure from hot standby without offsite power available. The corresponding sequence of events is provided in Table 15.1.5-1. The case is the same as Case A except that a loss of offsite power is assumed to trip when the ECCS actuation signal occurs, resulting in a reactor coolant pump coastdown. For the large break, this occurs almost immediately. There is also an additional delay for the safety injection pump start to allow the startup of the standby emergency gas turbine generators as described in Table 15.0-4.

The lower RCS flow generally causes the reactor coolant system cooldown for Case B to be slower and the associated reactivity transient is less severe, resulting in a later return to criticality and lower peak power. Most of the parameters behave in a similar manner as in Case A. As with Case A, long term decay heat can be removed by controlled steam relief from the intact steam generators and later, by the residual heat removal system.

The minimum DNBR in Case B is less limiting than the minimum DNBR in Case A because of the reduced core cooling and lower heat flux. However, the minimum DNBR remains well above the 95/95 limit for the W-3 correlation and the fuel cladding temperature would not increase significantly during this transient. This accident does not challenge the design pressures for either the reactor coolant pressure boundary or the main steam system.

(3) Case C – Spectrum of Breaks from Power with Offsite Power

A spectrum of break sizes from at-power conditions was analyzed to demonstrate that the period of the transient before post-trip shutdown does not result in DNBRs below the 95/95 limit.

At rated power, the increased reactivity causes an increase in core power. For small breaks, the response is similar to the steam flow increase event in that the power may not reach a reactor trip setpoint. For intermediate size breaks, the power increase results in an over power ΔT reactor trip. For large breaks, up to and including the double-ended rupture of a steam pipe, the reactor is tripped on low steam line pressure, which also causes ESF actuation (including safety injection, main feedwater isolation, and emergency feedwater isolation).

Figure 15.1.5-26 provides a summary of the key results of this analysis of the at-power breaks in the form of plots of initial steam flow, peak power, and minimum DNBR as a function of break area (per steam generator). A line is included on each plot for 100% and 75% initial power levels. As expected, the initial break flow is only a function of initial steam generator pressure and break area, so initial break flow decreases with decreasing break area, and some break areas are small enough that the feedwater control system may be able to keep up with the steam flow, resulting in a new steady state power below the overpower reactor trips. The peak power and minimum DNBR curves from 100% power, however, show three distinct regions: no trip for small breaks, over power ΔT trips for intermediate break sizes, and low main steam line pressure trips (and main steam line isolation) for the larger breaks. The low main steam line pressure signal occurs so rapidly for the larger breaks that the reactivity feedback has not caused power to increase before the trip and steam line isolation occur (the peak power and minimum DNBR are approximately equal to the initial full power value).

Figure 15.1.5-26 also shows that breaks initiating from lower power levels are less limiting than for full power.

The analysis shows that the minimum DNBR remains above the 95/95 limit for the WRB-2 correlation. Thus the fuel cladding temperature would not increase significantly during this transient. These cases do not challenge the design limits for the reactor coolant pressure boundary or the main steam system.

For Cases A and B, the normalized core average heat flux transient is virtually identical to the normalized maximum heat flux, and is considered representative of both parameters for this event. A plot for reactor vessel inlet temperature (showing all loops) is provided in place of inlet coolant temperature to illustrate the non-uniform inlet temperatures prior to mixing in the reactor vessel inlet. Because there is significant core subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions, peak cladding temperature, and fuel centerline temperature are not provided; these are not key parameters for this event. A plot of steam generator pressure is provided in place of steam line pressure to show the non-uniform and independent response of the steam generators during this event. Steam generator water mass is presented instead of steam generator water volume. Additionally, steam line break flow rate is labeled as steam flow rate for these cases. Pressurizer safety valve flow is not reported for this event because RCS pressure remains below the pressurizer safety valve set pressure and there are no releases from the RCS inside containment. Containment parameters are not presented for these core response analyses. Containment vessel response to steam system piping failures inside the containment vessel is described and analyzed in Section 6.2.

For the Case C at-power breaks, the RCS parameter transient response up to the time of reactor trip are similar to those for the RCCA Bank withdrawal at Power transients presented in Section 15.4.2, and the DNBRs are calculated using the same approach. Therefore, only a plot summarizing the results (peak power, initial break flow, and minimum DNBR versus break area at two power levels) is provided.

15.1.5.4 Barrier Performance

Information for the bounding transient documented in Section 15.1.5.3 indicates the maximum reactor coolant system and main steam pressures remain well below 110% of their design pressures.

In response to GDC 31, this event (including operation of the ECCS under low-temperature conditions) has been considered in the design of the reactor coolant pressure boundary to assure that the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture has been minimized. Fracture toughness of the reactor coolant pressure boundary and reactor vessel is described in Sections 5.2.3 and 5.3.1.

As discussed in Section 15.0.0.9, the integrity of the reactor coolant pumps is maintained such that loss of ac power and containment isolation will not result in pump seal damage.

The effects of blowdown loads from steam system piping failures (e.g., pipe whip and jet impingement) on plant structures, systems, and components is evaluated in Section 3.6.2.

Therefore the integrity of the reactor coolant pressure boundary and main steam system pressure boundary is maintained.

The containment vessel response to steam piping failures inside containment is described and analyzed separately in Section 6.2.

15.1.5.5 Radiological Consequences

The radiological consequences evaluation for this transient is based on the alternative source term (AST) guidance documented in Reference 15.1-3.

The radiological consequences evaluation assumes the reactor has been operating with a small number of defect fuels and leaking steam generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. The equilibrium concentrations assumed in the analysis are based on technical specification coolant concentration limits.

The steam releases used in the radiological assessment corresponds to the hot zero-power case with offsite power unavailable. The analysis documented in this section conservatively assumes the reactor is cooled by releasing steam from the non-faulted (intact) loops.

Following the postulated pipe failure, emergency feedwater to the faulted loop is isolated and the steam generator in that loop is allowed to steam dry. Radioiodines carried from the reactor coolant to the generator in the faulted loop via leaking tubes are assumed to be released directly to the environment. Iodine released from the intact loops via the main steam safety valves or main steam relief valves are assumed to be mixed with the secondary coolant and partitioned between the generator liquid and steam before release to the environment. Noble gases entering the secondary (all steam generators) are assumed to be released directly to the environment.

The evaluation of the offsite radiological consequences of a postulated main steam pipe failure conservatively takes no credit for filtration (i.e., the evaluation essentially assumes the location of the failure is outside containment and that one steam system isolation valve fails to close, so that one steam generator blows down through the postulate piping system break).

15.1.5.5.1 Evaluation Model

For evaluating the radiological consequences due to a postulated steam system piping failure, the activity released from the affected steam generator (steam generator connected to the broken steam line) is released directly to the environment. The unaffected steam generators are assumed to continually discharge steam and entrained activity up to the time initiation of the residual heat removal (RHR) system can be accomplished.

All radioactivity is released to the environment with no consideration given to radioactive decay or cloud depletion by ground deposition during transport to the exclusion area boundary (EAB) and low-population zone (LPZ).

Mathematical models used in the analysis are described in the following sections:

- The off-site and on-site doses are calculated with the RADTRAD code.
- The atmospheric dispersion factors (χ/Q values) used in the analysis are described in Section 15.0.3.3.
- The total effective dose equivalent (TEDE) doses to a receptor at the EAB and outer boundary of the LPZ were analyzed using the models described in Section 15.0.3.1 and Appendix 15A.

Figure 15A-3 depicts the leakage sources to the environment modeled in the dose computation.

For evaluating the radiological consequences due to a postulated steam system piping failure, the activity released from the affected steam generator (steam generator connected to the broken steam line) is assumed to be released directly to the environment. The unaffected steam generators are assumed to continually discharge steam and entrained activity via the safety and relief valves up to the time initiation of the RHR system can be accomplished.

All activity is released to the environment with no consideration given to radioactive decay or cloud depletion by ground deposition during transport to the EAB and LPZ.

15.1.5.5.2 Input Parameters and Initial Conditions

The major assumptions and parameters used in the analysis are itemized in Tables 15.1.5-2 and 15.0-10 through 15.0-14.

The concentrations of nuclides in the primary and secondary system prior to the transient are determined as follows:

Reactor coolant activities are based on the Technical Specification limit of 1.0 µCi/g I-131 dose equivalent (DE) with extremely large iodine spike values.

The secondary coolant iodine and alkali concentration is 10% of the reactor coolant concentration.

The iodine concentrations in the reactor coolant are calculated using two separate assumptions that ensure the calculations account for conservatively large quantities of radioactive iodine: (1) assuming a pre-transient iodine spike and (2) assuming transient initiated iodine spikes. The use of these two separate cases is consistent with the guidance in Reference 15.1-3.

(1) Pre-transient Iodine Spike

A reactor transient has occurred prior to the steam line piping failure transient and has raised the reactor coolant iodine concentration to 60 µCi/g DE I-131.

(2) Transient-Initiated Iodine Spike

The primary system transient associated with the steam line piping failure transient causes an iodine spike in the primary system. The increase in reactor coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the reactor coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (1 μCi/g DE I-131) specified in the Technical Specifications (i.e., concurrent iodine spike case). The assume iodine spike duration is 8 hours.

The activity released from the fuel is assumed released instantaneously and homogeneously through the reactor coolant.

The pre-accident noble gas concentrations in the reactor coolant are based on the Technical Specification limit of 300 μCi/g DE Xe-133. Also, the pre-accident alkali metal concentrations in the reactor coolant are based on 1% fuel defect.

A 600 gallons per day (gpd) steam generator primary-to-secondary leakage is assumed, which is the Technical Specification limit. It is assumed that 150 gpd of this leakage goes to the steam generator in the faulted loop.

The chemical form of radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during iodine spiking.

The only filtration system considered in the analysis which limits the consequences of the steam system piping failure transient is the main control room (MCR) heating, ventilation and air conditioning (HVAC) system.

The χ/Q values and breathing rates are listed in Table 15.0-13. The breathing rates are obtained from RG 1.183 (Ref. 15.1-3).

Other assumptions relating to the transport, reduction, and release of radioactive material to the environment are those identified in Appendix E of RG 1.183 (Ref. 15.1-3).

15.1.5.5.3 Results

The calculated TEDE doses have been analyzed for the limiting 2-hour dose at the EAB and for the duration of the transient at the LPZ outer boundary. These doses are calculated for both the accident-initiated iodine spike and the pre-transient iodine spike cases. Table 15.1.5-3 lists the results.

As shown in Table 15.1.5-3, for the case in which the iodine spike is initiated by the accident, the TEDE doses for the limiting 2-hour case are calculated to be 0.32 rem at the EAB and 0.28 rem at the LPZ outer boundary. These doses are less than 10% of the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34.

As shown in Table 15.1.5-3, for the case in which the steam line piping failure occurs coincidently with a pre-transient iodine spike, the TEDE doses are calculated to be 0.19 rem at the EAB and 0.11 rem at the LPZ outer boundary. These doses are less than the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34.

The doses for the MCR for the steam line piping failure are bounded by the doses calculated for the LOCA event described in Section 15.6.5.5. Consequently, no doses are provided for the steam line piping failure event.

15.1.5.6 Conclusions

In all three core and system response analysis cases presented in this section, the minimum DNBR stays above the 95/95 limit, and no fuel failures are predicted as a result of this accident. Because DNB is not predicted, the issue of DNB propagation under low pressure conditions representative of this event does not need to be addressed.

The RCS pressure and steam pressure remain well below 110% of their system design pressures, so the integrity of the reactor coolant pressure boundary and the integrity of the main steam system pressure boundary are maintained. This meets the acceptance criteria that the pressures be maintained within acceptable design limits.

Steam system piping failures do not lead to a more serious fault condition.

The resultant doses are well within the guideline values of 10 CFR 50.34.

Table 15.1.5-1
Time Sequence of Events for the Steam System Piping Failure

Event Description	Case A Time [sec]	Case B Time [sec]
Steam pipe rupture occurs	0.0	0.0
Low steamline pressure analytical limit reached	1.5	1.5
RCP coastdown begins	N/A	4.5
MSIVs closed	10.0	10.0
Automatic isolation of EFW to faulted SG (Case B)	N/A	50.2
Safety injection pumps start	21.5	121.5
Boron reaches core	44.9	141.4
Automatic isolation of EFW to faulted SG (Case A)	51.7	N/A
Peak core heat flux occurs	89.8	152.8
Faulted SG water mass depleted	330	1420

Table 15.1.5-2
Parameters Used in Evaluating the Radiological Consequences of Steam System Piping Failure (Sheet 1 of 2)

Parameter	Value
Source Data	
Core thermal power level (MWt)	4540 (2% above the design core thermal power)
Transient initiated spike	Initial concentration equal to the 1.0 μCi/g DE I-131 with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 500. (See Table 15.0-11.) The duration is 8 hours.
Pre-transient spike	An assumed pre-transient iodine spike which has resulted in 60 μCi/g DE I-131 in the reactor coolant. (See Table 15.0-10.)
Reactor coolant noble gas and other radionuclides (both cases)	The noble gas concentrations in the reactor coolant are based on the technical specification limit of 300 μCi/g DE Xe-133. (See Table 15.0-12.) The alkali metal concentrations in the reactor coolant are based on 1% fuel defect. (See Table 11.1-2.)
Secondary system initial iodine and alkali concentration	10% of reactor coolant concentrations.
Reactor coolant mass (lb)	646,000
Secondary coolant mass, 4 steam generators (lb)	968,000
Offsite power	Lost after trip
Total steam generator tube leakage (gpd)	600
Primary-to-secondary leakage duration (h)	14
Iodine chemical form	elemental:97%, organic:3%

Table 15.1.5-2
Parameters Used in Evaluating the Radiological Consequences of Steam System Piping Failure (Sheet 2 of 2)

Parameter	Value
Activity Release Data for the Steam Generator in the Faulted Loop	
Primary-to-secondary leak rate (gpd)	150
Flow flashing fraction (lb/min)	
0 to 14 h	0.874
Steam release (lb)	
0 to 0.00112 h	120,000
0.00112 to 0.00278 h	158,000
0.00278 to 0.0612 h	2,190,000
0.0612 to 0.412 h	657,000
0.412 to 14 h	0
Iodine Partition Coefficient	1
Activity Release Data for the Steam Generator in the Intact Loops	
Primary-to-secondary leak rate (gpd)	450
Steam released (lb)	
0 to 8 h	1,540,000
8 to 14 h	1,540,000
Iodine partition coefficient	100
Particulate partition coefficient for moisture carryover in the steam generators	1000
Radiological Dose Parameters	
χ/Q	See Table 15.0-13.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.1.5-3
Radiological Consequences of Steam System Piping Failure

Dose Location	TEDE Dose (rem)
Transient-initiated iodine spike	
EAB (0 to 2 hours)	0.32
LPZ outer boundary	0.28
Pre-transient iodine spike	
EAB (0 to 2 hours)	0.19
LPZ outer boundary	0.11

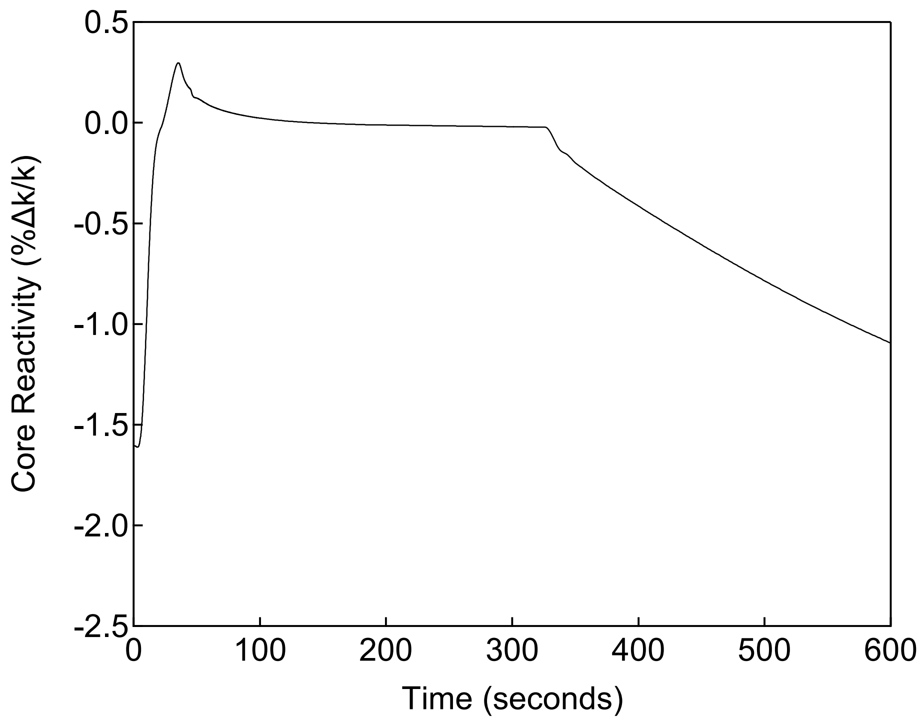


Figure 15.1.5-1 Core Reactivity versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

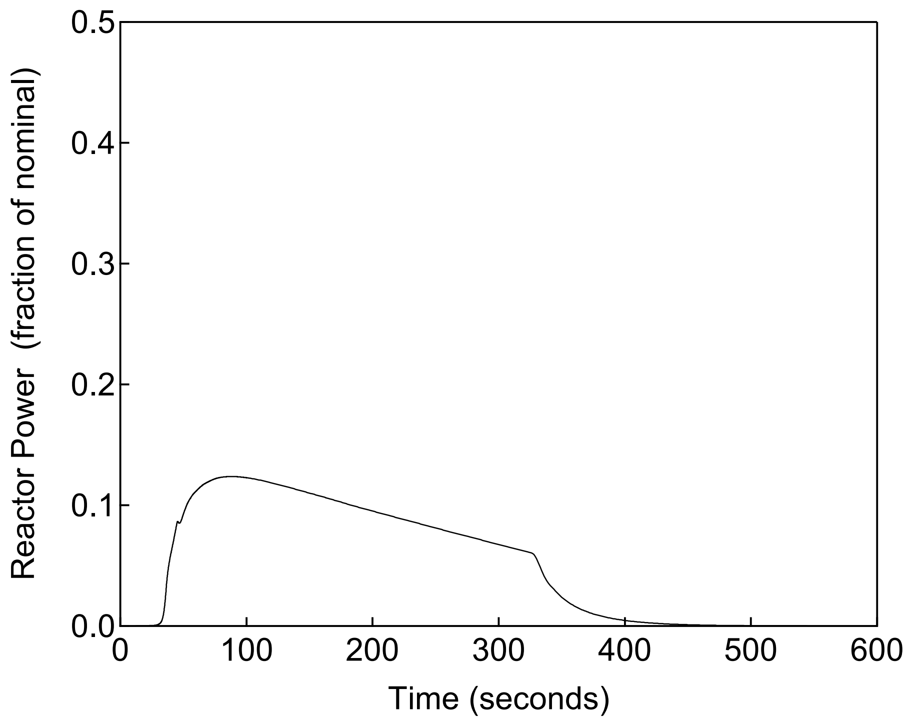


Figure 15.1.5-2 Reactor Power versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

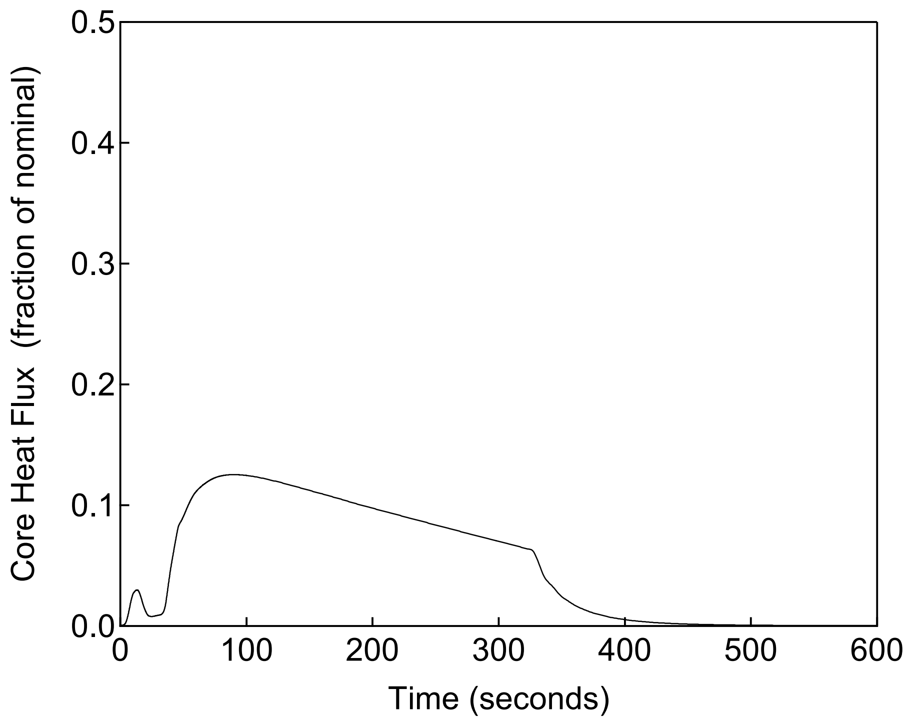


Figure 15.1.5-3 Core Heat Flux versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

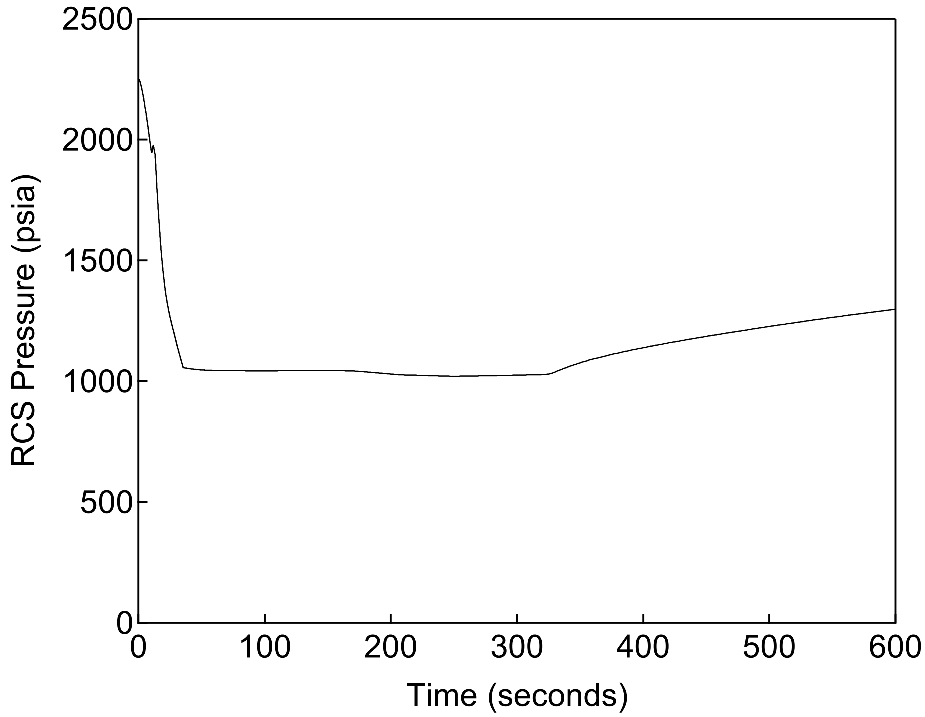


Figure 15.1.5-4 RCS Pressure versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

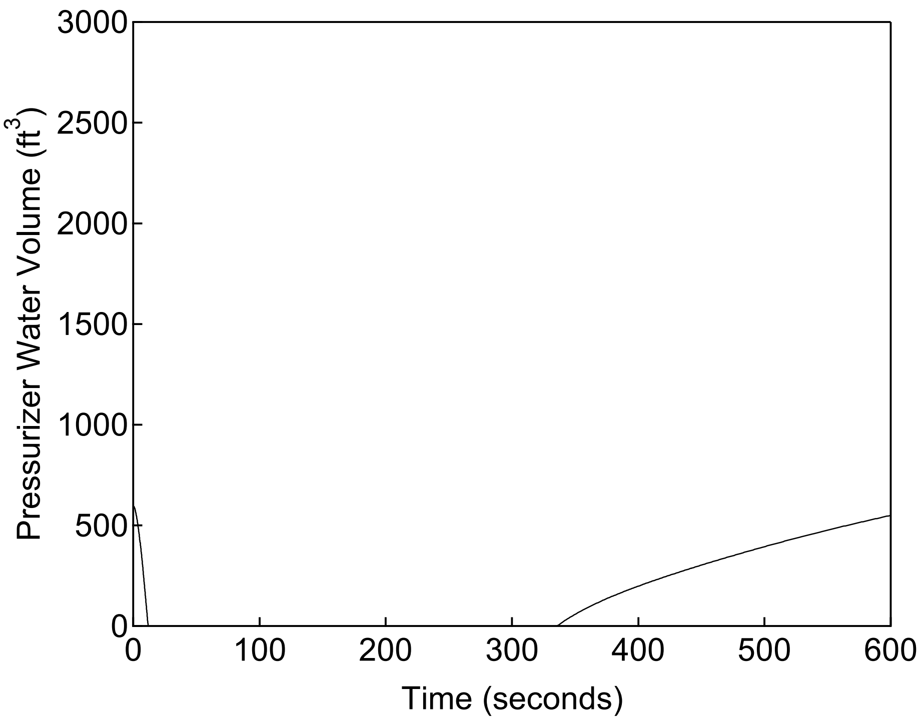


Figure 15.1.5-5 **Pressurizer Water Volume versus Time**

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

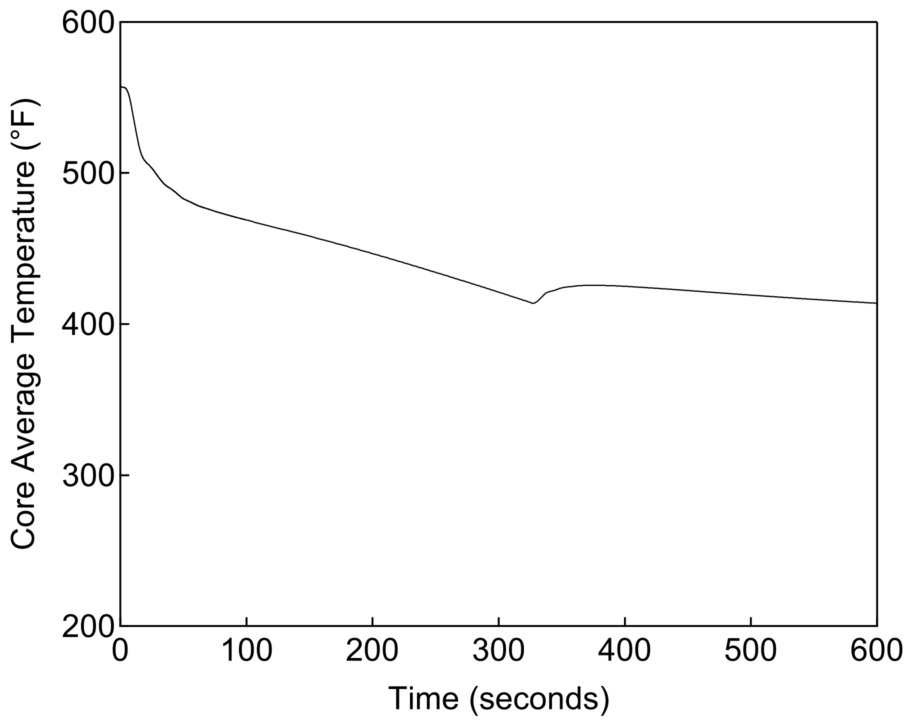


Figure 15.1.5-6 Core Average Temperature versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

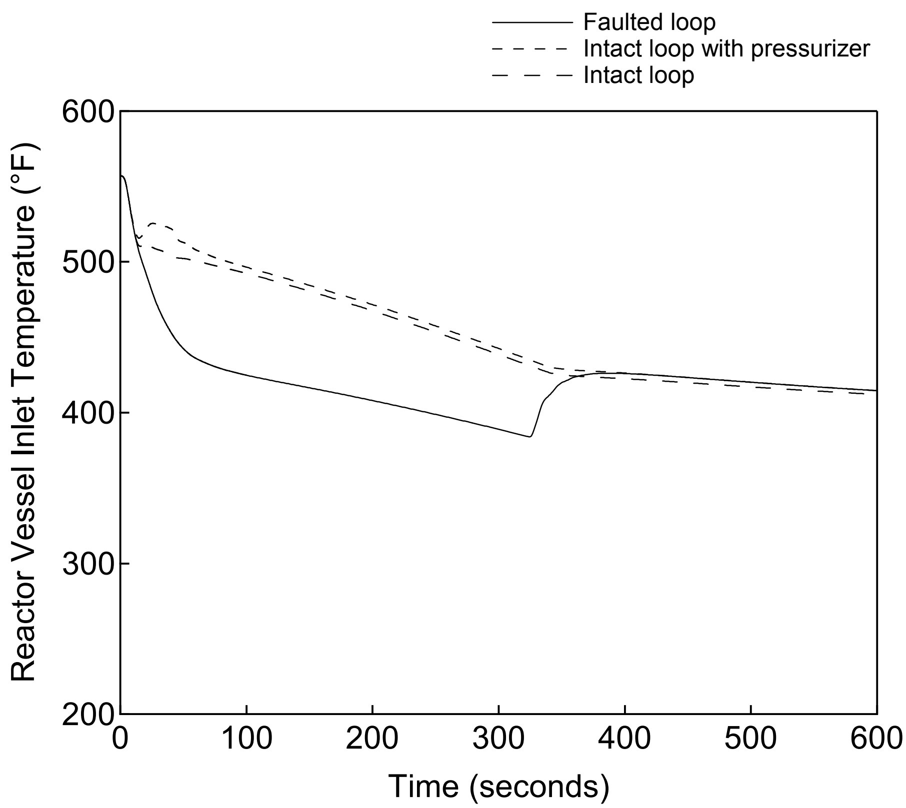


Figure 15.1.5-7 Reactor Vessel Inlet Temperature versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

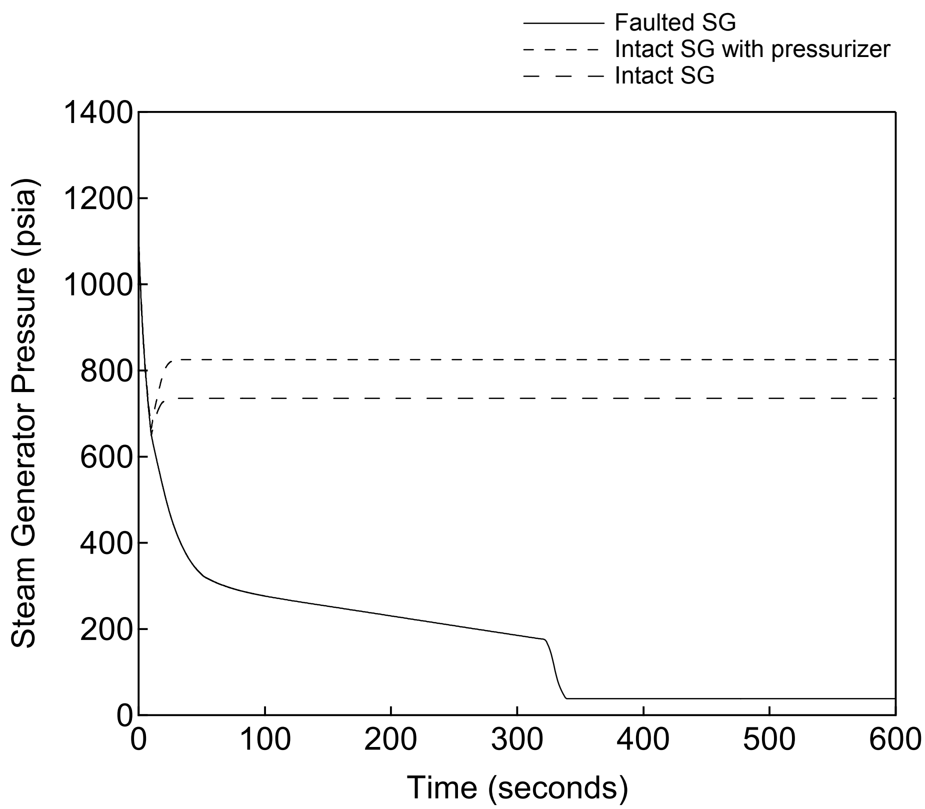


Figure 15.1.5-8 Steam Generator Pressure versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

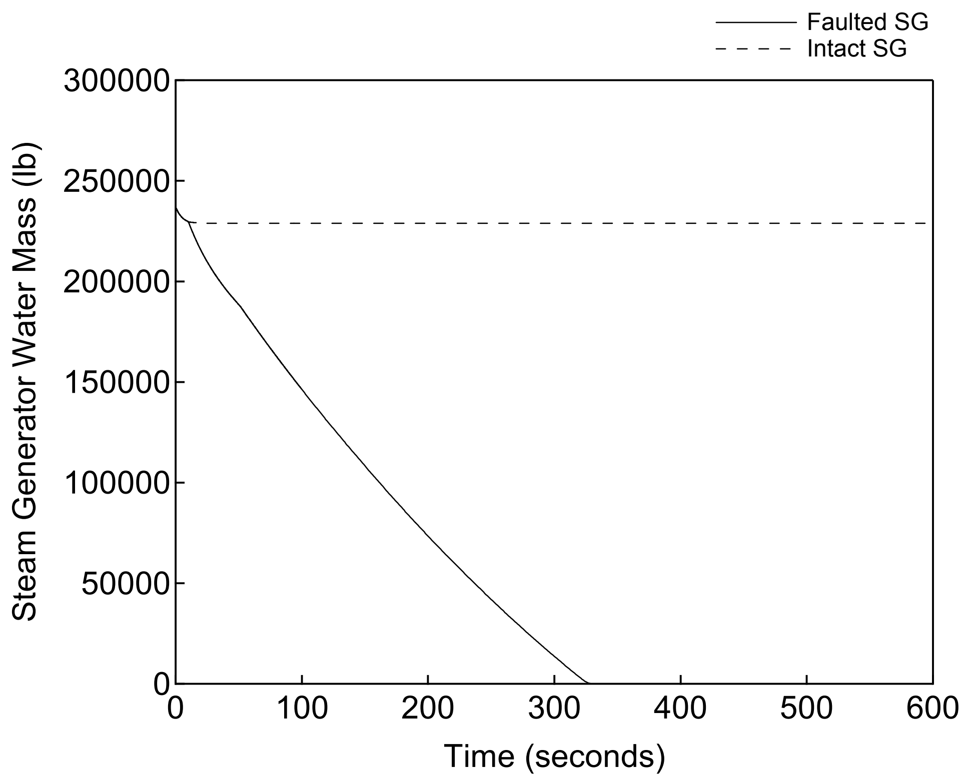


Figure 15.1.5-9 Steam Generator Water Mass versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

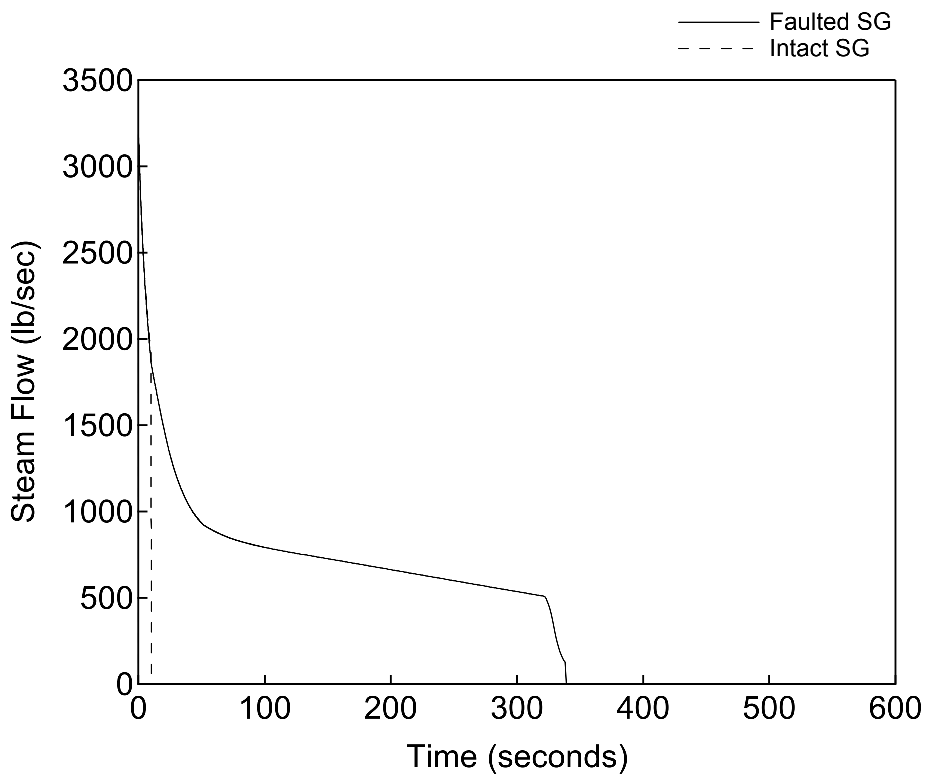


Figure 15.1.5-10 Steam Flow Rate versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

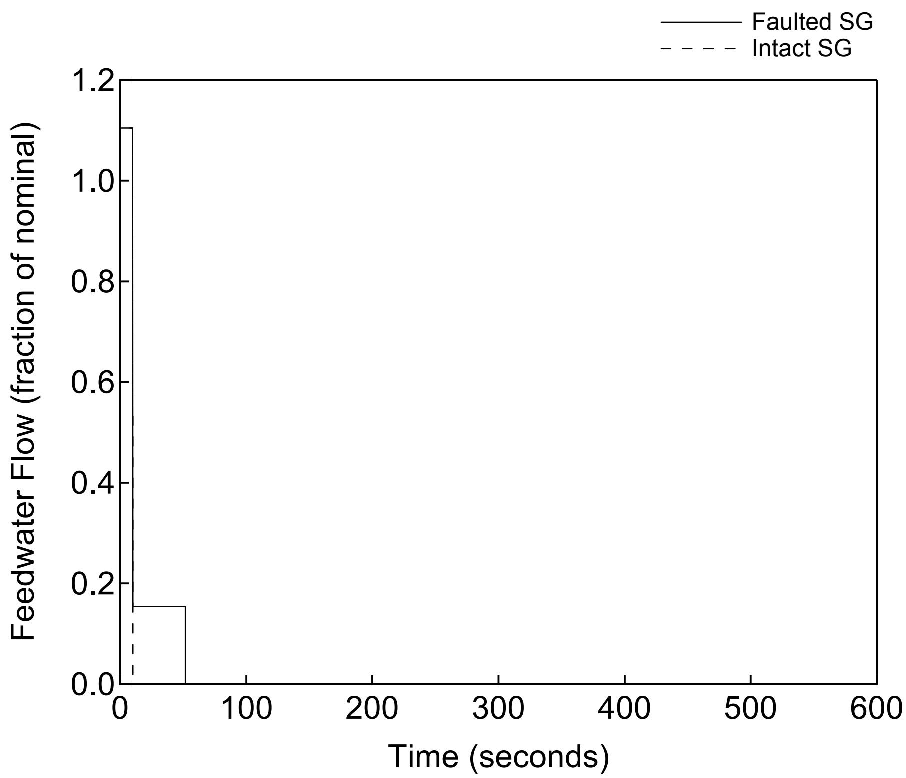


Figure 15.1.5-11 Feedwater Flow Rate versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

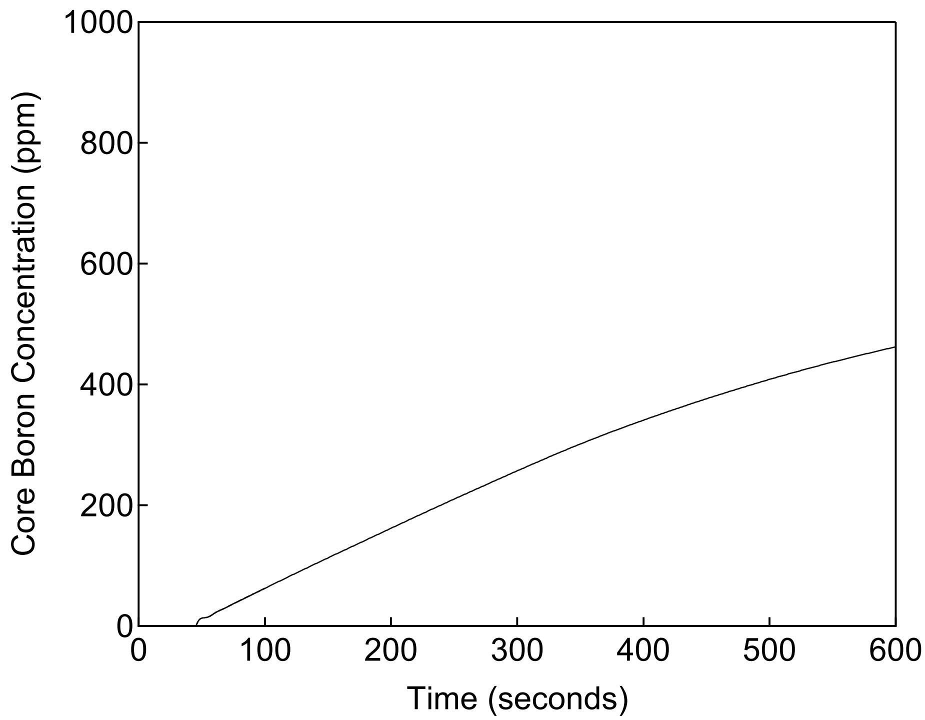


Figure 15.1.5-12 Core Boron Concentration versus Time

Steam System Piping Failure
- Case A: Double Ended Break from Hot Standby with Offsite Power

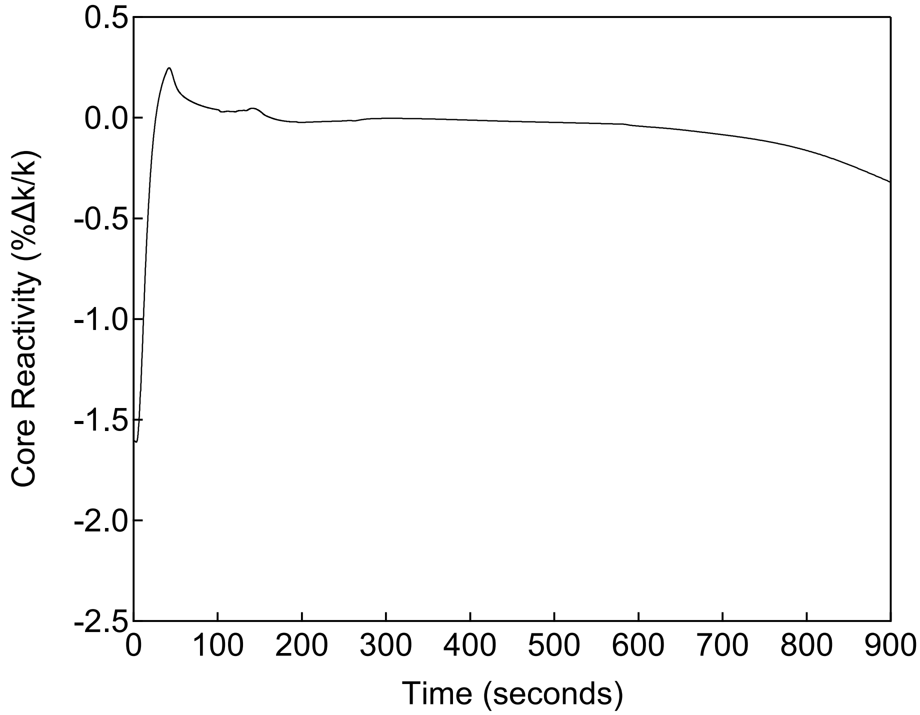


Figure 15.1.5-13 Core Reactivity versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

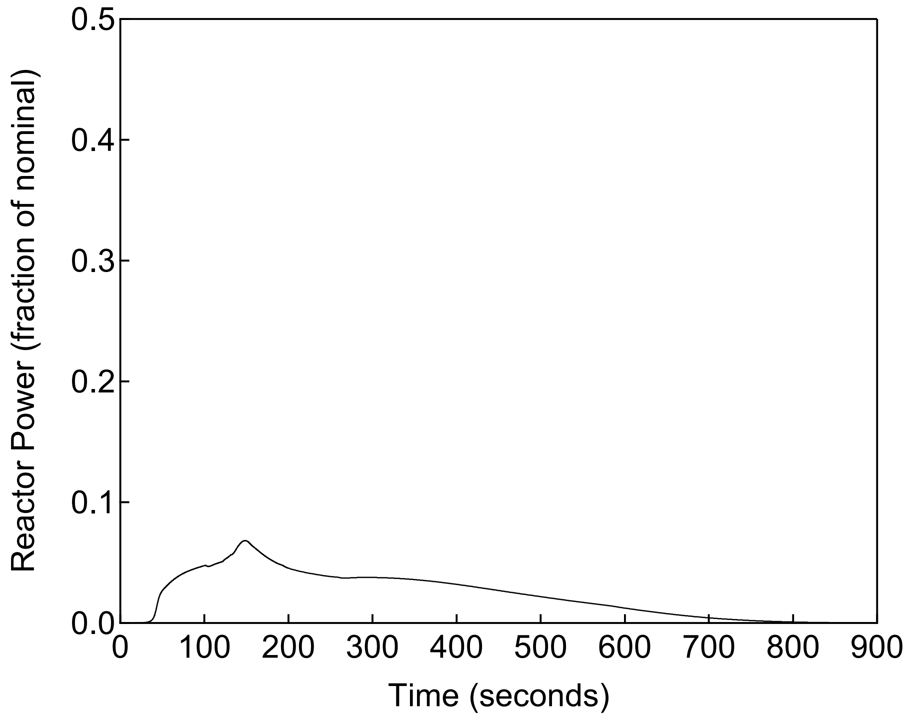


Figure 15.1.5-14 Reactor Power versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

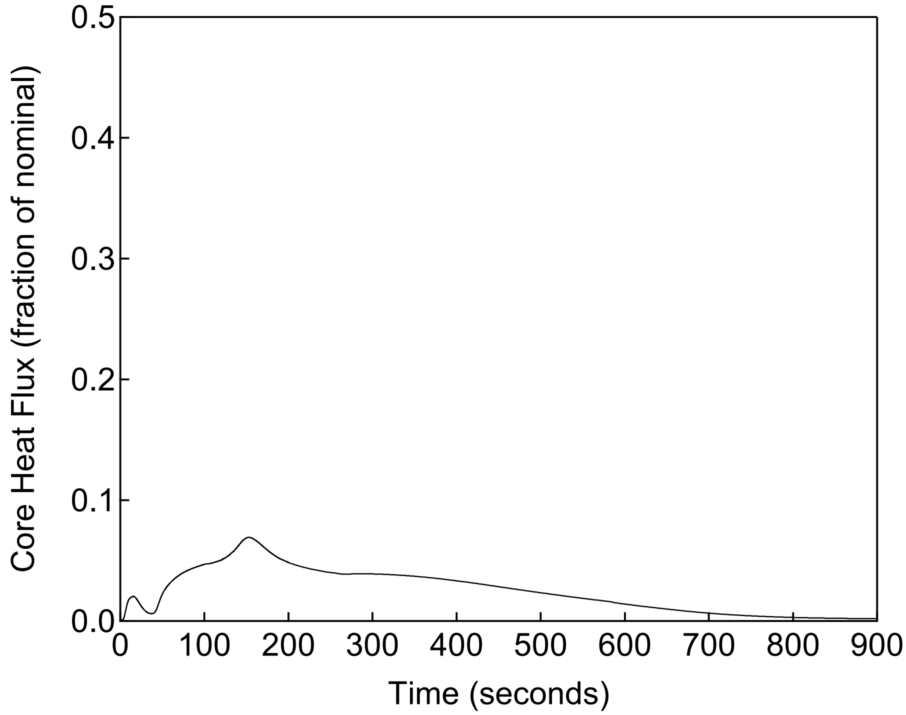


Figure 15.1.5-15 Core Heat Flux versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

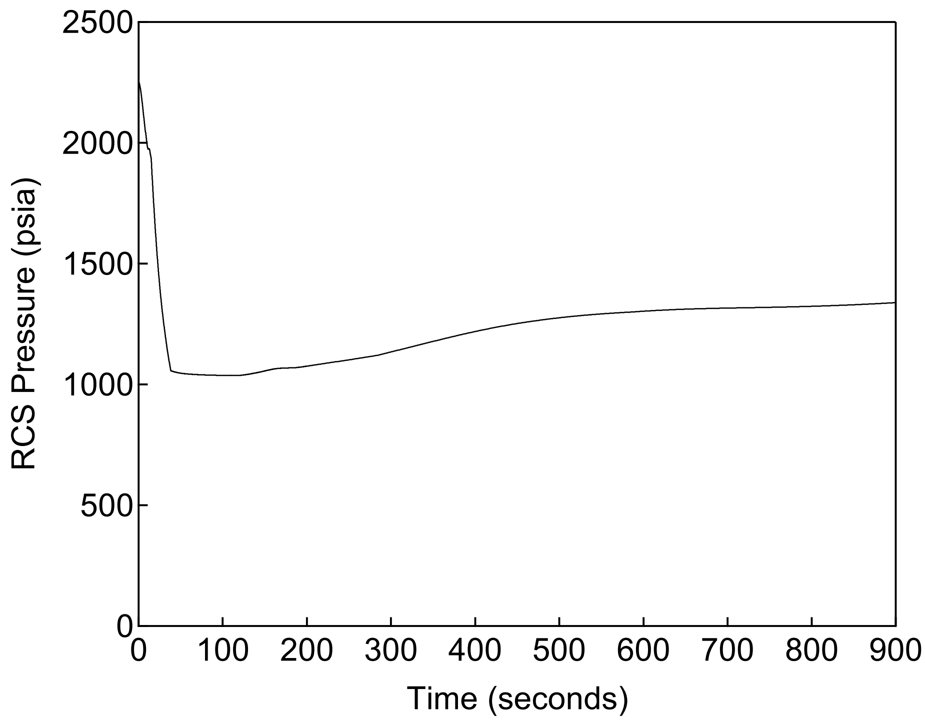


Figure 15.1.5-16 RCS Pressure versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

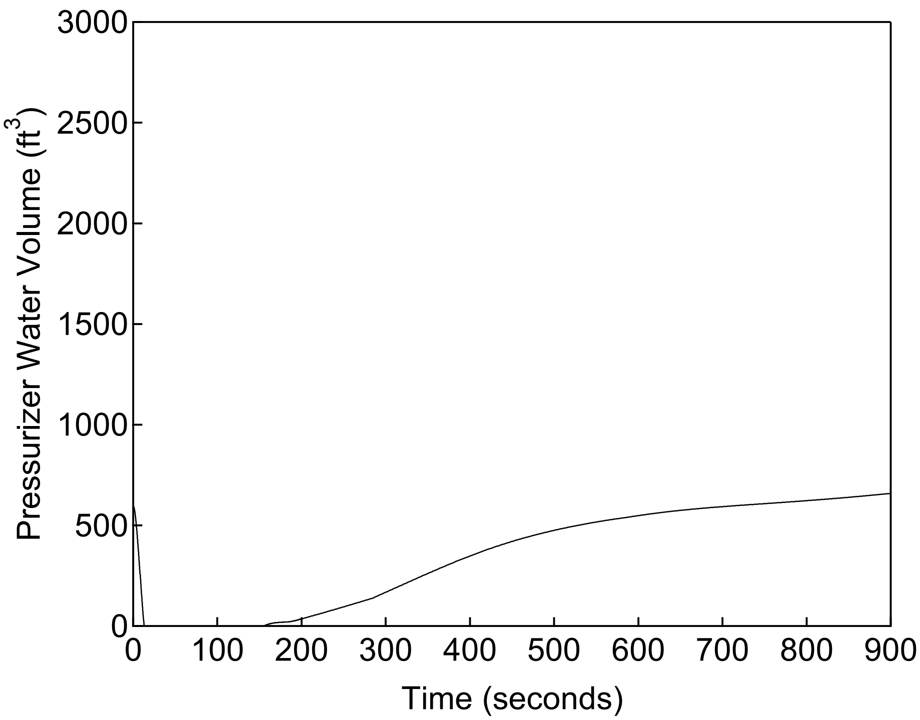


Figure 15.1.5-17 Pressurizer Water Volume versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

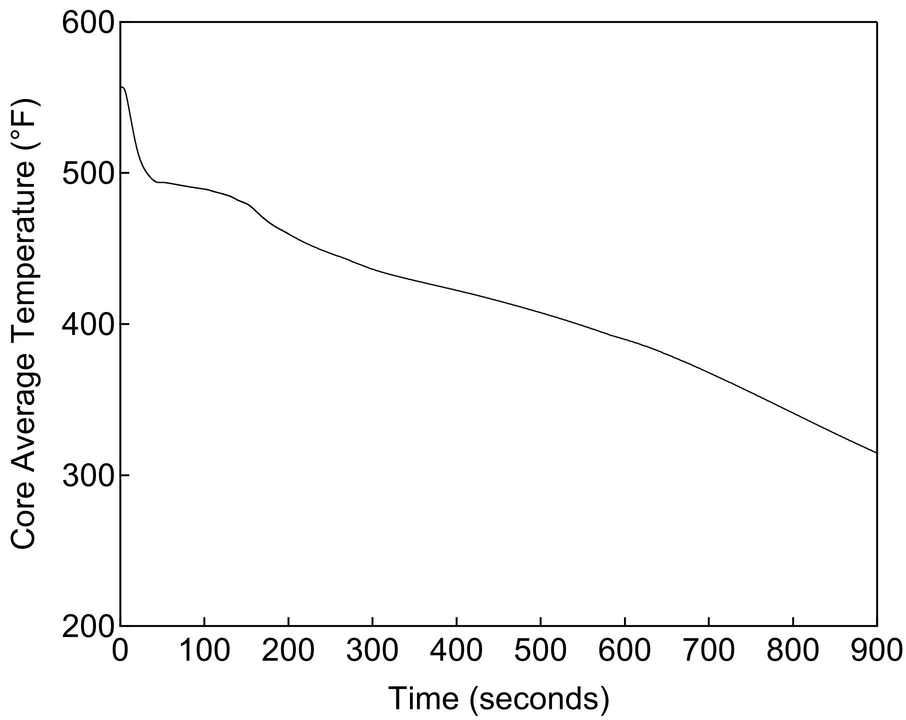


Figure 15.1.5-18 Core Average Temperature versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

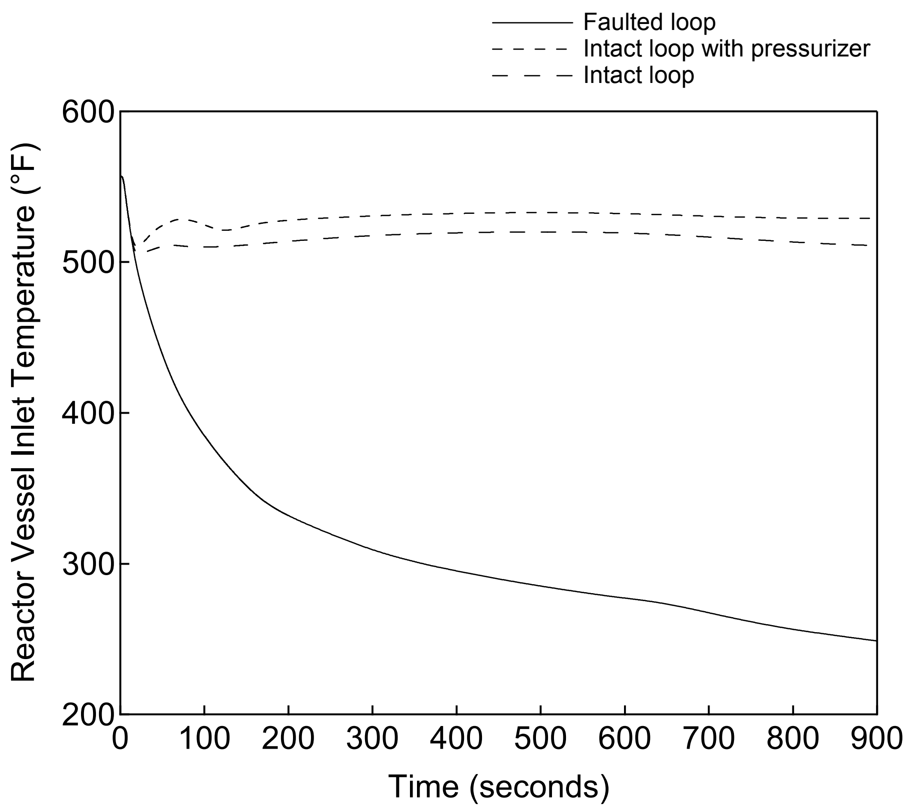


Figure 15.1.5-19 Reactor Vessel Inlet Temperature versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

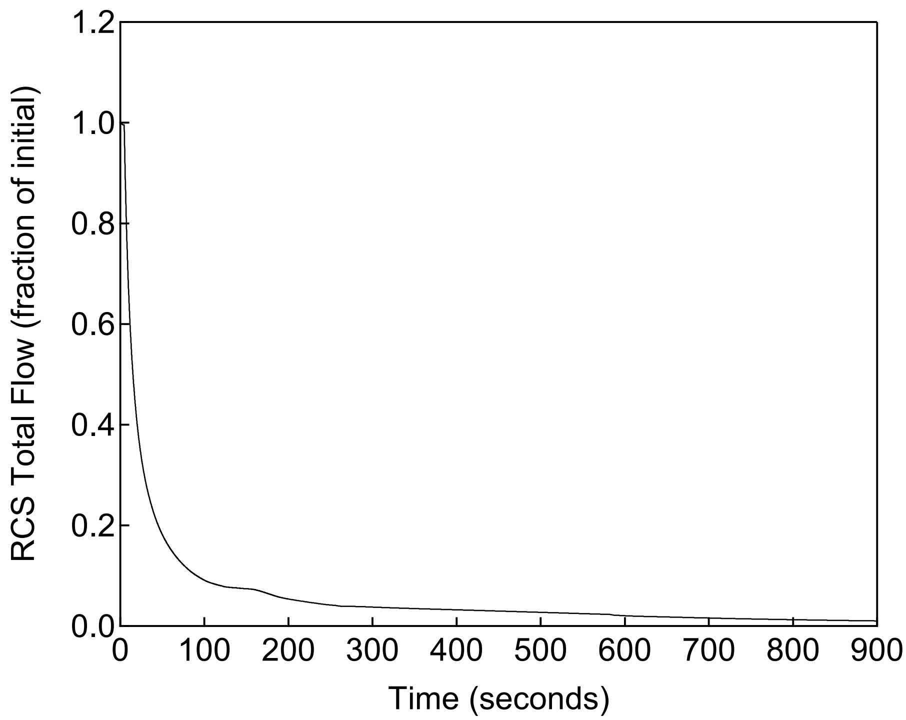


Figure 15.1.5-20 RCS Total Flow versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

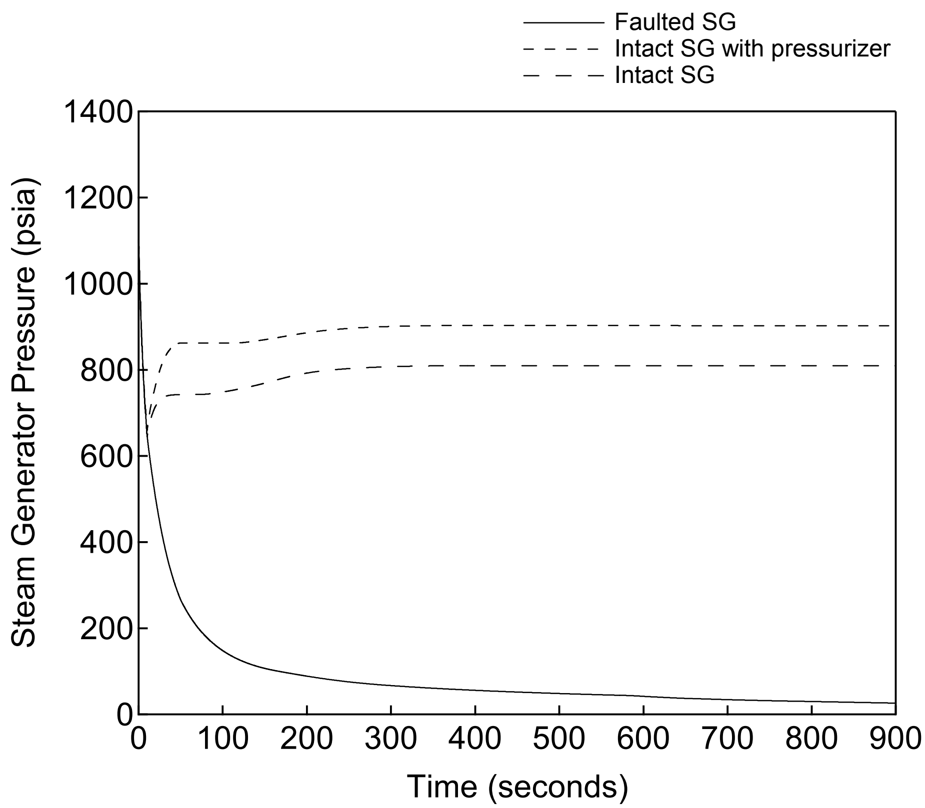


Figure 15.1.5-21 Steam Generator Pressure versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

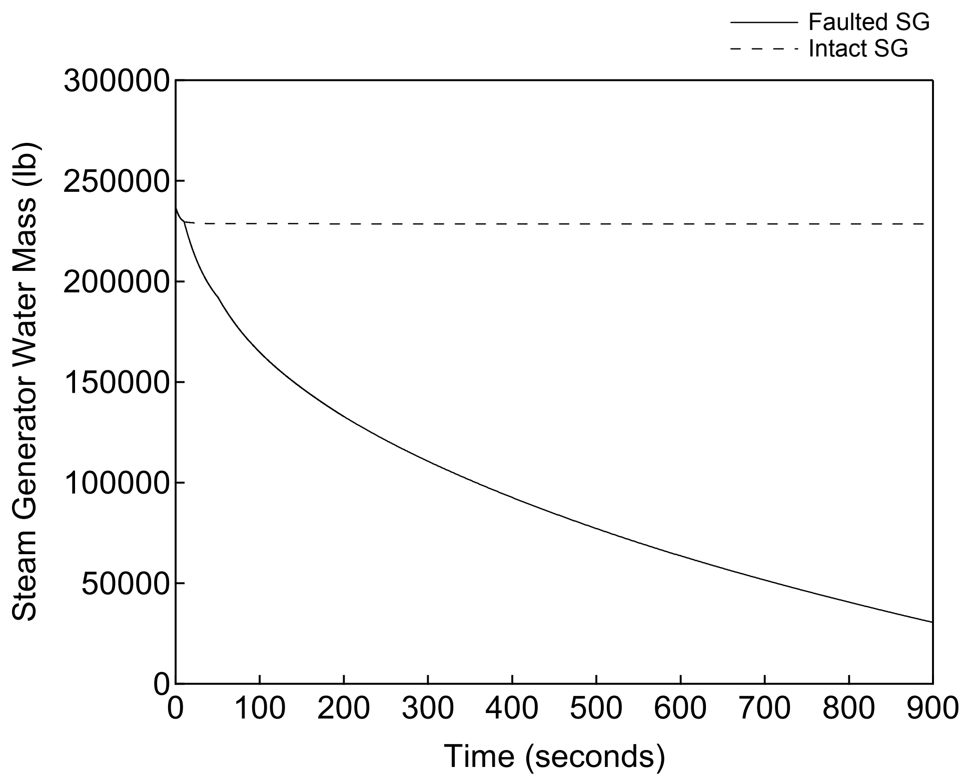


Figure 15.1.5-22 Steam Generator Water Mass versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

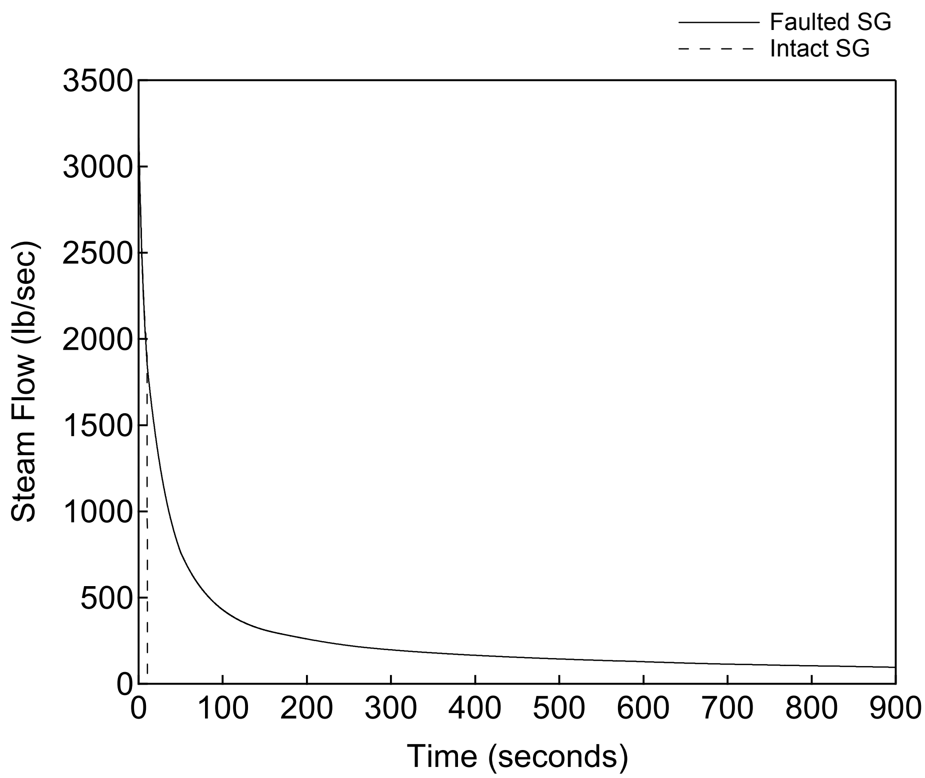


Figure 15.1.5-23 Steam Flow Rate versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

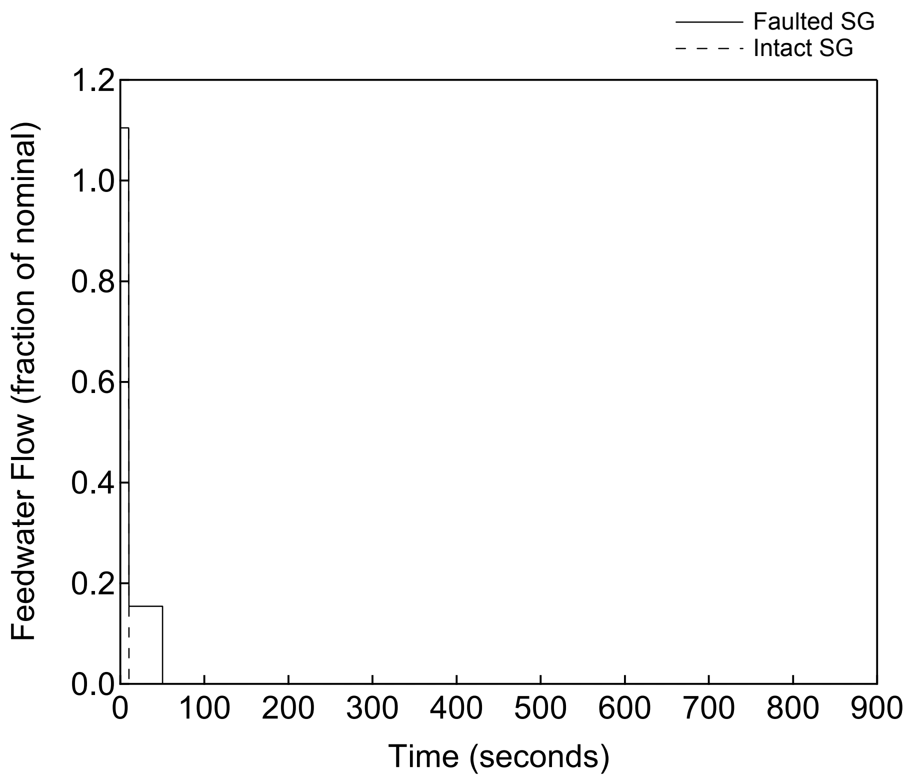


Figure 15.1.5-24 Feedwater Flow Rate versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

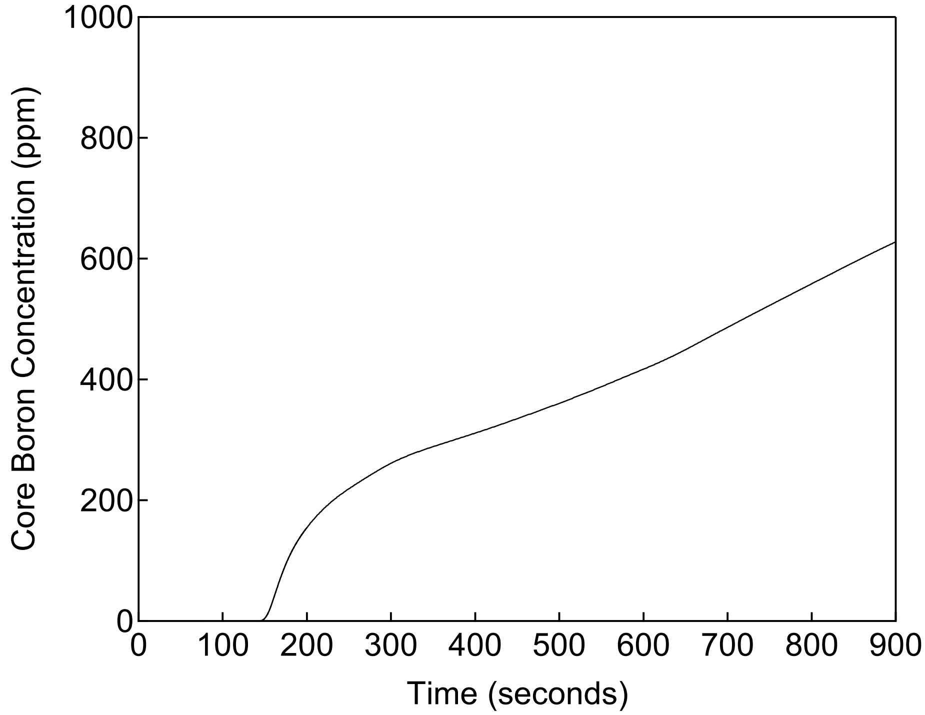


Figure 15.1.5-25 Core Boron Concentration versus Time

Steam System Piping Failure
- Case B: Double Ended Break from Hot Standby without Offsite Power

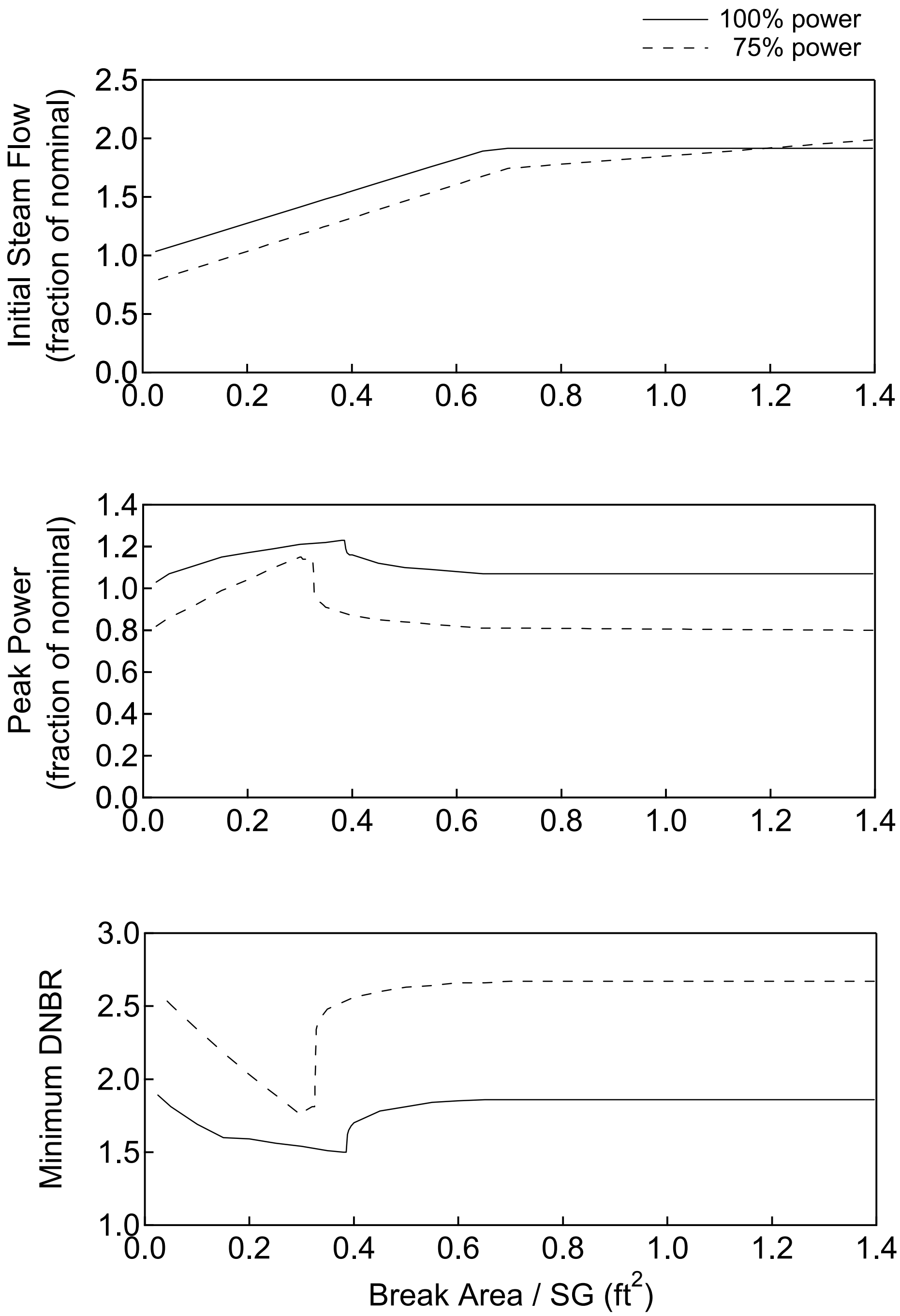


Figure 15.1.5-26 Initial Steam Flow, Peak Power, and Minimum DNBR versus Break Area

Steam System Piping Failure
- Case C: Spectrum of Breaks from Power Conditions with Offsite Power

15.1.6 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.1.7 References

15.1-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.1-2 Non-LOCA Methodology, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.1-3 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.1-4 Combined License Applications for Nuclear Power Plants (LWR Edition), NRC Regulatory Guide 1.206, June 2007.

15.1-5 Moody, F. J., Maximum Flow Rate of Single Component, Two-phase Mixture, Journal of Heat Transfer, Trans. of the ASME, No.1, Feb., 1965.

15.1-6 Thermal Design Methodology, MUAP-07009-P (Proprietary) and, MUAP-07009-NP (Non-Proprietary), May 2007.

15.2 Decrease in Heat Removal by the Secondary System

This section describes analyses that have been performed for events that could result in a decrease in heat removal by the secondary system. By decreasing the heat removal capability of the secondary system, the temperature in the primary reactor coolant system (RCS) is increased.

Analyses of the following events are described in this section:

- Loss of External Load
- Turbine Trip
- Loss of Condenser Vacuum
- Closure of Main Steam Isolation Valve (BWR)
- Steam Pressure Regulator Failure (not applicable to the US-APWR)
- Loss of Non-Emergency AC Power to the Station Auxiliaries
- Loss of Normal Feedwater Flow
- Feedwater System Pipe Break Inside and Outside Containment

These events are considered anticipated operational occurrences (AOOs) as defined in Section 15.0.0.1, except for the double-ended feedwater system pipe break which is classified as a postulated accident (PA).

For the US-APWR, the core analysis does not predict fuel failures from any of these events. The radiological consequences for the accidents in this section are bounded by the radiological consequences of the main steam line break accident (see Section 15.1.5).

15.2.1 Loss of External Load

The loss of external load is modeled by assuming an instantaneous step load decrease in both steam flow and feedwater flow from their full value (100%) to zero at the beginning of the transient. The direct reactor trip on turbine trip is not credited in the analysis. By using this bounding assumption, all credible loss of load scenarios are bounded, as well as other events in this event group, such as turbine trip.

15.2.1.1 Identification of Causes and Frequency Classification

In a loss of external load event, an electrical disturbance causes loss of a significant portion of the generator load. A loss of external load event may result from an abnormal grid frequency, a trip of the generator and or turbine trip, or spurious closure of the main turbine stop or control valves or main steamline isolation valves. The loss of load event is different from the loss of alternating current (ac) power event discussed in Section 15.2.6 in that offsite ac power remains available for this event to operate the station auxiliaries. Therefore, the onsite safety grade gas turbine generators (GTGs) are not required for this event. The US-APWR relies on GTGs instead of diesel generators for emergency power in loss of power events.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.2-1). Event frequency conditions are described in Section 15.0.0.1.

15.2.1.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the loss of external load event is described in the results section.

A loss of external load leads to fast closure of the main turbine control valves. The sudden reduction in steam flow leads to an increase in pressure and temperature in the shell side of the steam generators. As a result, the reactor coolant system (RCS) temperature and pressure increases, the coolant density decreases, and the pressurizer water volume increases. The automatic turbine bypass system would relieve the secondary side excess steam by dumping the steam to the main condenser. Excess pressure generated on the primary side would be relieved to the pressurizer relief tank by the pressurizer pressure safety valves. If the automatic turbine bypass system and the pressurizer pressure control system are functioning normally, the RCS temperature and pressure increase would be minor.

In the case where the main condenser is not available for steam dump by the automatic turbine bypass system, the steam is released to the atmosphere via the main steam relief valves. If the load loss is large enough, the main steam safety valves would be actuated. Also, main feedwater flow is lost if the main condenser is unavailable. In this case, feedwater flow to the steam generators is supplied by the emergency feedwater system (EFWS).

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could in turn, cause a loss of offsite power, which could, in turn cause a reactor coolant pump (RCP) coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

Depending on the magnitude of the loss of electrical load, the RTS, main steam safety valves, and pressurizer safety valves may be required to mitigate the transient. No single active failure will prevent this equipment from performing their required functions.

For this event, no normal reactor control systems or engineered safety systems are required to function and are thus not credited in the evaluation. However, the emergency feedwater (EFW) may be automatically actuated on loss of main feedwater; thus providing further mitigation of the transient.

The following automatic reactor trip signals are assumed to be available to provide protection from this transient:

- High pressurizer pressure
- High pressurizer water level
- Low steam generator water level
- Over temperature ΔT
- Turbine trip

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event (as discussed in Section 15.0.0.5). Long-term cooling using the residual heat removal (RHR) system following this event is discussed in Section 15.0.0.8.

For a complete loss of external load event, the maximum turbine overspeed would be approximately 10%, resulting in an overfrequency of approximately 6 Hz. This overfrequency is not expected to affect the voltage and frequency sensors. The effect of overfrequency on a reactor coolant pump motor is to increase the pump flow rate, which increases the margin to the design limits. Overfrequency will have no effect on safety-related equipment such as safety-related pump motors and the RTS. Safety-related equipment loads are supplied by the 120-V ac instrumental and control power system buses, which are connected to inverters. The inverters are supplied from direct current (dc) buses energized by batteries or by regulated ac voltage from two offsite transmission systems.

The main steam safety valves and the pressurizer safety valves are designed for overpressure protection for all load losses without crediting the operation of the turbine bypass system, pressurizer spray, automatic rod cluster control assembly control, or direct reactor trip on turbine trip. The main steam safety valves are sized to remove steam flow at the nuclear steam supply system thermal rating without exceeding 110 percent of the steam supply design pressure. With the main steam safety valves operating and the plant operating at the maximum turbine load, the pressurizer safety valves have sufficient capacity to accommodate a complete loss of heat sink and maintain the RCS pressure below 110 percent of the RCS design pressure.

15.2.1.3 Core and System Performance

15.2.1.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following a loss of external load and subsequent turbine trip. This evaluation model is described in Section 15.0.2.2.1. Additional details regarding the MARVEL-M code are provided in Reference 15.2-4.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

15.2.1.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for the loss of external load event:

- Consistent with the use of RTDP, the assumed initial values of reactor power, reactor coolant average temperature, and RCS pressure are assumed to be the nominal values as defined in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to be the minimum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. rod cluster control assembly insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- The reactor is assumed to be automatically tripped by the high pressurizer pressure signal. Table 15.0-4 summarizes the reactor trip setpoint and signal delay time used in the analysis.

- In this analysis, a bounding scenario that assumes an instantaneous step load decrease in both steam flow and feedwater flow from their full value (100%) to 0 initiates the event. The automatic reactor trip following turbine trip is conservatively ignored, delaying the reactor trip until the plant trips on other RTS signals. Since the transient is modeled in this manner, the results bound both the loss of load and the turbine trip (Section 15.2.2) events.

- In the automatic rod control mode, the control rod banks would be inserted to decrease power before the reactor trip occurs. Therefore, it is conservatively assumed that the reactor is in manual rod control.

- Additionally, the event is analyzed with the pressurizer spray actuating at 2275 psia and the safety valves credited to operate at 2525 psia. The availability of this equipment minimizes the RCS pressure, which is conservative in calculating the minimum DNBR. A separate case to evaluate RCS and main steam system peak pressures is described in Section 15.2.1.4, Barrier Performance.

15.2.1.3.3 Results

Table 15.2.1-1 lists the key events and times at which they occur, relative to the initiation of the transient.

Figure 15.2.1-1 demonstrates that the minimum DNBR remains above the 95/95 limit and no fuel failures are predicted.

It should be noted that the base analysis for the loss of external load and turbine trip is to evaluate peak RCS and main steam system pressures, and is presented as part of the barrier performance analysis in Section 15.2.1.4. Therefore, only the DNBR versus time parameter plot is provided for the core response analysis. The responses of the other parameters shown in Figures 15.2.1-2 through 15.2.1-7 are approximately the same for this case.

15.2.1.4 Barrier Performance

15.2.1.4.1 Evaluation Model

The barrier performance evaluation for this transient employs the same basic model as is used for the core and system performance evaluation (described in Section 15.2.1.3.1), except that certain input parameters and initial conditions are different so the calculations will produce the maximum RCS and maximum main steam system pressures instead of minimum DNBR.

15.2.1.4.2 Input Parameters and Initial Conditions

The same input parameters are used as in Section 15.2.1.3.2 with the exception of the initial conditions and pressurizer spray, which are discussed below.

- The initial power level is taken as 102 percent of the licensed core thermal power level with initial reactor coolant temperature 4°F below the nominal value and the pressurizer pressure 30 psi below the nominal value. This combination of initial condition uncertainties maximizes RCS pressure. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

- For the peak RCS pressure evaluation, the loss of external load/turbine trip event is analyzed with the pressurizer spray assumed to be unavailable. The unavailability of this equipment maximizes the RCS pressure for the barrier performance evaluation. However, the safety valves are assumed to be operable. The pressurizer safety valves begin to open at 2525 psia, corresponding to 101% of RCS design pressure.

15.2.1.4.3 Results

Table 15.2.1-2 lists the key events and times at which they occur, relative to the initiation of the transient.

Figures 15.2.1-2 through 15.2.1-7 are plots of system parameters versus time for the Barrier Performance Evaluation case.

The loss of external load/turbine trip event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The RCP outlet pressure (Figure 15.2.1-3) is the highest pressure in the RCS and is presented in place of RCS pressure for the purpose of confirming the reactor coolant pressure boundary limits are not exceeded. The maximum RCS pressure remains well below 110% of the design pressure. In addition, the steam generator pressure (Figure 15.2.1-7) does not exceed 110% of the main steam system design pressure. Therefore, the integrity of the reactor coolant pressure boundary and main steam system pressure boundary are maintained.

The Figure 15.2.1-1 parameter plot for DNBR for the loss of external load/turbine trip event is presented as part of the core response analysis in Section 15.2.1.3. The response of reactor power, average core heat flux, and maximum core heat flux are almost indistinguishable for this transient. Therefore, only a plot of reactor power is provided. Because there is subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided. These parameters are bounded by the more severe feedwater system pipe break event evaluated in Section 15.2.8, which provides plots for hot leg, cold leg, and saturation temperatures. An RCS average temperature plot is provided to characterize the temperature response for this event. Steam generator pressure (Figure 15.2.1-7) is provided in place of steam line pressure for the purpose of demonstrating that the main steam system pressure meets the acceptance limit. All pressurizer safety valve flow is steam since the pressurizer does not fill. Containment parameters are not reported for this event because there are no releases directly from the RCS or steam generators inside containment.

15.2.1.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.2.1.6 Conclusions

The sudden reduction in steam flow due to a loss of external load/turbine trip leads to an increase in pressure and temperature in the secondary side of the steam generators. As a result, the RCS temperature and pressure increases. However, the resulting transient does not cause the minimum DNBR to decrease below the 95/95 limit, and no fuel failures are predicted.

The RCS pressure and the main steam system pressure remain well below 110% of their system design pressures. Thus, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained for the loss of external load/turbine trip event.

This event does not lead to a more serious fault condition.

Table 15.2.1-1
Time Sequence of Events for
Loss of External Load/Turbine Trip Transient - DNBR Analysis

Event	Time (sec)
Turbine trip, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.7
Reactor trip initiated (rod motion begins)	8.5
Pressurizer safety valves open	8.6
Minimum DNBR occurs	9.5
Main steam safety valves open	9.7
Peak RCP outlet pressure occurs	10.3
Peak main steam system pressure occurs	14.3

Table 15.2.1-2
Time Sequence of Events for Loss of External Load/Turbine Trip Transient
- RCS & Main Steam Pressure Analysis

Event	Time (sec)
Turbine trip, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.9
Pressurizer safety valves open	8.6
Reactor trip initiated (rod motion begins)	8.7
Peak RCP outlet pressure occurs	10.9
Main steam safety valves open	11.5
Peak main steam system pressure occurs	14.9

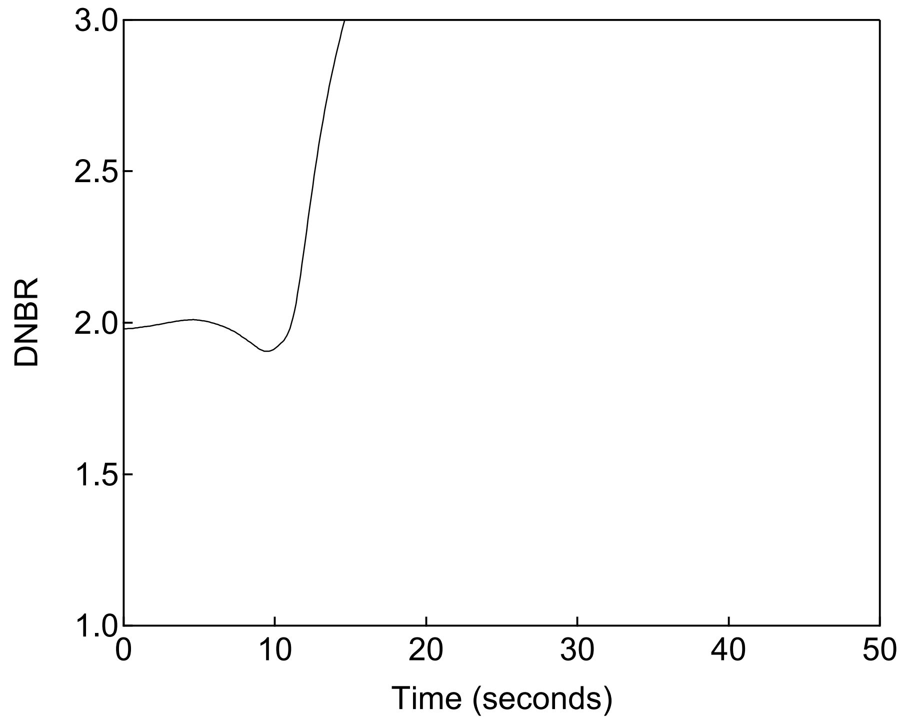


Figure 15.2.1-1 DNBR versus Time

Loss of External Load/Turbine Trip Transient - DNBR Analysis

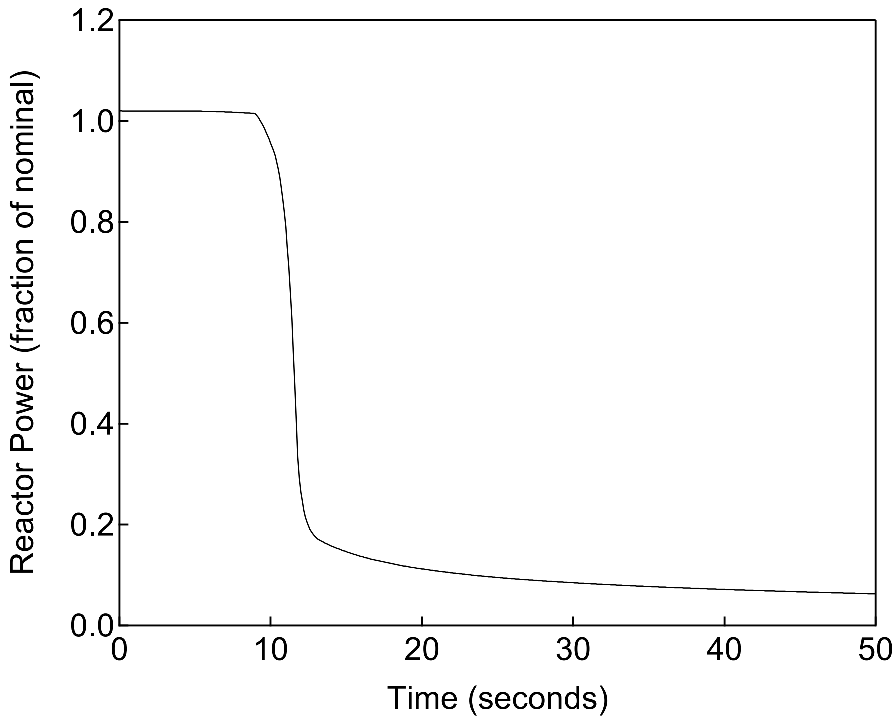


Figure 15.2.1-2 Reactor Power versus Time

Loss of External Load/Turbine Trip Transient
- RCS & Main Steam Pressure Analysis

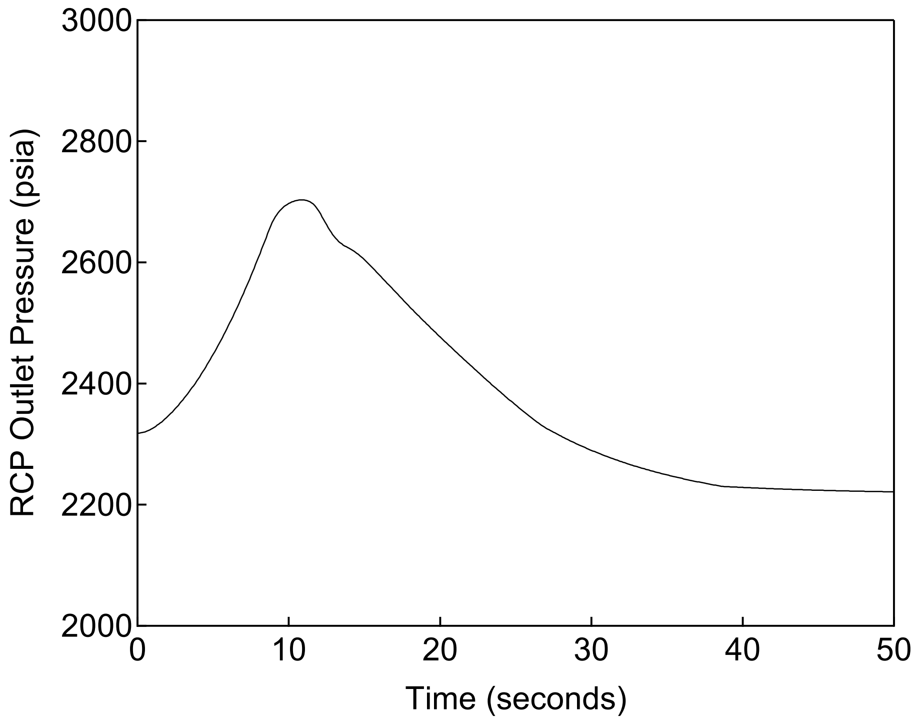


Figure 15.2.1-3 RCP Outlet Pressure versus Time

Loss of External Load/Turbine Trip Transient - RCS & Main Steam Pressure Analysis

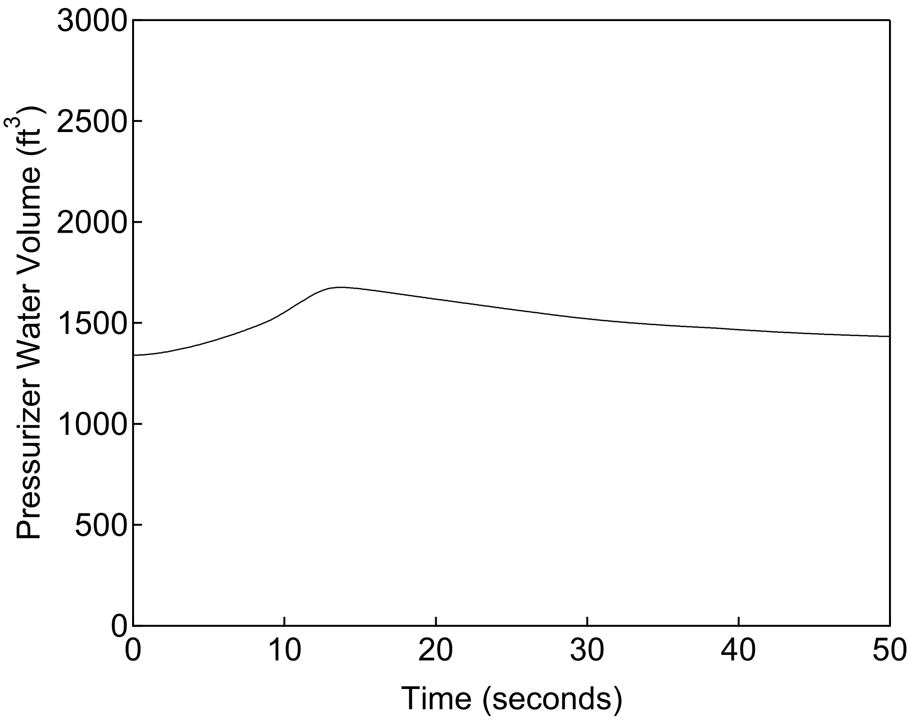


Figure 15.2.1-4 Pressurizer Water Volume versus Time

Loss of External Load/Turbine Trip Transient - RCS & Main Steam Pressure Analysis

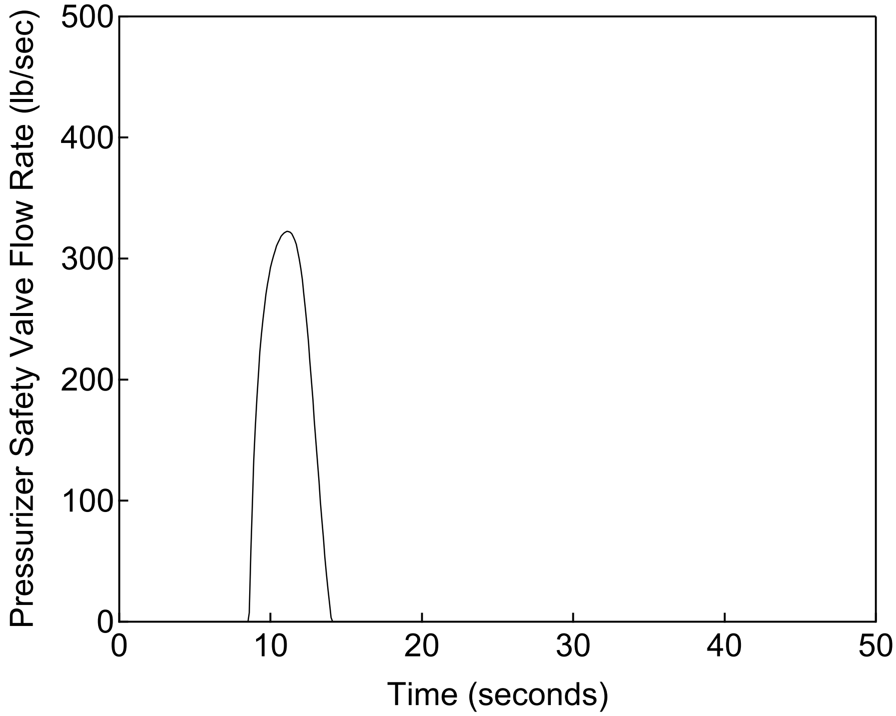


Figure 15.2.1-5 Pressurizer Safety Valve Flow Rate versus Time

Loss of External Load/Turbine Trip Transient
- RCS & Main Steam Pressure Analysis

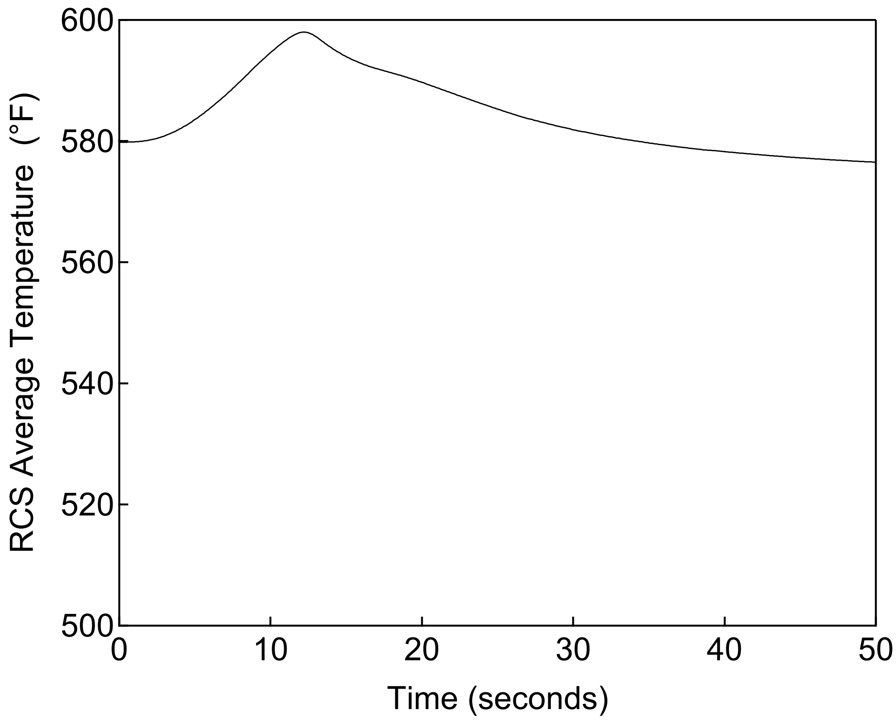


Figure 15.2.1-6 **RCS Average Temperature versus Time**

Loss of External Load/Turbine Trip Transient - RCS & Main Steam Pressure Analysis

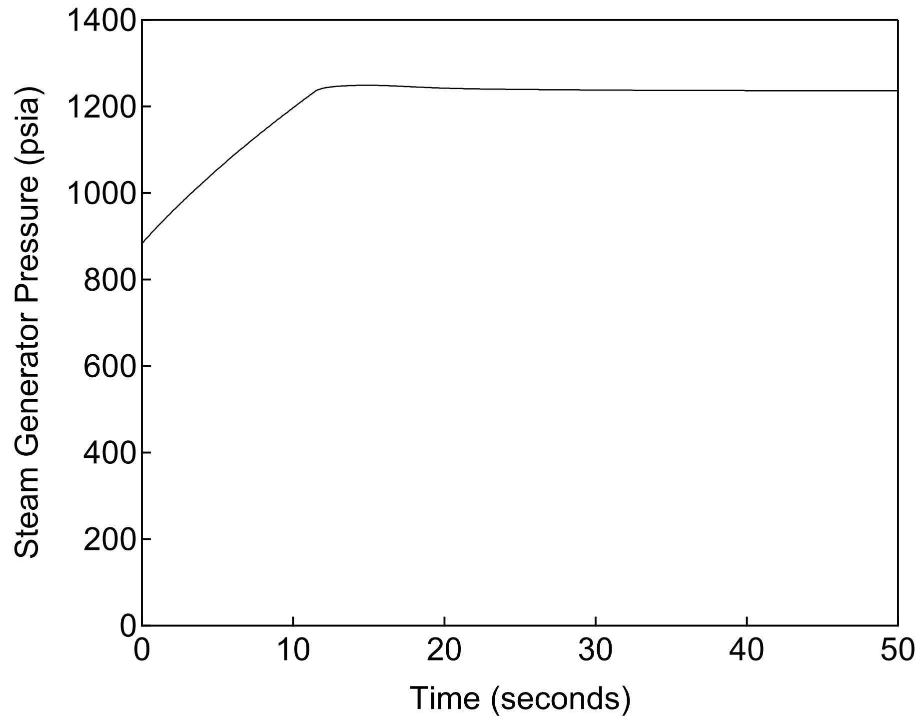


Figure 15.2.1-7 **Steam Generator Pressure versus Time**

Loss of External Load/Turbine Trip Transient
- RCS & Main Steam Pressure Analysis

15.2.2 Turbine Trip

15.2.2.1 Identification of Causes and Frequency Classification

In a turbine trip event, the main turbine stop valves rapidly close on loss of fluid pressure actuated by one the following turbine trip initiation signals:

- Low bearing oil pressure
- Low emergency trip header pressure
- Low condenser vacuum
- Turbine overspeed
- Thrust bearing wear
- High exhaust hood temperature
- High shaft vibration
- Low shaft-driven lube oil pump discharge pressure
- Generator trip
- Manual trip
- High-high steam generator water level
- Reactor trip

The turbine trip event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.2-1). Event frequency conditions are described in Section 15.0.0.1.

15.2.2.2 Sequence of Events and Systems Operation

The sequence of events for the turbine trip transient is similar to the loss of external load transient (Section 15.2.1) except that the steam flow following a turbine trip transient is isolated by closure of the main turbine stop valves rather than the main turbine control valves. The sudden reduction in steam flow leads to an increase in pressure and temperature in the shell side of the steam generators. As a result, the reactor coolant system (RCS) temperature and pressure increases, the coolant density decreases, and the pressurizer water volume increases.

However, the loss of external load scenario evaluated in Section 15.2.1 was developed by assuming a step load decrease in both steam flow and feedwater flow from their full valve (100%) to zero at the beginning of the transient. This scenario thus bounds both the loss of external load and turbine trip events. Therefore, a separate detailed transient analysis is not presented for the turbine trip event.

In addition to turbine trip signals generated by the turbine and condenser signals listed above, the automatic reactor trip signals assumed to be available to provide protection from the loss of external load transient are also applicable to the turbine trip event. See Section 15.2.1.2 for additional details.

A turbine trip could cause a disturbance to the utility grid, which could in turn, cause a loss of offsite power, which could, in turn cause a reactor coolant pump (RCP) coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event (as discussed in Section 15.0.0.5). Long-term cooling using the residual heat removal system following this event is discussed in Section 15.0.0.8.

15.2.2.3 Core and System Performance

The response of this event is bounded by the plant response for the loss of external load event described in Section 15.2.1.3.

15.2.2.4 Barrier Performance

The response of this event is bounded by the plant response for the loss of external load event described in Section 15.2.1.4.

15.2.2.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.2.2.6 Conclusions

The turbine trip event is bounded by the loss of external load event described in Section 15.2.1. Based on the results presented in Section 15.2.1.3.3, the resulting transient does not cause the minimum DNBR to decrease below the 95/95 limit.

The results of Section 15.2.1.4.3 demonstrate that RCS pressure and the main steam system pressure remain well below 110% of their system design pressures. Thus, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained for the turbine trip event.

In addition, this event does not lead to a more serious fault condition.

15.2.3 Loss of Condenser Vacuum

15.2.3.1 Identification of Causes and Frequency Classification

Loss of condenser vacuum is one of the initiators that lead to a turbine trip, which is discussed in Section 15.2.2 and bounded by the analysis in Section 15.2.1. In the loss of condenser vacuum transient, the main condenser is unavailable as a steam sink for the automatic turbine bypass system. The loss of condenser vacuum is modeled by assuming an instantaneous step load decrease and feedwater flow from full value (100%) to zero at the beginning of the transient. Since the loss of external load/turbine trip transient is analyzed without credit taken for the turbine bypass system and main feedwater flow is assumed to be lost initially, the results and conclusions of Section 15.2.1 are also applicable for the loss of condenser vacuum transient.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.2-1). Event frequency conditions are described in Section 15.0.0.1.

15.2.3.2 Sequence of Events and Systems Operation

Loss of condenser vacuum is one of the initiators that lead to a turbine trip, which is discussed in Section 15.2.2 and bounded by the analysis in Section 15.2.1. Therefore, the sequence of events described in Section 15.2.1.2 is also applicable to this event.

15.2.3.3 Core and System Performance

The response of this event is bounded by the plant response to the loss of load event described in Section 15.2.1.3.

15.2.3.4 Barrier Performance

The response of this event is bounded by the plant response to the loss of load event described in Section 15.2.1.4.

15.2.3.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.2.3.6 Conclusions

The loss of condenser vacuum leads to a turbine trip. The turbine trip event is bounded by the loss of external load event described in Section 15.2.1. Based on the results presented in Section 15.2.1.3.3, the resulting transient for a loss of condenser vacuum does not cause the minimum DNBR to decrease below the 95/95 limit.

The results of Section 15.2.1.4.3 demonstrate that RCS pressure and the main steam system pressure remain well below 110% of their system design pressures. Thus, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained for the loss of condenser vacuum and other events resulting in turbine trip.

This event does not lead to a more serious fault condition.

15.2.4 Closure of Main Steam Isolation Valve

15.2.4.1 Identification of Causes and Frequency Classification

Inadvertent closure of the main steam isolation valves would lead to a turbine trip. The automatic turbine bypass system would be unavailable for this event since the main steam isolation valves are upstream of the turbine bypass system. Loss of load/turbine trip without credit taken for the turbine bypass system is analyzed in Section 15.2.1. Therefore, the results and conclusions of Section 15.2.1 are also applicable for the inadvertent closure of main steam isolation valves transient.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.2-1). Event frequency conditions are described in Section 15.0.0.1.

15.2.4.2 Sequence of Events and Systems Operation

Inadvertent closure of the main steam isolation valves would lead to a turbine trip, which is discussed in Section 15.2.2 and bounded by the analysis in Section 15.2.1. Therefore, the sequence of events described in Section 15.2.1.2 is also applicable to this event.

15.2.4.3 Core and System Performance

The response of this event is bounded by the plant response for the loss of load event described in Section 15.2.1.3.

15.2.4.4 Barrier Performance

The response of this event is bounded by the plant response for the loss of load event described in Section 15.2.1.4.

15.2.4.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.2.4.6 Conclusions

The inadvertent closure of the main steam isolation valves leads to a turbine trip. The turbine trip event is bounded by the loss of external load event described in Section 15.2.1. Based on the results presented in Section 15.2.1.3.3, the resulting transient for the inadvertent closure of main steam isolation valves does not cause the minimum DNBR to decrease below the 95/95 limit.

The results of Section 15.2.1.4.3 demonstrate that reactor coolant system pressure and the main steam system pressure remain well below 110% of their system design pressures. Thus, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained for the inadvertent closure of the main steam isolation valves.

This event does not lead to a more serious fault condition.

15.2.5 Steam Pressure Regulator Failure

There are no steam pressure regulators in the US-APWR whose malfunction or failure could result in a steam flow transient.

15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

15.2.6.1 Identification of Causes and Frequency Classification

The loss of non-emergency alternating current (ac) power is assumed to result in the loss of all power to the station auxiliaries. The causes are a complete loss of the external (offsite) grid accompanied by a turbine-generator trip or loss of the onsite ac distribution system.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.2-1). Event frequency conditions are described in Section 15.0.0.1.

In addition to the SRP acceptance criteria for AOOs, MHI conservatively adopts an additional acceptance criterion to ensure the establishment of natural circulation flow.

15.2.6.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the loss non-emergency ac power event is described in the results section.

The loss of non-emergency ac power has the following effects: (1) all the reactor coolant pumps (RCPs) are tripped simultaneously, (2) an immediate load rejection with fast closure of the main turbine control valves, (3) loss of power to the condensate and main feedwater pumps, resulting in loss of feedwater, and (4) loss of condenser vacuum.

The RCPs will coast down following their trip, which reduces the flow through the core and increases the reactor coolant system (RCS) coolant temperature and pressure. Reactor trip will occur following loss of power to the control rod drive mechanisms, or from one of the following reactor trip signals assumed to be available to provide protection from this transient:

- Over temperature ΔT
- High pressurizer pressure
- High pressurizer water level
- Low reactor coolant flow
- Low reactor coolant pump speed
- Low steam generator water level
- Turbine trip

However, it is assumed in the analysis that reactor will trip on the low steam generator water level signal and that loss of ac power occurs when the rods begin to drop. This assumption leads to more conservative results than assuming that power is lost at time zero because the steam generator water mass is minimized when the reactor is tripped.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event (as discussed in Section 15.0.0.5). Long-term cooling using the residual heat removal (RHR) system following this event is discussed in Section 15.0.0.8.

The four gas turbine generators (GTGs) and the four batteries are Class 1E emergency power sources. The Instrumentation and Control power supply systems consist of four Class 1E 120V ac power systems and five non-Class 1E 120V ac power systems. The uninterruptible power supply (UPS) unit is the main power source for each system. Even if there is a fluctuation of in-feed power, the voltage and frequency of the inverter's output power is regulated. The UPS unit has one in-feed ac power source and one in-feed direct current (dc) power source. Even if ac power source is lost, the UPS unit supplies power to its loads. In this event, power to the vital instrument equipment is supplied by the dc power system.

The dc power system consists of four Class 1E 125V dc power systems and four non-Class 1E 125V dc power systems. Each dc power system consists of a battery charger, a battery, and dc power distribution equipment. A Class 1E battery supplies power to loads for two hours under a loss of ac power condition. A non-Class 1E battery supplies power to loads for one hour under a loss of ac power condition. Batteries are continuously charged when ac power is provided to the input of the battery charger from normal ac sources or from the Class 1E GTGs.

When a loss of offsite power (LOOP) occurs, a Class 1E GTG is automatically started and reaches its rated voltage within 100 seconds from receiving an undervoltage signal from the Class 1E medium voltage bus. The Class 1E GTGs supply power to the plant loads.

After reactor trip, decay and sensible heat from the fuel and coolant continue to generate steam. Since the main condenser is unavailable for steam dump by the turbine bypass system for this event, steam may be automatically relieved through the main steam relief valves. If the relief valves are unavailable, the main steam safety valves are used to relieve the steam. The relief valves or safety valves are used to dissipate residual decay heat as the no-load temperature is approached and to maintain the plant at hot standby conditions.

The following features are assumed to be available to mitigate this event.

- Emergency feedwater system (EFWS) automatic actuation

Without main feedwater flow to the steam generators, makeup flow to the steam generators is supplied by the emergency feedwater system (EFWS). The EFWS consists of two motor-driven pumps, two steam turbine-driven pumps, two emergency feedwater pits, and associated piping and valves. The four pumps take suction from the two emergency feedwater pits. Each of the four pumps is normally connected to only one steam generator with the cross-connect valves closed on the discharge paths in a “one-on-one” arrangement. The four-pump configuration and the cross-connected discharge header allow the system to remove the decay heat using three steam generators. Although the cross-connect valves could be opened by operator action, this is not credited in the accident analysis. Power is supplied to the motor-driven pumps by the safety GTGs while the turbine-driven pumps utilize steam from the secondary system to drive their turbines. The EFW pumps start on loss of offsite power, low steam generator water level, any emergency core cooling system signal, or manual actuation.

Following loss of power to the RCPs, core cooling is maintained by natural circulation of the reactor coolant aided by EFW on the secondary side.

The reactor trip system (RTS) and EFWS are required to function following this event. No single active failure prevents any of these systems from performing their functions. The worst single failure is loss of one EFWS train. The remaining three EFWS trains supply EFW to their respective steam generators.

15.2.6.3 Core and System Performance

The minimum DNBR for this event is not calculated because it is bounded by the minimum DNBR for the complete loss of flow event analyzed in Section 15.3.1.2. The reason why the complete loss of flow event is more conservative for calculating the minimum DNBR is that the loss of offsite power causes an immediate reactor trip from the control rods dropping from loss of power and RCP coastdown.

Confirmation that natural circulation flow is established and removes the long term decay heat is discussed in Section 15.2.6.4, Barrier Performance.

15.2.6.4 Barrier Performance

15.2.6.4.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following a loss of non-emergency ac power. This evaluation model is described in Section 15.0.2.2.1. Additional details on the MARVEL-M code are provided in Reference 15.2-4.

15.2.6.4.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative pressurizer water volume transient results for a loss of non-emergency ac power event.

- The initial power level is taken as 102 percent of the licensed core thermal power level with initial reactor coolant temperature 4°F below the nominal value and the pressurizer pressure 30 psi above the nominal value. This combination of initial condition uncertainties maximizes peak pressurizer water volume. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to have the maximum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- The core residual decay heat generation is conservatively based on the 1979 version of ANSI/ANS 5.1 (Ref. 15.2-3).

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. Rod cluster control assembly insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- The reactor is assumed to be automatically tripped by the low steam generator water level signal. Table 15.0-4 summarizes the reactor trip setpoint and signal delay times assumed in the analysis. Offsite power is assumed to be lost when the reactor trips.

- The elevation head associated with the core active fuel height is modeled to confirm natural circulation flow is established and is capable of removing decay and residual heat through the steam generators.

- The EFW is actuated on the low steam generator water level signal. It is assumed that one EFWS train fails leaving three pump trains to supply EFW to their respective steam generators.

- The event is analyzed with the main steam relief valves assumed to be unavailable. The unavailability of this equipment maximizes the main steam line pressure and peak pressurizer water level for the barrier performance evaluation cases. However, the main steam safety valves are assumed to be operable.

- The event is analyzed with the pressurizer safety valves, pressurizer spray, and heaters available until the LOOP. The LOOP occurs concurrently with the reactor trip, in order to maximize the pressurizer water volume to determine if the pressurizer will fill and relieve water through its safety valves.

15.2.6.4.3 Results

The sequence of events for the loss of non-emergency ac power to the station auxiliaries is listed in Table 15.2.6-1.

Figures 15.2.6-1 to 15.2.6-10 are plots of system parameters versus time for the Barrier Performance Evaluation case that demonstrates that natural circulation flow is established and is adequate to remove long-term decay heat following the event.

The response of reactor power, average core heat flux, and maximum core heat flux are almost indistinguishable for this transient. Therefore, only a plot of reactor power is provided. Plots for average and hot channel exit temperatures and steam fractions are not provided because this event is bounded by the more severe feedwater system pipe break event evaluated in Section 15.2.8, which also demonstrates margin to subcooling. An RCS average temperature plot is provided instead of a core average or inlet coolant temperature to characterize the temperature response for this event. Containment parameters are not reported for this event because there are no releases directly from the RCS or steam generators inside containment.

The loss of non-emergency ac power event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The results of the pressurizer water volume case demonstrate that the RCS pressure and main steam system pressure remain well below 110% of their respective system design pressures. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

Figure 15.2.6-4 shows that the maximum pressurizer water volume remains below the pressurizer capacity throughout the transient. Therefore, all pressurizer safety valve flow is steam since the pressurizer does not fill and water relief through the pressurizer safety valves is precluded.

15.2.6.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.2.6.6 Conclusions

Based on the bounding analysis in Section 15.3.1.2, the minimum DNBR remains above the 95/95 limit so that fuel integrity is not degraded for the loss of non-emergency ac power to the station auxiliaries transient.

The RCS pressure and the main steam system pressure remain below the 110% of their respective system design pressures so that the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained for this event.

The pressurizer does not fill with water for this event. Therefore, water relief through pressurizer safety valves is precluded. Natural circulation and EFW are sufficient to provide decay heat removal from the steam generators following reactor trip and RCP coastdown.

This event does not lead to a more serious fault condition.

Table 15.2.6-1
Time Sequence of Events for Loss of Non-Emergency AC Power to the Station Auxiliaries - Pressurizer Water Volume Analysis

Event	Time (sec)
Main feedwater flow stops	0
Low steam generator water level analytical limit reached	66
Reactor trip initiated (rod motion begins), ac power is lost, reactor coolant pumps begin to coastdown	68
Pressurizer safety valves open	70
Maximum RCS pressure occurs	71
Main steam safety valves open	75
Emergency feedwater initiated	208
Core decay heat decreases to emergency feedwater system heat removal capacity	2319
Maximum pressurizer water volume occurs	2319

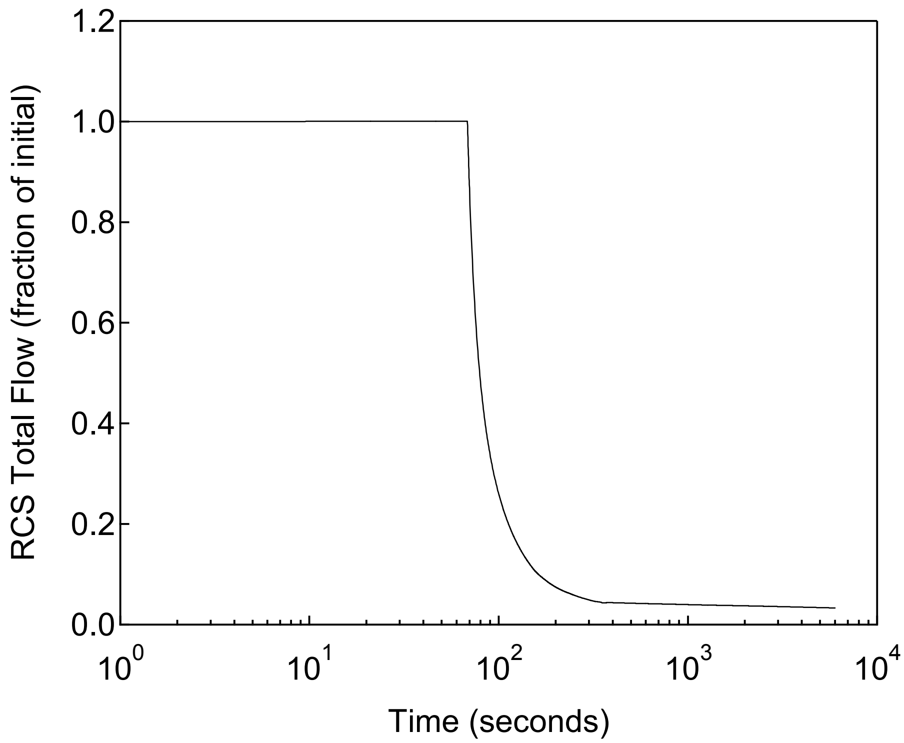


Figure 15.2.6-1 RCS Total Flow versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

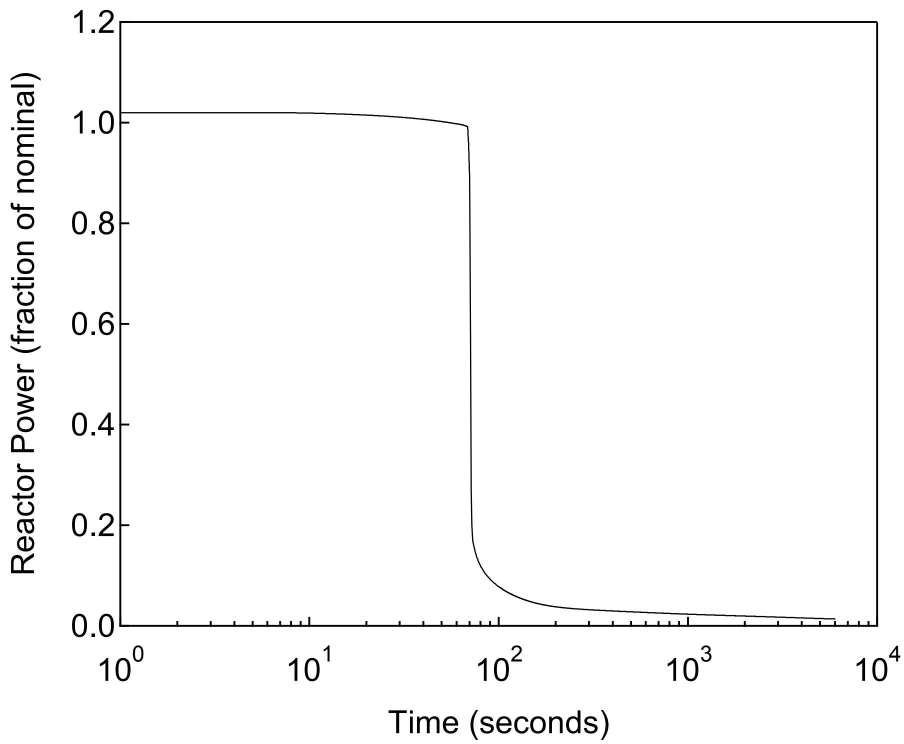


Figure 15.2.6-2 Reactor Power versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

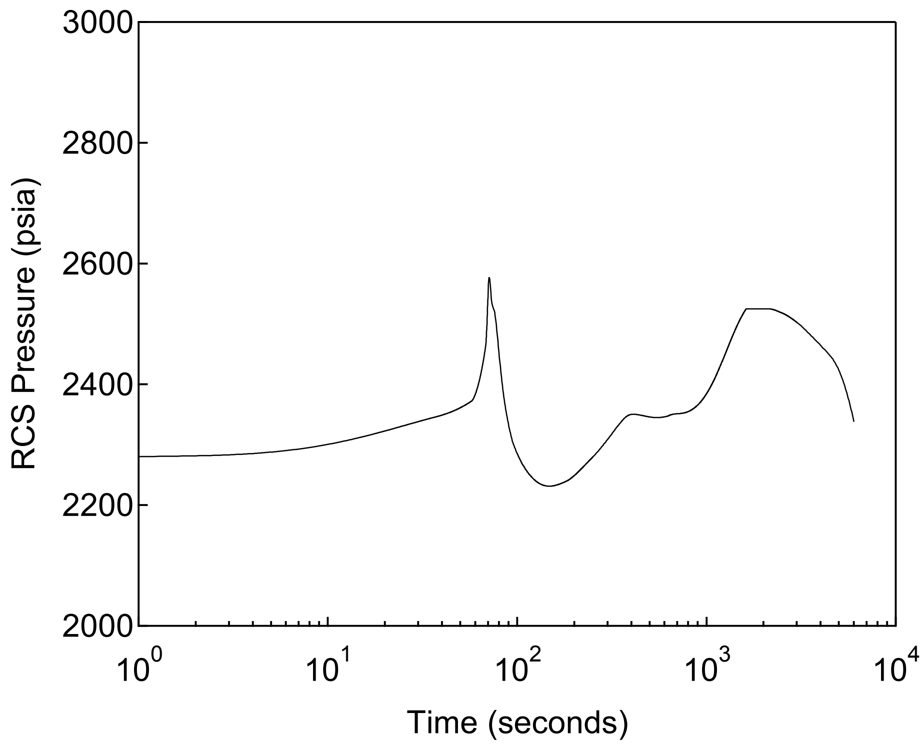


Figure 15.2.6-3 RCS Pressure versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

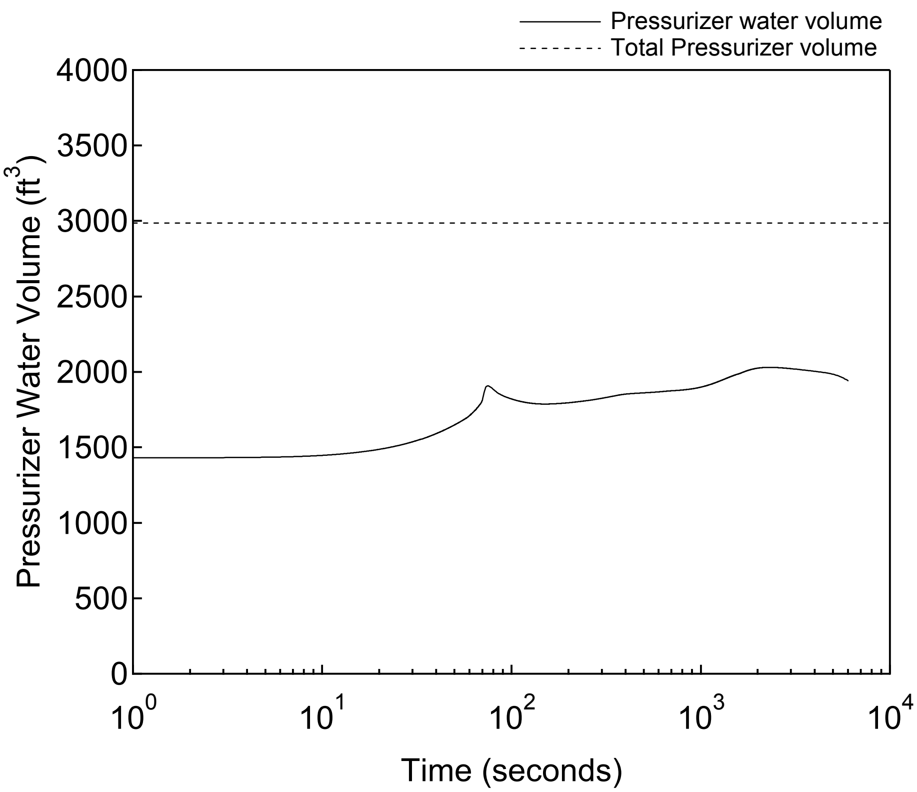


Figure 15.2.6-4 Pressurizer Water Volume versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

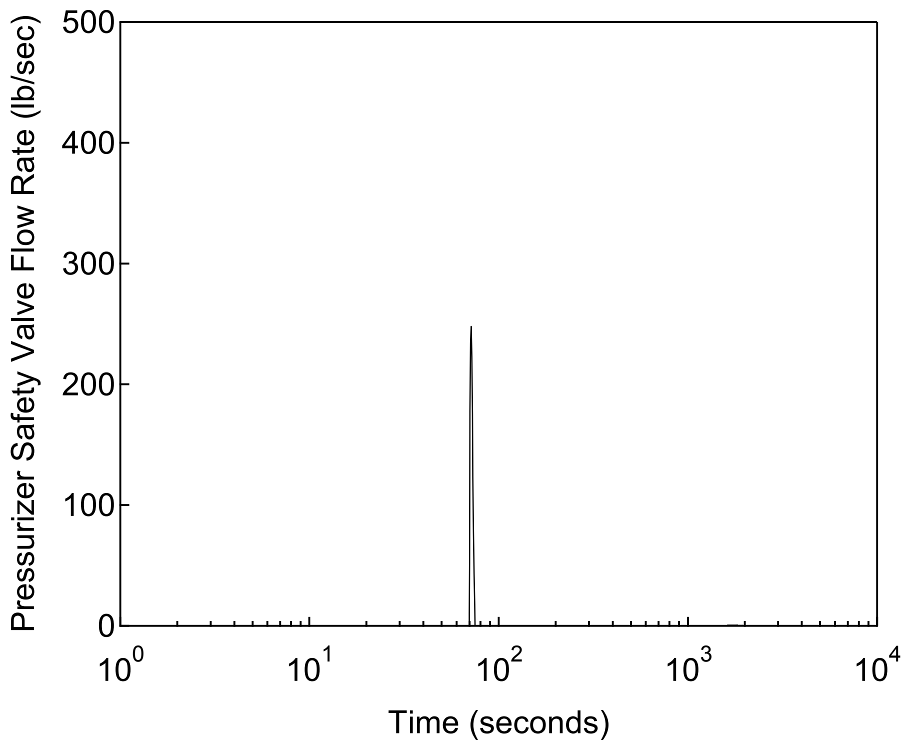


Figure 15.2.6-5 Pressurizer Safety Valve Flow Rate versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

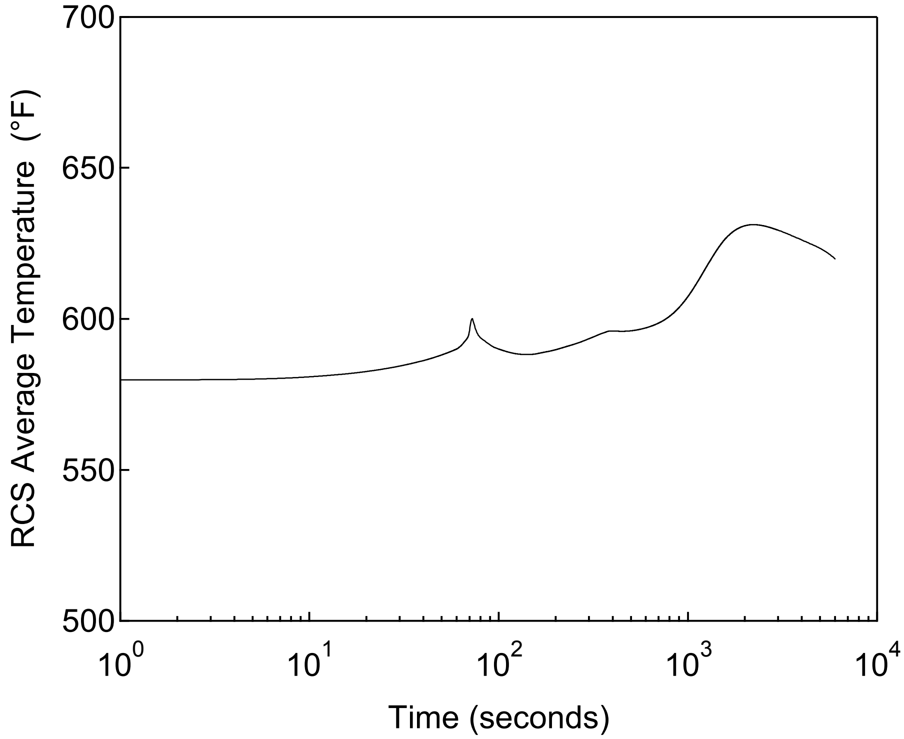


Figure 15.2.6-6 RCS Average Temperature versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

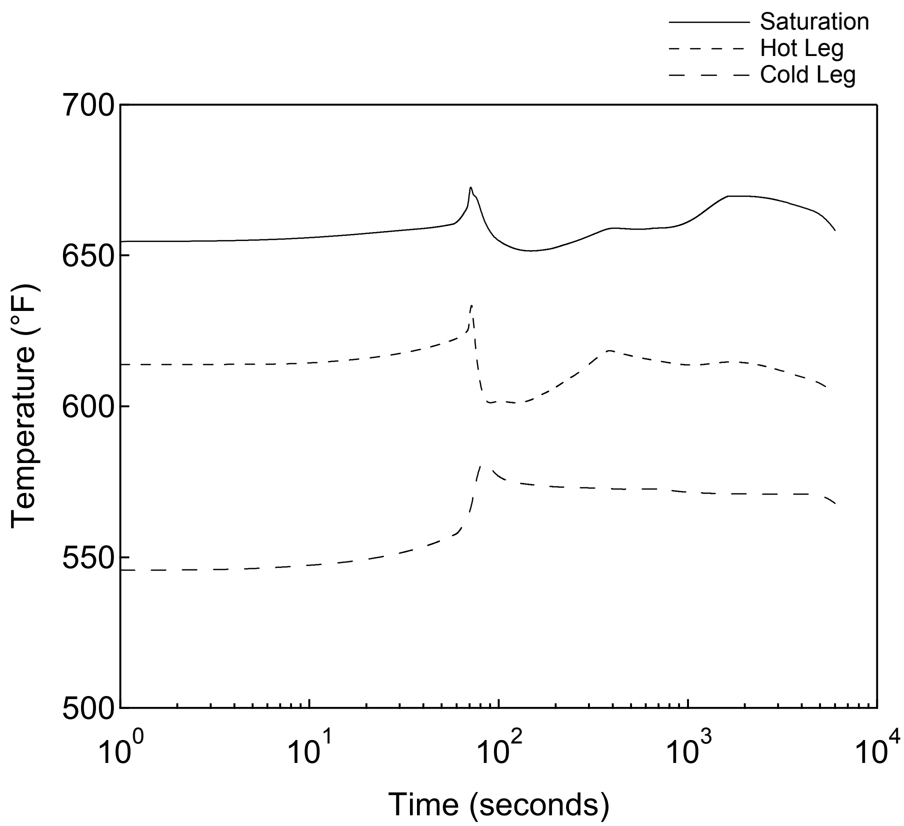


Figure 15.2.6-7 Temperature of Loop with EFW versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

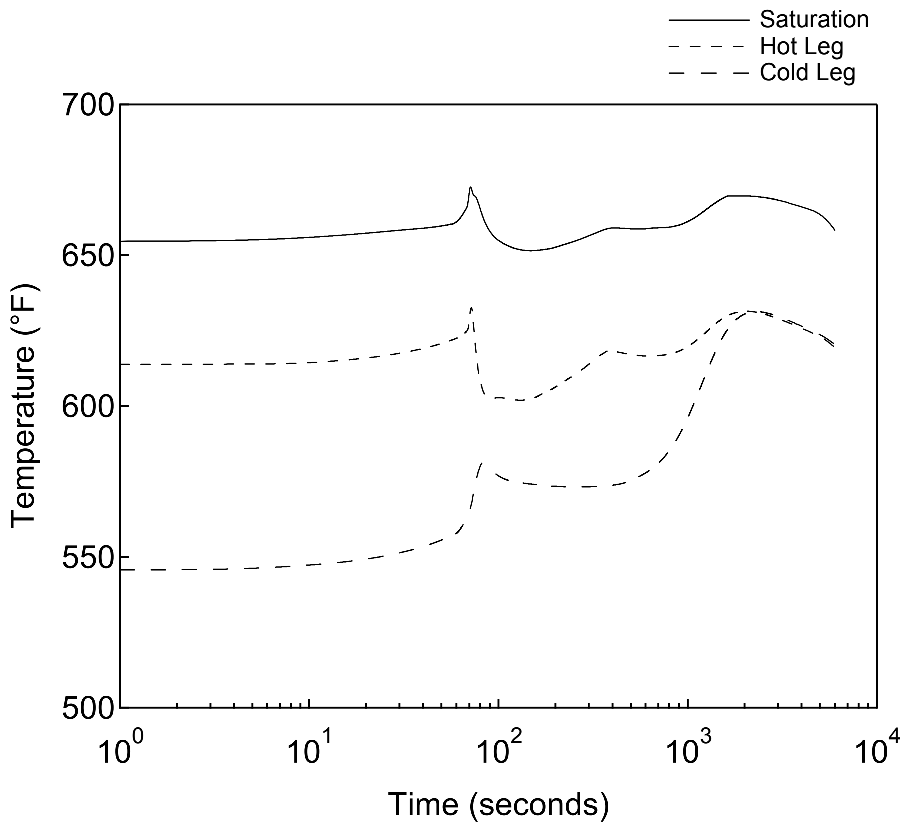


Figure 15.2.6-8 Temperature of Loop without EFW versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

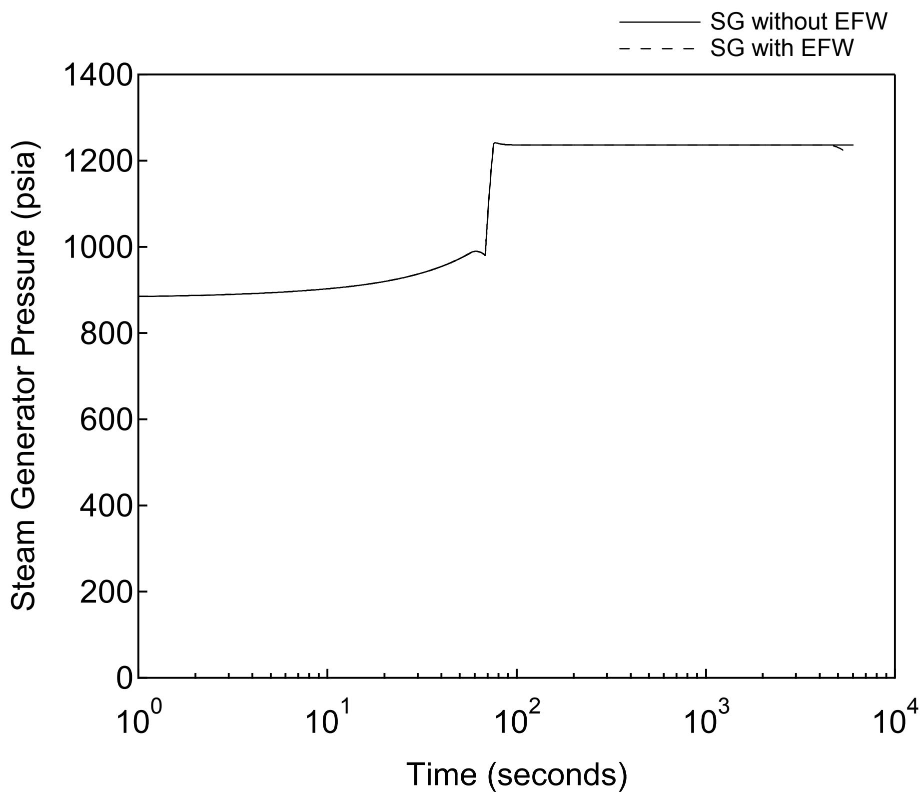


Figure 15.2.6-9 Steam Generator Pressure versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

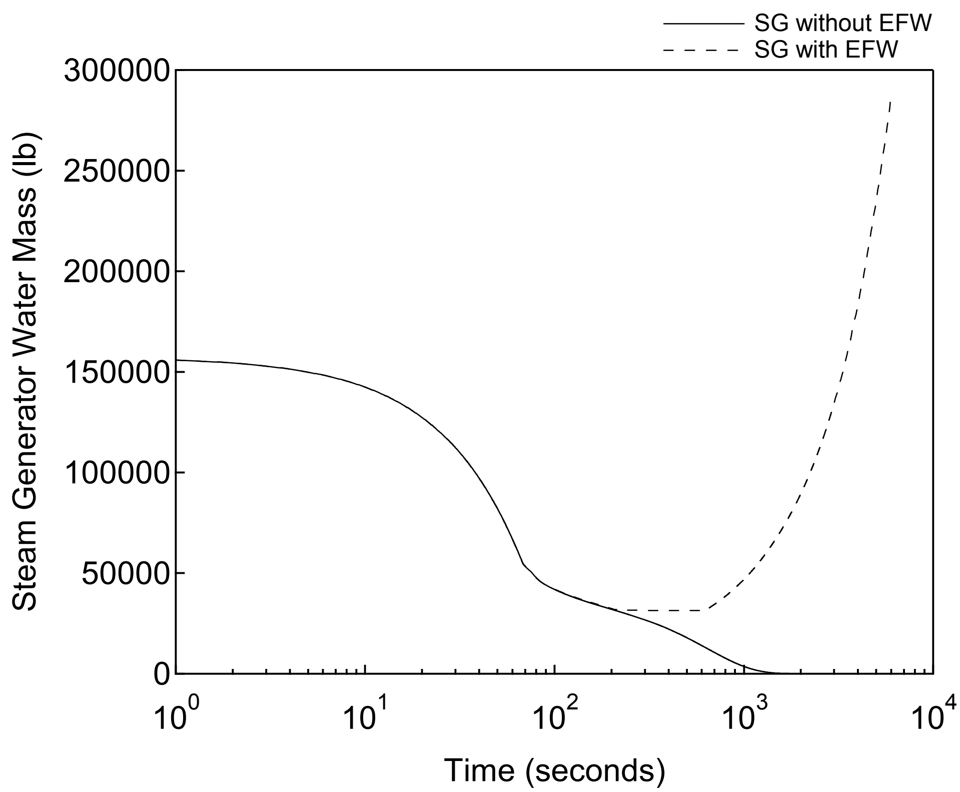


Figure 15.2.6-10 Steam Generator Water Mass versus Time

Loss of Non-Emergency AC Power to the Station Auxiliaries

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Frequency Classification

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power. The loss of feedwater flow results in a reduction of the secondary system's ability to remove heat generated by the reactor core. As a result, the reactor coolant temperature and pressure increase and will eventually require a reactor trip to protect the fuel and reactor coolant pressure boundary.

Normal feedwater flow can also be reduced by a feedwater system break, which can cause low steam generator water levels to occur prior to mitigative actions taken for the break. This scenario is addressed in the feedwater system pipe break transient (Section 15.2.8).

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.2-1). Event frequency conditions are described in Section 15.0.0.1.

15.2.7.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the loss of normal feedwater event is described in the results section.

The following sequence of events applies to the loss of normal feedwater flow due to pump failures or valve malfunctions. When normal feedwater is lost, the steam supply to the turbine is produced from boil off of the remaining water inventory in the steam generators. As a result, the water level of the steam generators will decrease. The reactor will trip on a low steam generator water level signal. This signal will also initiate the emergency feedwater (EFW).

After reactor trip, steam is produced from decay heat and sensible heat from the fuel and coolant. Since the main condenser is unavailable for steam dump by the turbine bypass system for this event, steam may be automatically relieved through the main steam relief valves. If the relief valves are unavailable, the main steam safety valves are used to relieve the steam. The relief valves or safety valves are used to dissipate residual decay heat as the no-load temperature is approached and to maintain the plant at hot standby conditions.

Unlike the loss of offsite power event, for the loss of normal feedwater event, the reactor coolant pumps (RCPs) are assumed to continue to supply water at their normal flow rates. The RCPs can be manually tripped at some later time following reactor trip to reduce heat addition to the reactor coolant system (RCS). For the analysis, it is conservatively assumed that the RCPs continue to operate throughout the event.

The reactor trip system (RTS) and EFWS are required to function following this event. No single active failure prevents any of these systems from performing their functions.

The worst single failure is the loss of one emergency feedwater system (EFWS) train, leaving three EFWS trains to supply EFW to their respective steam generators.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3. Indication of EFW flow is available in the main control room. Non safety-related systems are not assumed to mitigate the consequences of this event (as discussed in Section 15.0.0.5). Long-term cooling using the residual heat removal (RHR) system following this event is discussed in Section 15.0.0.8.

Loss of non-emergency ac power to the station auxiliaries was analyzed in Section 15.2.6. In that event, it was assumed that normal feedwater is initially lost and the reactor trips on low steam generator water level. Then ac power was assumed to be lost at the time of the reactor trip. The sequence of events for loss of non-emergency ac power is similar to that of loss of normal feedwater flow except that ac power is not lost in the latter event. As a result, the RCPs continue to operate instead of coasting down. Therefore, the case loss of normal feedwater flow with loss of offsite power is addressed by the results presented in Section 15.2.6.

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection from this transient:

- Over temperature ΔT
- High pressurizer pressure
- High pressurizer water level
- Low steam generator water level

The following features are assumed to be available to mitigate the accident:

- EFWS automatic actuation

Without main feedwater flow to the steam generators, makeup flow to the steam generators is supplied by the EFWS. The EFWS consists of two motor-driven pumps, two steam turbine-driven pumps, two emergency feedwater pits, and associated piping and valves. The four pumps take suction from the two emergency feedwater pits. Each of the four pumps is normally connected to only one steam generator with the cross-connect valves closed on the discharge paths in a “one-on-one” arrangement. The four-pump configuration and the cross-connected discharge header allow the system to remove the decay heat using three steam generators. Although the cross-connect valves could be opened by operator action, this is not credited in the accident analysis.

15.2.7.3 Core and System Performance

15.2.7.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following a loss of normal feedwater flow. This evaluation model is described in Section 15.0.2.2.1. Additional details on the MARVEL-M code are provided in Reference 15.2-4.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

Three cases are analyzed for the loss of normal feedwater flow event. The three cases are essentially the same except in the assumptions for initial values of reactor power, reactor coolant average temperature, and RCS pressure, and in the assumptions concerning whether or not pressurizer spray and the pressurizer heater are available. The core and system evaluation case assumes a combination of parameters, as described below, that limit DNBR. The barrier performance cases described in Section 15.2.7.4 use the same assumptions, except for initial conditions, pressurizer spray, and pressurizer heater as described in Section 15.2.7.4.2 to analyze cases that maximize peak RCS pressure and pressurizer water volume.

15.2.7.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for a loss of normal feedwater flow event:

- Consistent with the use of the RTDP, the assumed initial values of reactor power, reactor coolant average temperature, and RCS pressure are assumed to be the nominal values without uncertainties as defined in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to be the maximum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- The core residual decay heat generation is conservatively based on the 1979 version of ANSI/ANS 5.1 (Ref. 15.2-3).

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. Rod cluster control assembly insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- The reactor is assumed to be automatically tripped by the low steam generator water level signal. Table 15.0-4 summarizes the reactor trip setpoint and signal delay times assumed in the analysis.

- The EFW is actuated on the low steam generator water level signal. It is assumed that one EFWS train fails, leaving three trains to supply EFW to their respective steam generators.

- The event is analyzed with the pressurizer spray and the safety valves assumed to be available, and with the pressurizer heaters assumed to be unavailable.

- The event is analyzed with the main steam relief valves assumed to be unavailable.

15.2.7.3.3 Results

Prior to reactor trip, the loss of normal feedwater flow transient response is the same as the loss of non-emergency ac power to the station auxiliaries transient, discussed in Section 15.2.6. The difference between the two transients is that the RCPs are tripped and will coast down in the loss of non-emergency ac power transient, while they remain operational during the loss of normal feedwater flow event. Per Figure 15.2.7-1, the 95/95 limit DNBR criterion is met for the loss of normal feedwater flow transient.

It should be noted that the base analysis for the loss of normal feedwater flow is to evaluate peak RCS and main steam system pressures, and is presented in barrier performance Section 15.2.7.4. Therefore, only the DNBR versus time parameter plot is provided for the core response analysis. The responses of the other parameters shown in Figures 15.2.7-2 through 15.2.1-10 are approximately the same for this case.

15.2.7.4 Barrier Performance

15.2.7.4.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following a loss of normal feedwater. This evaluation model is described in Section 15.0.2.2.1. Additional details on the MARVEL-M code are provided in Reference 15.2-4.

Two analysis cases are performed as part of the barrier performance analyses, one to maximize peak RCS pressure and the other to maximize peak pressurizer water volume. These cases are identical to the core and system evaluation case for limiting DNBR, except for differences in initial conditions and control system operation assumptions.

15.2.7.4.2 Input Parameters and Initial Conditions

The following combination of initial conditions and input parameters are selected differently from the limiting DNBR case described in Section 15.2.7.3.2 in order to maximize RCS pressure. All other input parameters are the same as described in Section 15.2.7.3.2.

- The initial power level is 102% of the licensed core thermal power level with initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi above. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

- The event is analyzed with the pressurizer spray assumed to be unavailable. The unavailability of this equipment maximizes the RCS pressure for the barrier performance evaluation. However, the pressurizer heaters and the safety valves are assumed to be operable.

The following combination of initial conditions and input parameters are selected differently from the limiting DNBR case described in Section 15.2.7.3.2 in order to maximize the peak pressurizer water volume. All other input parameters are the same as described in Section 15.2.7.3.2.

- The only difference in initial conditions for the case that maximizes peak pressurizer water volume is that the initial reactor coolant temperature is 4°F below the nominal value and the pressurizer pressure is 30 psi above the nominal value. The initial core thermal power (102%) is the same as the case that maximizes RCS pressure.

- The pressurizer safety valves, spray, and heaters are all assumed to be available.

15.2.7.4.3 Results

The sequence and timing of events relative to initiation of the transient for the loss of normal feedwater flow is listed in Table 15.2.7-1.

Figures 15.2.7-2 to 15.2.7-10 are plots of system parameters versus time for the evaluated peak RCS pressure case. The loss of normal feedwater flow event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The RCP outlet pressure (Figures 15.2.7-3) is the highest pressure in the RCS and is presented in place of RCS pressure for the purpose of confirming the reactor coolant pressure boundary limits are not exceeded. The maximum reactor coolant pressure remains well below 110% of the design pressure. In addition, the steam generator pressure (Figures 15.2.7-9) does not exceed 110% of the main steam system design pressure. Therefore, the integrity of the reactor coolant pressure boundary and main steam system pressure boundary are maintained.

Figure 15.2.7-11 shows that under conditions that maximize pressurizer water volume, the maximum pressurizer water volume remains below the pressurizer capacity throughout the transient. Therefore, the pressurizer does not fill with water and water relief through pressurizer safety valves is precluded. EFW is sufficient to provide decay heat removal from the steam generators following reactor trip until the RHR system can be used. No other plots are provided for the peak pressurizer water volume case since the other parameters of interest are enveloped by the base analysis case.

The Figure 15.2.7-1 parameter plot for DNBR for the loss of normal feedwater flow event is presented as part of the core response analysis in Section 15.2.7.3. The response of reactor power, average core heat flux, and maximum core heat flux are almost indistinguishable for this transient. Therefore, only a plot of reactor power is provided. A plot of RCP outlet pressure is provided instead of a plot of RCS pressure, since this is the highest pressure in the RCS with the RCPs running. An RCS average temperature plot is provided instead of a core average or inlet coolant temperature to characterize the temperature response for this event. Plots for average and hot channel exit temperatures and steam fractions are not provided because this event is bounded by the more severe feedwater system pipe break event evaluated in Section 15.2.8, which also demonstrates margin to subcooling. Steam generator pressure is provided in place of steam line pressure for the purpose of demonstrating that the main steam system pressure meets the acceptance limit. All pressurizer safety valve flow is steam since the pressurizer does not fill; the plant does not have pressurizer relief valves used for pressure control. Containment parameters are not reported for this event because there are no releases directly from the RCS or steam generators inside containment.

15.2.7.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.2.7.6 Conclusions

The minimum DNBR remains above the 95/95 limit so that fuel integrity is not degraded for the loss of normal feedwater flow transient.

The RCS pressure and the main steam system pressure remain below 110% of their respective system design pressures so that the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained for this event.

The pressurizer does not fill with water for this event. Therefore, water relief through pressurizer safety valves is precluded. The EFW is sufficient to provide decay heat removal using the steam generators following reactor trip.

This event does not lead to a more serious fault condition.

Table 15.2.7-1
Time Sequence of Events for Loss of Normal Feedwater Flow - RCS Pressure Analysis Case

Event	Time (sec)
Main feedwater flow stops	0
Pressurizer safety valves open	44
Low steam generator water level analytical limit reached	66
Reactor trip initiated (rod motion begins)	68
Maximum RCP outlet pressure occurs	71
Main steam safety valves open	72
Emergency feedwater initiated	128
Core decay heat decreases to emergency feedwater system heat removal capacity	584

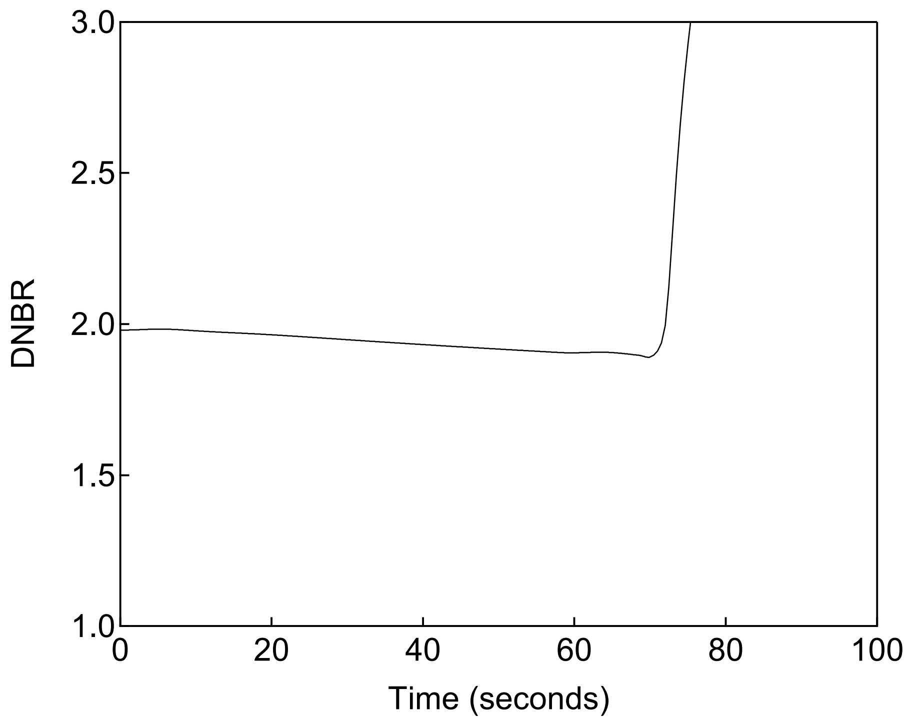


Figure 15.2.7-1 **DNBR versus Time**

Loss of Normal Feedwater Flow
- DNBR Analysis

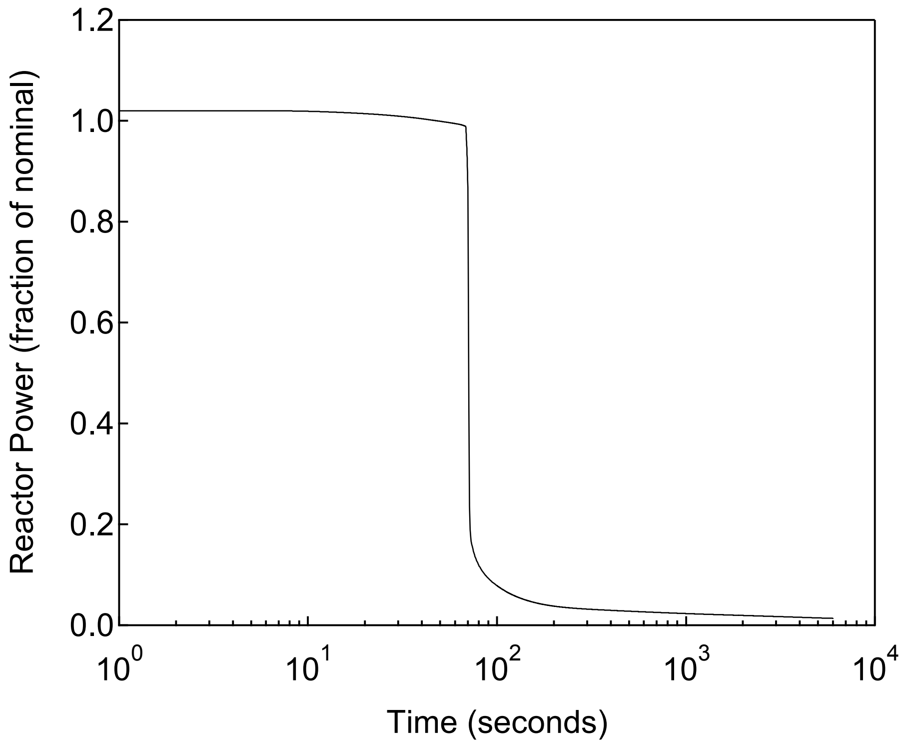


Figure 15.2.7-2 Reactor Power versus Time

Loss of Normal Feedwater Flow
- RCS Pressure Analysis

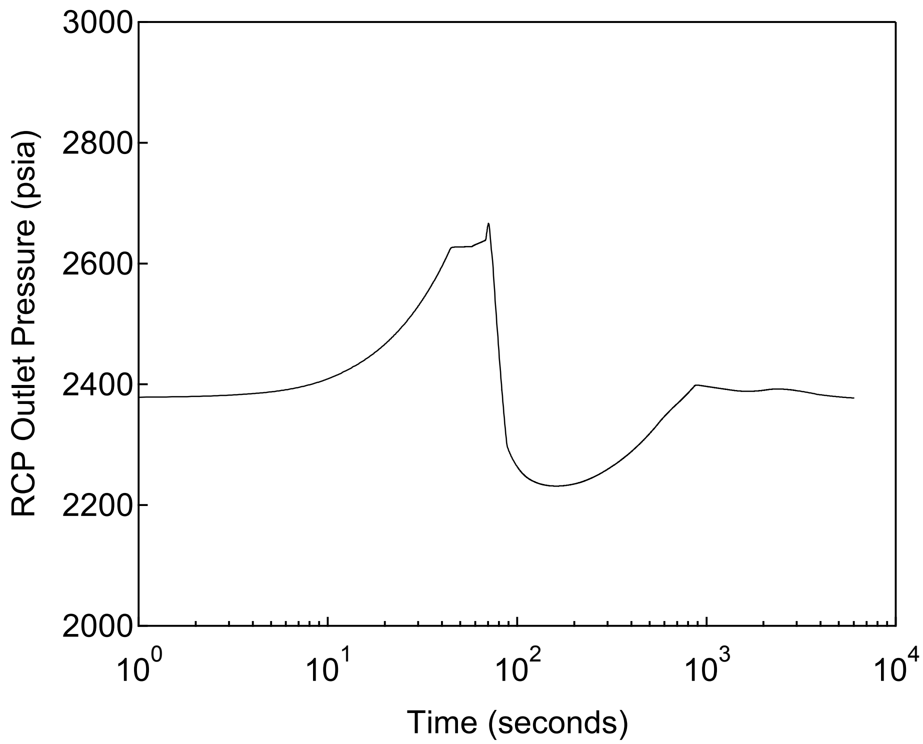


Figure 15.2.7-3 RCP Outlet Pressure versus Time

Loss of Normal Feedwater Flow
- RCS Pressure Analysis

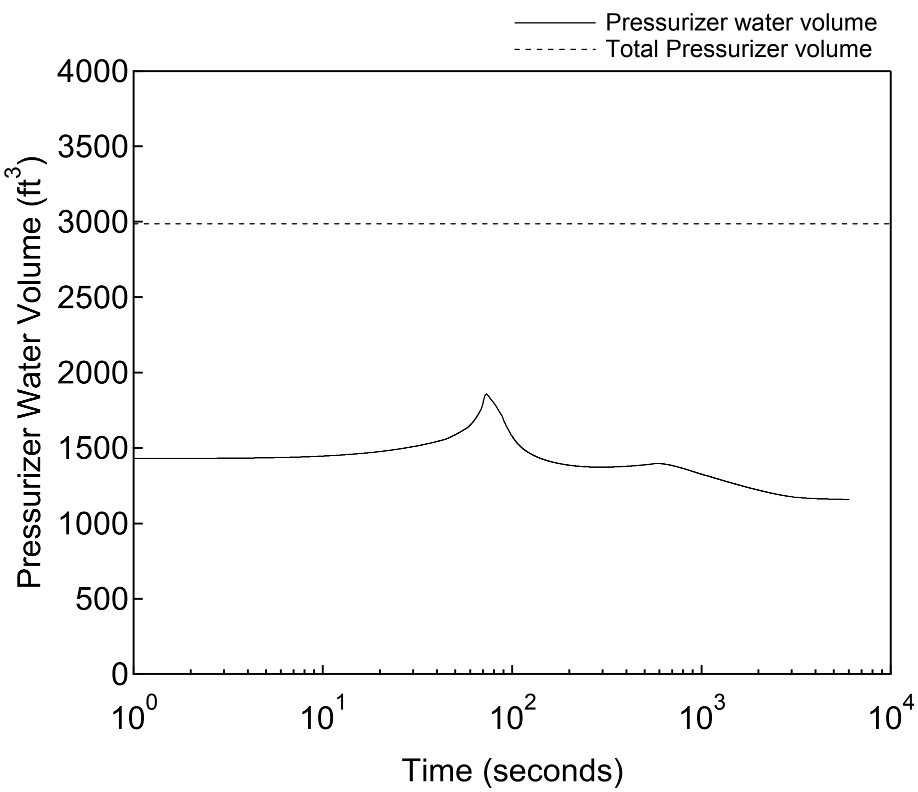


Figure 15.2.7-4 Pressurizer Water Volume versus Time

Loss of Normal Feedwater Flow - RCS Pressure Analysis

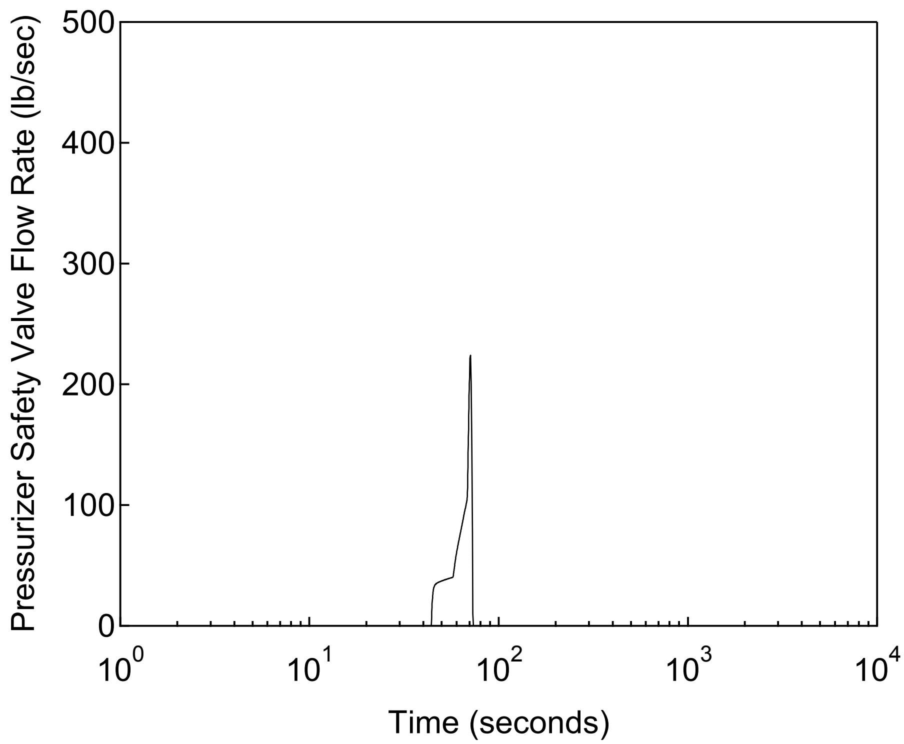


Figure 15.2.7-5 Pressurizer Safety Valve Flow Rate versus Time

Loss of Normal Feedwater Flow
- RCS Pressure Analysis

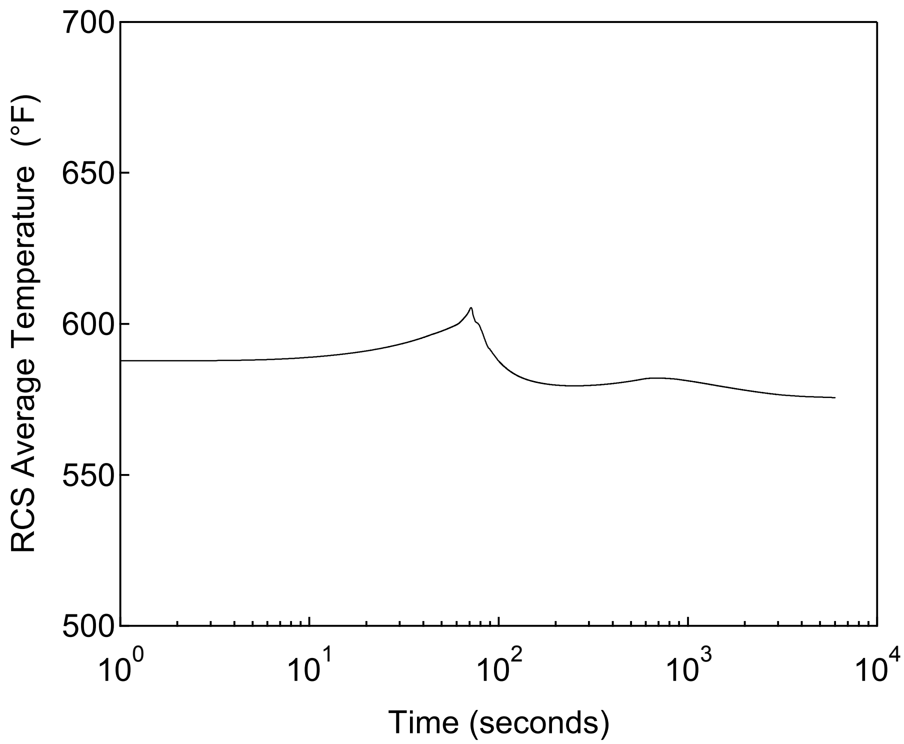


Figure 15.2.7-6 RCS Average Temperature versus Time

Loss of Normal Feedwater Flow
- RCS Pressure Analysis

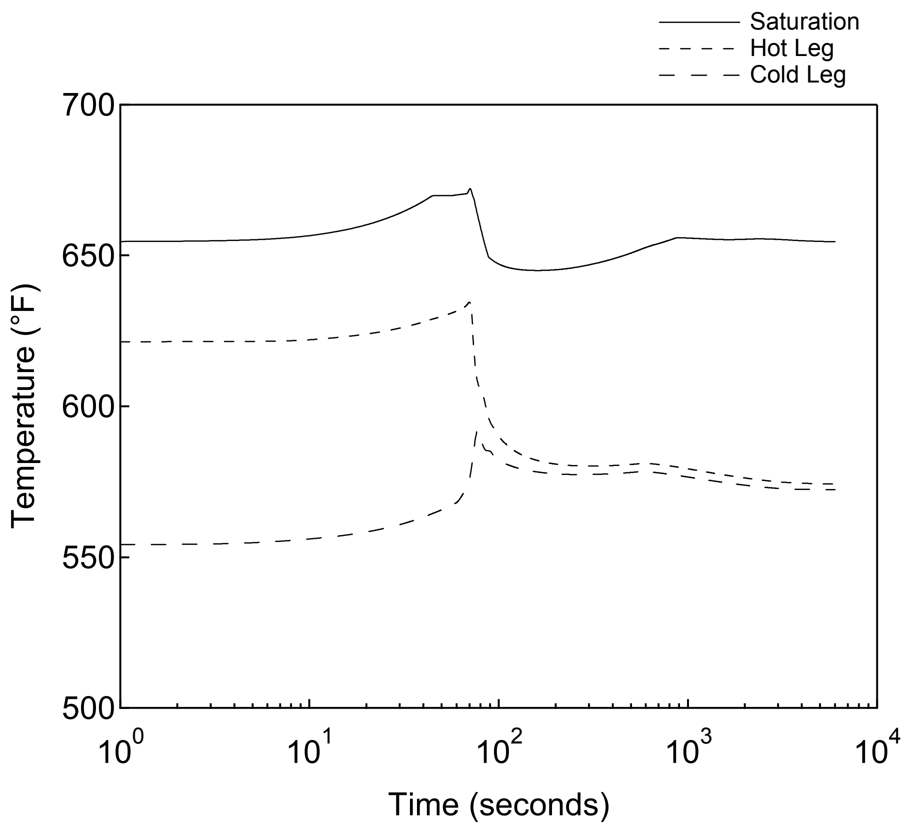


Figure 15.2.7-7 Temperature of Loop with EFW versus Time

Loss of Normal Feedwater Flow
- RCS Pressure Analysis

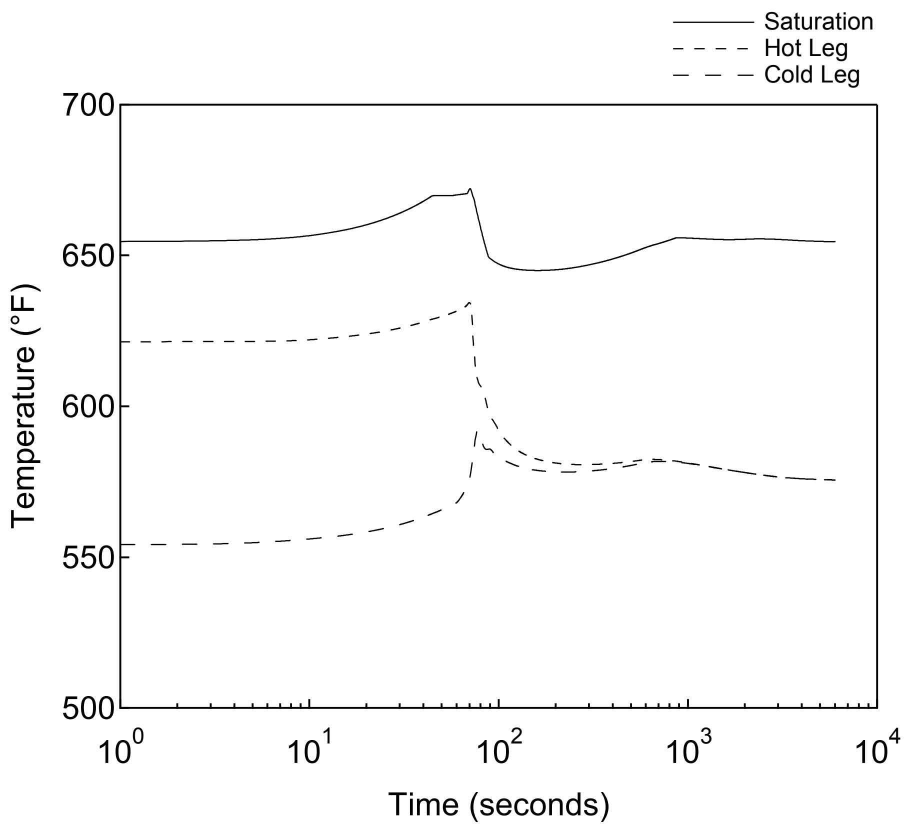


Figure 15.2.7-8 Temperature of Loop without EFW versus Time

Loss of Normal Feedwater Flow
- RCS Pressure Analysis

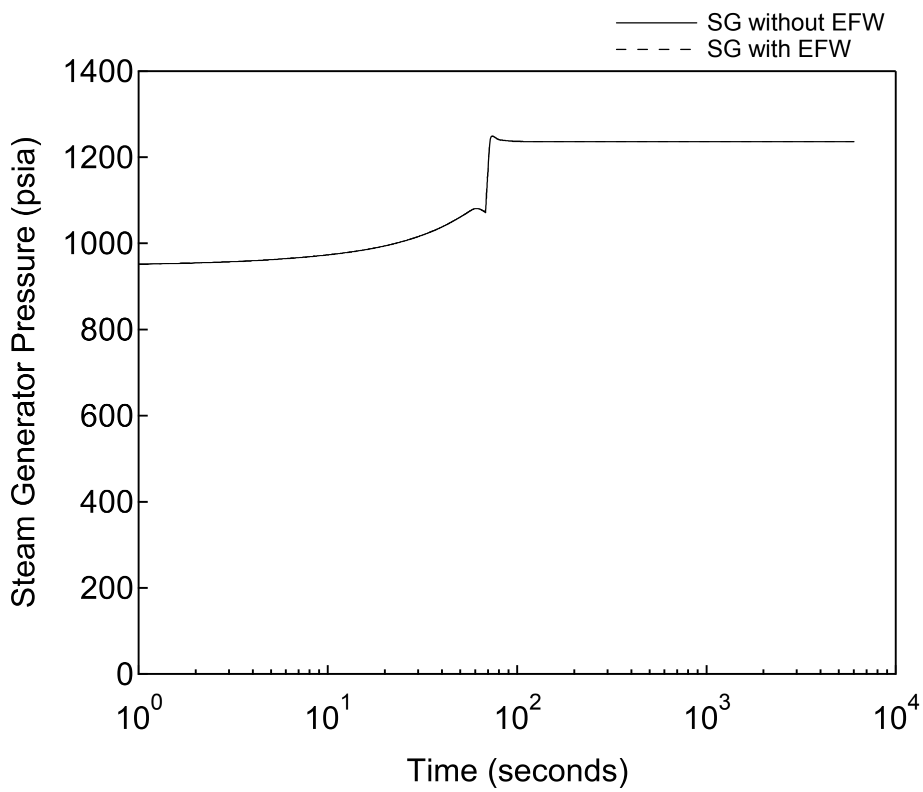


Figure 15.2.7-9 Steam Generator Pressure versus Time

Loss of Normal Feedwater Flow
- RCS Pressure Analysis

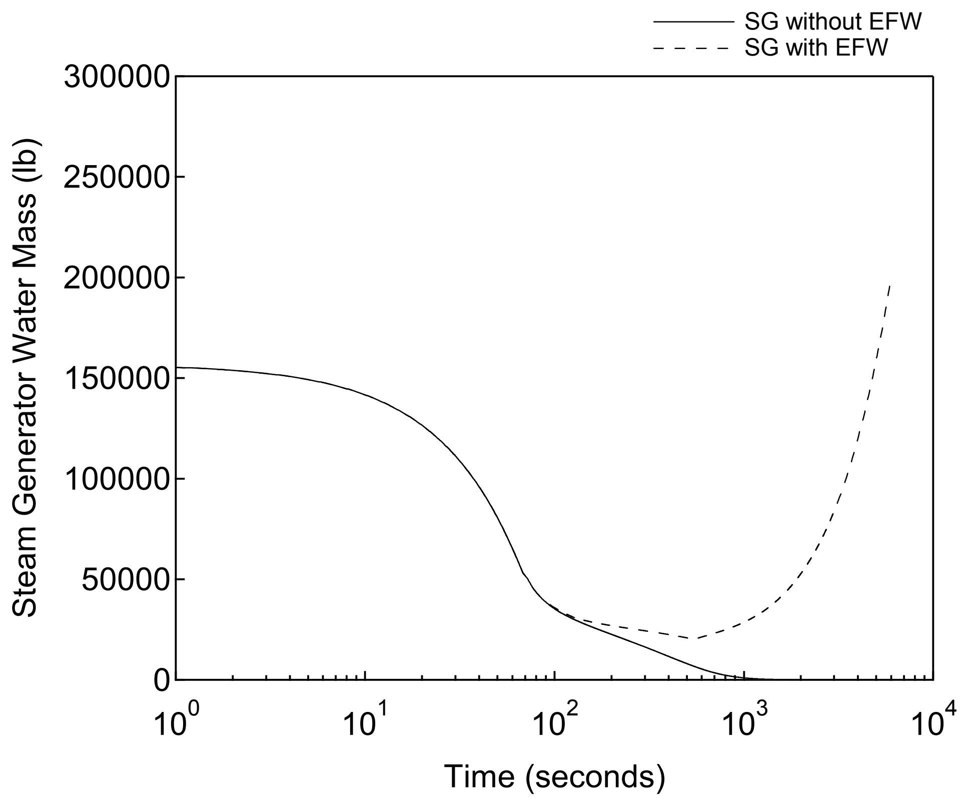


Figure 15.2.7-10 Steam Generator Water Mass versus Time

Loss of Normal Feedwater Flow
- RCS Pressure Analysis

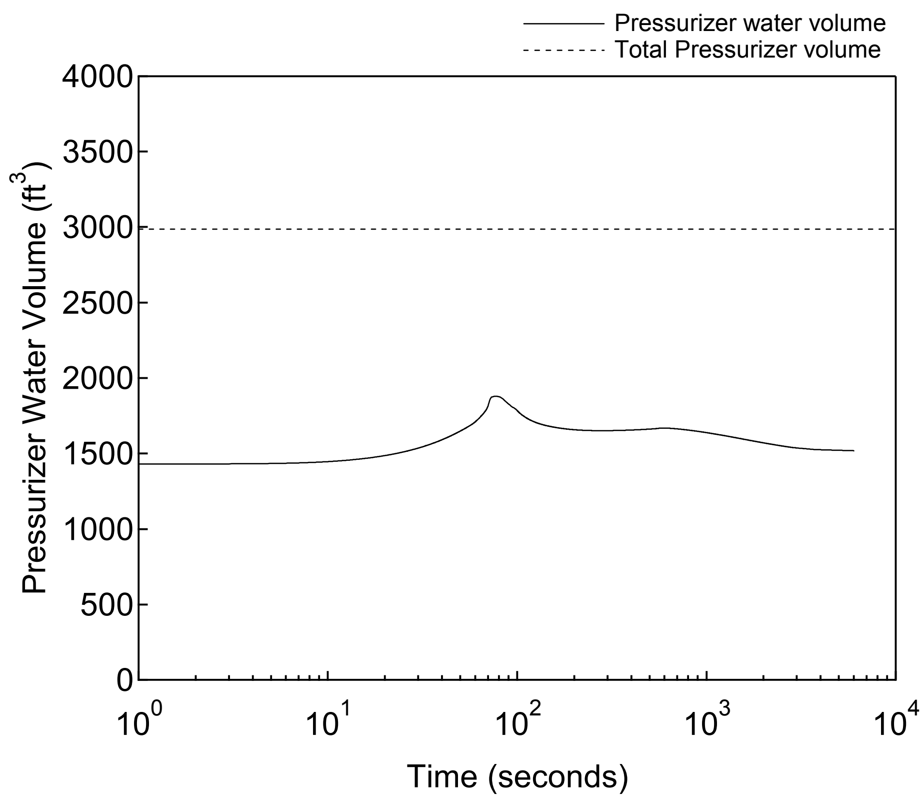


Figure 15.2.7-11 Pressurizer Water Volume versus Time

Loss of Normal Feedwater Flow - Pressurizer Water Volume Analysis

15.2.8 Feedwater System Pipe Break Inside and Outside Containment

The feedwater system pipe break is a non-uniform transient that involves modeling the flow from one of the secondary loops. Unlike the secondary piping rupture resulting in reactor coolant system (RCS) cool down analyzed in Section 15.1.5, the feedwater system pipe break analyzed in this section causes a loss of inventory from the saturated liquid mass in the steam generator resulting in RCS heat-up and pressurization. Unless the heat-up of the RCS is mitigated, there will be a possibility of water relief through the pressurizer safety valve.

15.2.8.1 Identification of Causes and Frequency Classification

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 shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the main feedwater check valve and the steam generator, fluid from the steam generator may also be discharged through the break. A break upstream of the main feedwater check valve would affect the plant as a loss of feedwater event.

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

- Feedwater flow to the steam generators is reduced. Because feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip.
- Fluid in the steam generator may be discharged through the break and would not be available for decay heat removal after trip.
- The break may be large enough to prevent the addition of main feedwater after trip.

The severity of the effects of a main feedwater line break depends on the location of the break, the size of the break, initial plant operating conditions, control system availability, and safety system availability.

For breaks that are small enough to not be considered a major feedwater line rupture, the plant continues to operate without the need for reactor trip system (RTS) or engineered safety feature actuation. Minor feedwater system pipe break accidents are classified as anticipated operational occurrences (AOOs). Minor feedwater pipe breaks resulting in continued feedwater addition, but at a rate insufficient to maintain steam generator level, are bound by the loss of normal feedwater response evaluated in Section 15.2.7.

The most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. A double-ended rupture of the feedwater piping between the main feedwater check valve and steam generator bounds the remaining large break cases. Smaller breaks will result in a slower reduction in steam generator heat removal capability. This double-ended break of a main feedwater system pipe is classified as a postulated accident (PA).

Event frequency conditions are described in Section 15.0.0.1. In addition to the general AOO and PA acceptance criteria described in Section 15.0.0.1, MHI conservatively adopts two additional acceptance criteria: (1) to evaluate hot leg boiling and (2) to not allow the pressurizer to overfill.

15.2.8.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the feedwater pipe rupture event is described in the results section.

This analysis evaluates the effects of the double-ended rupture of one feedwater line between the main feedwater check valve and the steam generator, which results in the rapid blowdown of one steam generator through the ruptured piping. The emergency feedwater system (EFWS) train that would normally supply the broken loop will spill out of the feedwater line and not contribute to the removal of heat from the primary system. In addition, a single failure of one of the other three EFWS trains is assumed.

The break is assumed to occur concurrent with the low steam generator water level reactor trip signal resulting from the loss of feedwater flow assumed as a precondition. This conservative precondition minimizes the total steam generator inventory available to remove heat from the RCS and makes the RTS response independent of the steam generator pressure and level dynamics of the feedwater line break prior to the reactor trip. A loss of offsite power is assumed to occur at that time concurrent with the turbine trip. The emergency feedwater (EFW) is also started on a low steam generator water level signal. EFW flow does not enter the affected steam generator. EFW is subsequently automatically isolated to the affected steam generator by isolation logic developed from low main steam line pressure signals. As a single failure, one of the remaining intact steam generators is assumed to not receive EFW flow.

Only the case without offsite power available is presented because this case is more limiting than the case assuming the reactor coolant pumps (RCPs) remain running until the emergency core cooling system (ECCS) signal is reached. In the case presented, the RCPs are tripped concurrent with the turbine trip. The only difference between the cases with and without offsite power available is the operating status of the RCPs.

The protective actions to mitigate the accident are isolation of the affected steam generator by closing the EFW isolation valve and terminating the EFW supply to the affected steam generator, and cooling of the RCS by supplying EFW to the intact steam generators. The EFWS has two motor-driven and two turbine-driven emergency feedwater pumps.

Each emergency feedwater pump supplies emergency feedwater independently to each steam generator taking water from the emergency feedwater pits. The EFWS is sized to have the capability of supplying sufficient EFW to preclude the pressurizer filling with water during a postulated feedwater system pipe break, assuming a single failure in loss of one EFWS train. The protective actions are automated for the US-APWR.

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection from this transient:

- Low steam generator water level in any loop
- High pressurizer pressure
- High pressurizer water level
- Over temperature ΔT
- ECCS actuation

The following features are assumed to be available to mitigate the accident:

- EFWS automatic actuation
- EFW isolation
- ECCS

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event (as discussed in Section 15.0.0.5). Long-term cooling using the residual heat removal (RHR) system following this event is discussed in Section 15.0.0.8.

15.2.8.3 Core and System Performance

It should be noted that fuel rod failure resulting from DNB is of primary concern when the reactor is operating at power, not during a heat-up following a reactor trip. As a result of the way the transient is initiated, DNBR is not a parameter calculated during this transient following a reactor trip. The minimum DNBR for the pre-trip portion of this event is not calculated because it is bounded by the minimum DNBR for the loss of normal feedwater event analyzed in Section 15.2.7. Subcooling is evaluated to preclude steam binding in the steam generator U-tubes for the steam generators receiving EFW flow during natural circulation flow conditions and to preclude the need to model reflux boiling heat transfer during the transient. This analysis is presented as part of the Barrier Performance analysis in Section 15.2.8.4 below.

RCP seal reliability and integrity during loss of alternating-current power and loss of coolant to the seals (e.g., a result of containment isolation) must comply with 10 CFR 50.34(f)(1)(iii). For this event, the containment is isolated by the ECCS actuation signal. Therefore, seal water to the reactor coolant pumps may be lost. RCP seal integrity following containment isolation is discussed in Section 15.0.0.9.

15.2.8.4 Barrier Performance

15.2.8.4.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following a feedwater pipe break. This evaluation model is described in Section 15.0.2.2.1.

The effects of blowdown loads from steam system piping failures (e.g., pipe whip and jet impingement) on plant structures, systems, and components is evaluated in

Section 3.6.2.

The feedwater system pipe break causes non-balanced operation, e.g. a faulted steam generator loop, intact loops with EFW supply and an intact loop without EFW supply. The capability to model up to 4 separate loops in the MARVEL-M code is used for the analysis. The MARVEL-M code also models reactor thermal kinetics (decay heat), RCS response including temperatures, pressure, pressurizer level, and flow (natural circulation), as well as the non-uniform primary-to-secondary heat transfer caused by the break and EFWS single failure and heat removal from the main steam safety valves in the intact steam generators.

The MARVEL-M code calculates the break flow using the Moody correlation, taking account of the flow restriction at the feedwater inlet nozzle, assuming saturated liquid (quality=0) flow. This assumption maximizes the rate at which the affected steam generator inventory is depleted and primary-to-secondary heat transfer is decreased, resulting in a conservative RCS heat-up. In reality, the feedline water discharge could entrain steam when the water level in the affected steam generator decreases significantly. In this case, the energy removed from the RCS by the steam release could mitigate the heat-up of the RCS. This effect is conservatively neglected in the analysis to present the worst-case RCS heat-up results for the feedwater system pipe break event.

The methodology for analysis of the feedwater line break event using MARVEL-M is further described in Reference 15.2-4.

Three cases are analyzed for the feedwater line break event. Each case is designed to evaluate the worst case conditions for (1) peak RCS pressure, (2) hot leg boiling, or (3) pressurizer water volume.

15.2.8.4.2 Input Parameters and Initial Conditions

Three cases designed to evaluate the worst case conditions for (1) peak RCS pressure, (2) hot leg boiling, or (3) pressurizer water volume are evaluated. The primary differences between these cases are the assumptions concerning initial conditions and the assumed availability of pressurizer spray and pressurizer heaters. The following table summarizes the differences in these parameters for the three cases followed by a discussion of the input parameters that are consistently used in all three cases.

Case No.	Description	Uncertainties Assumed with Table 15.0-3 Nominal Values			Control System	
		Pressure (psi)	T_{avg} (°F)	Power	Pressurizer Spray	Pressurizer Heaters
1	Peak RCS Pressure	+30	+4	+2%	Off	On
2	Hot Leg Boiling	-30	+4	+2%	On	Off
3	Peak Pressurizer Water Volume	+30	-4	+2%	On	On

- The core residual decay heat generation is conservatively based on the 1979 version of ANSI/ANS 5.1 (Ref. 15.2-3).

- The break is assumed to occur concurrent with the low steam generator water level reactor trip signal resulting from a loss of feedwater flow assumed as a precondition. That is, steam generator water levels are at the low steam generator water level setpoint for all steam generators when the break occurs. After the low steam generator water level reactor trip signal is generated, the saturated water in the steam generator on the broken loop is rapidly discharged. Primary-to-secondary heat transfer area is reduced when the level is below the top of the tubes in the affected steam generator. This minimizes the heat removal by the affected steam generator and maximizes the RCS temperature.

- No credit is taken for the reactor trip on the over temperature ΔT, high pressurizer pressure, or high pressurizer water level.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to be the maximum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. Rod cluster control assembly insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- The elevation head associated with the core active fuel height is modeled to confirm natural circulation flow is established and is capable of removing decay and residual heat through the steam generators.

- The EFW is actuated on the low steam generator water level signal. EFW flow does not enter the faulted steam generator. The assumed single failure is the loss of one EFWS train, which results in one of the remaining intact steam generators not receiving EFW flow. The remaining two steam generators are intact and supplied with EFW flow.

- The event is analyzed assuming steam relief from the pressurizer safety valves. One of the cases verifies that the pressurizer does not over fill for this event.

- The event is analyzed with the main steam relief valves assumed to be unavailable. The unavailability of this equipment maximizes the RCS and main steam pressure and the pressurizer water level for the barrier performance evaluation. However, the safety valves are assumed to be operable.

- Heat deposited in the RCS metal is not credited during RCS heat-up. The steam generator heat transfer area is decreased as the shell-side water inventory decreases.

15.2.8.4.3 Results

The sequence of events for the feedwater system pipe break is listed in Table 15.2.8-1. The results for the analyzed transient cases are provided in Figures 15.2.8-1 to 15.2.8-16. Figures 15.2.8-1 through 15.2.8-12 depict the transient parameters of interest for the base analysis case that maximizes RCS pressure. Figures 15.2.8-13 through 15.2.8-15 depict the RCS loop temperatures for conditions that maximize hot leg boiling. And Figure 15.2.8-16 depicts the pressurizer water volume transient for the peak pressurizer water volume case. A limited number of selected plots, as previously described, are provided for the hot leg boiling and peak pressurizer water volume cases since the other parameters of interest are enveloped by the base analysis case.

The transient results for all of the cases are very similar. For the peak RCS pressure case, the steam generator mass is depleted very rapidly as shown by the affected steam generator mass and break flow in Figures 15.2.8-11 and 15.2.8-12. Primary-to-secondary heat transfer area is reduced when the level is below the top of the tubes in the affected steam generator, resulting in a heat-up of the faulted loop as shown in Figure 15.2.8-7. The RCS loop average temperature and steam generator mass figures depicts the differences between the response of this steam generator and the other steam generators receiving EFW flow. As the transient progresses, the three intact steam generators are heated up and pressurized to the main steam safety valve pressure. The intact steam generator without EFW flow gradually boils off its inventory through the safety valves, and its heat transfer also starts to decrease when the water level reaches the top of the U-tubes. The two remaining intact steam generators with EFW flow also boil off the EFW flow through the safety valves, which provides RCS cooling. The transient “turns around” at the point where decay heat balances the steam generator heat removal capability. This occurs at time t=1683 seconds, at which time the pressurizer water volume peaks and begins to decrease.

Figure 15.2.8-16 shows that the maximum pressurizer water volume remains below the pressurizer capacity throughout the transient. Therefore, the pressurizer does not fill with water. In addition, Figure 15.2.8-15 shows that boiling in the hot leg does not occur in the intact loops receiving EFW flow; subcooling margin is maintained.

Examination of Figures 15.2.8-2 and 15.2.8-10 shows that the RCS pressure and main steam system pressure are well below 110% of their design values, meeting both the PA (120%) and AOO (110%) acceptance criterion for RCS and main steam system pressure.

The containment may be pressurized by the blowdown from the affected steam generator. However, the containment pressure is bounded by the large break loss-of-coolant accident discussed in Section 15.6.5. Similarly, containment temperature response is bounded by the containment vessel response to steam system piping failures as discussed in Section 6.2. Therefore, containment integrity is maintained.

As explained in Section 15.2.8.3, DNBR is not evaluated for this event. Fuel & cladding temperatures are not key parameters and parameter plots for them are not provided for this event. The response of reactor power, average core heat flux, and maximum core heat flux are almost indistinguishable for this transient. Therefore, only a plot of reactor power is provided. Similarly, reactivity and steam flow are not key parameters and are not presented. A plot of RCP outlet pressure is provided instead of a plot of RCS pressure since this is the highest pressure in the RCS with the reactor coolant pumps running. An RCS average temperature plot is provided instead of a core average or inlet coolant temperature to characterize the temperature response for this event. Because there is subcooling margin and DNB does not occur, no plot is provided for the average exit temperature. A parameter plot for steam generator water mass is provided in place of steam generator water volume. This parameter plot shows the effect of EFW flow, which is described in the text, on steam generator inventory. A feedwater line break flow rate parameter plot is provided in place of feedwater flow. Mass and energy release and containment vessel response is discussed in Section 6.2 and are therefore not presented in this section.

In response to GDC 31, this event (including operation of the ECCS under low-temperature conditions) has been considered in the design of the reactor coolant pressure boundary to assure that the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture has been minimized. Fracture toughness of the reactor coolant pressure boundary and reactor vessel is described in Sections 5.2.3 and 5.3.1.

The effects of blowdown loads from feedwater system failures (e.g., pipe whip and jet impingement) on plant structures, systems, and components is evaluated in Section 3.6.2.

15.2.8.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.2.8.6 Conclusions

The minimum DNBR remains above the 95/95 limit so that fuel integrity is not degraded for the loss of normal feedwater flow transient.

The RCS pressure and the main steam system pressure remain below their respective pressure limits so that the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained for this event.

The pressurizer does not fill with water for this event. Therefore, water relief through pressurizer safety valves is precluded.

Additionally, subcooling margin is maintained since boiling in the hot leg does not occur in the intact loops receiving EFW flow.

The radiological consequences of this event are bounded by the main steam line break accident.

This event does not lead to a more serious fault condition.

Table 15.2.8-1
Time Sequence of Events for Feedwater System Pipe Break
- RCS Pressure Analysis

Event	Time (sec)
Loss of feedwater flow occurs	0
Pressurizer safety valves open	44
Low steam generator water level analytical limit reached	66
Feedwater line break initiated	66
Reactor trip initiated (rod motion begins)	68
RCP coastdown begins	68
Peak RCP outlet pressure occurs	71
Main steam safety valves open	72
EFW isolated to faulted loop	107
EFW pumps start	208
Peak pressurizer water volume occurs	1683

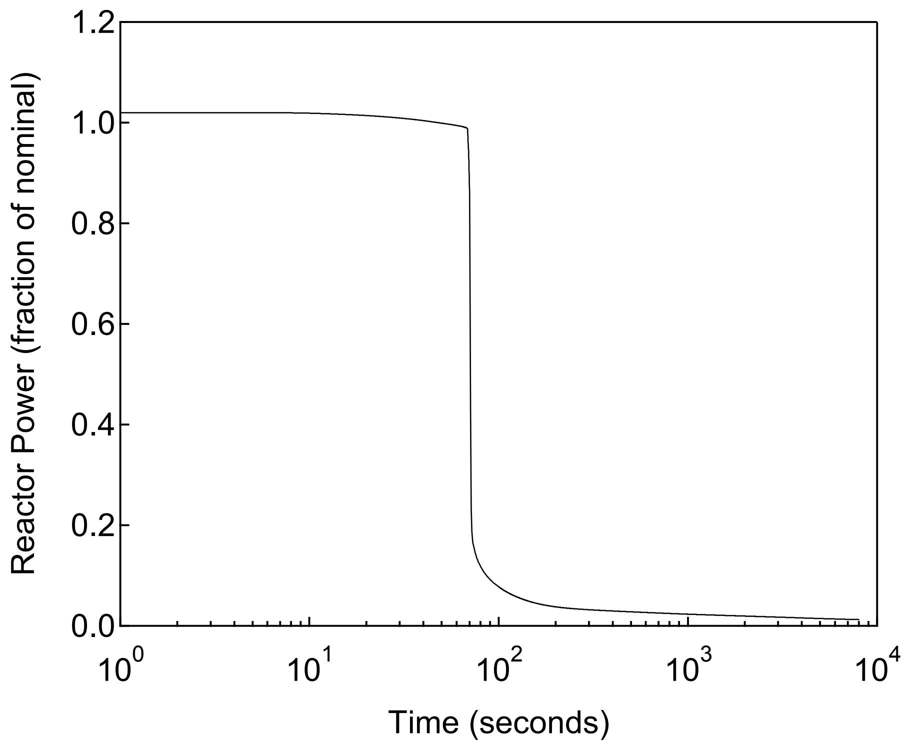


Figure 15.2.8-1 Reactor Power versus Time

Feedwater System Pipe Break - RCS Pressure Analysis

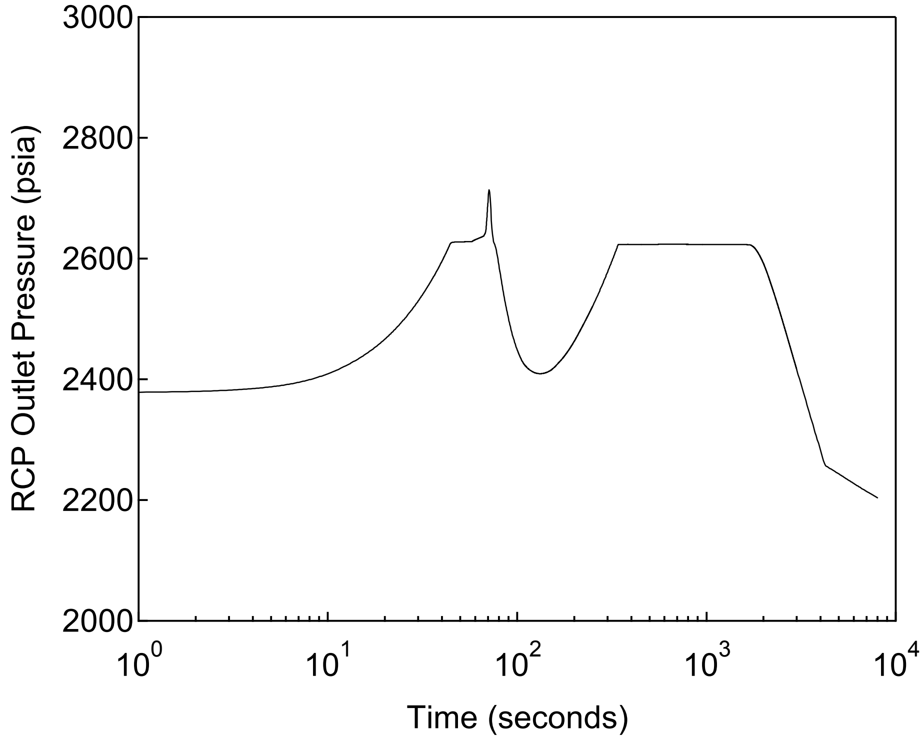


Figure 15.2.8-2 RCP Outlet Pressure versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

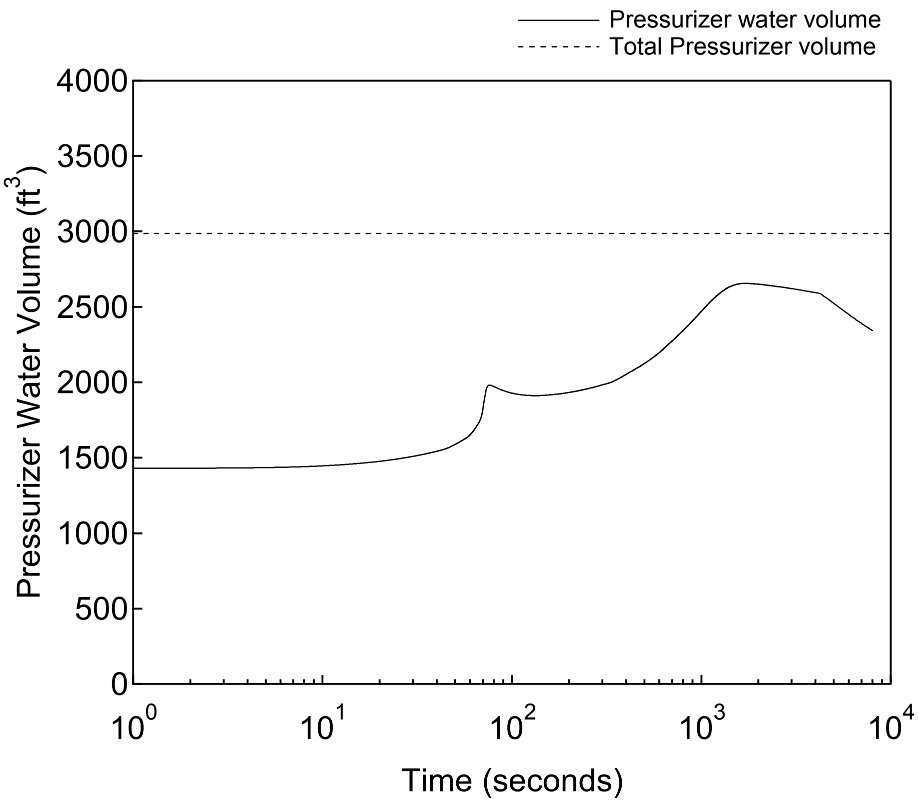


Figure 15.2.8-3 Pressurizer Water Volume versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

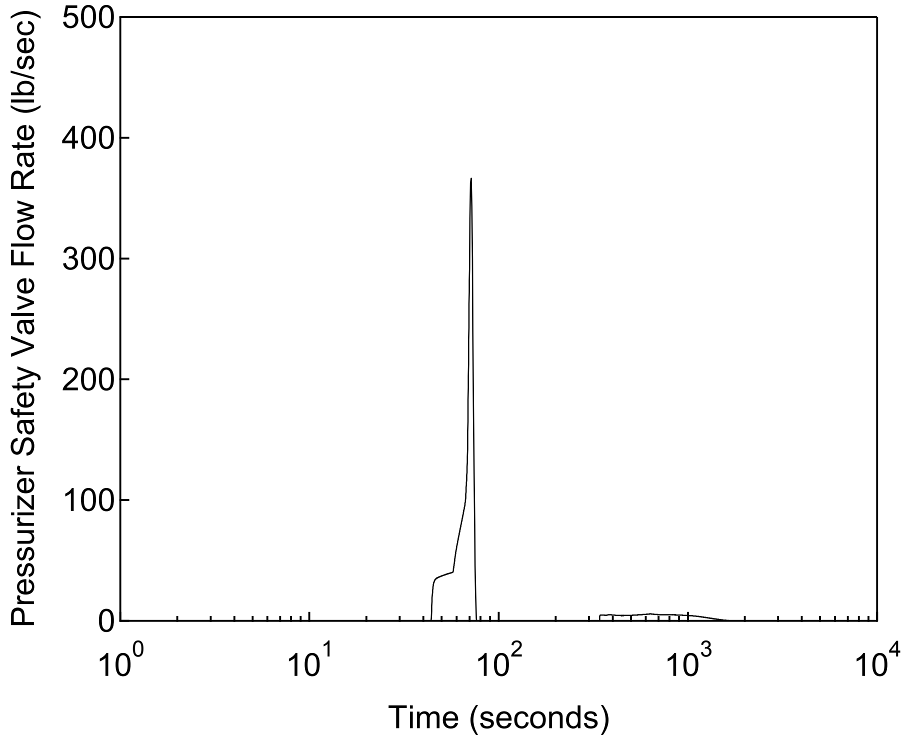


Figure 15.2.8-4 Pressurizer Safety Valve Flow Rate versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

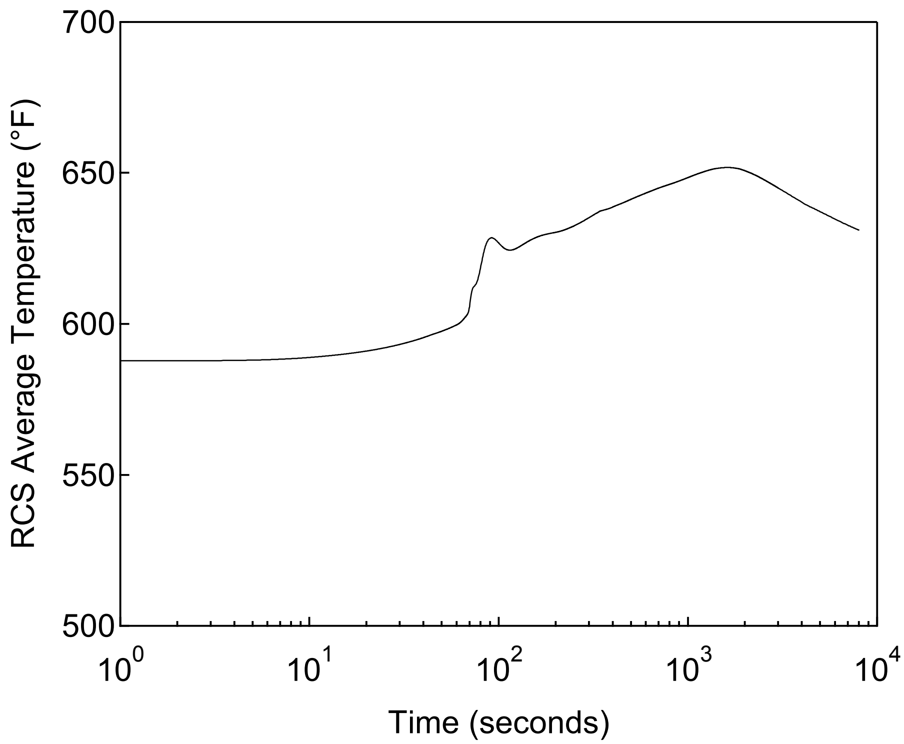


Figure 15.2.8-5 RCS Average Temperature versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

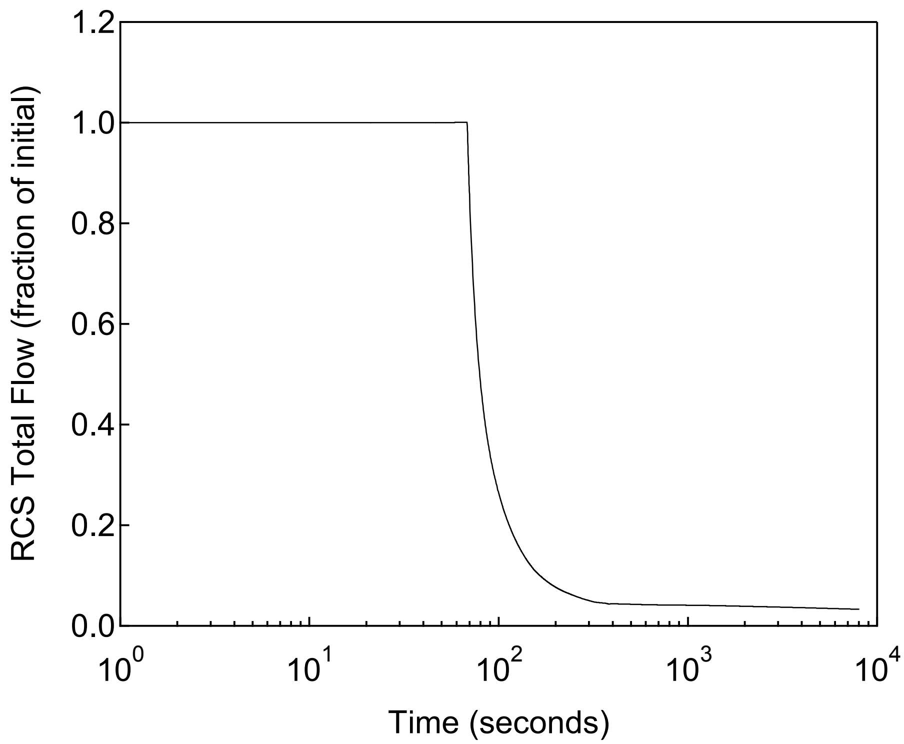


Figure 15.2.8-6 RCS Total Flow versus Time

Feedwater System Pipe Break - RCS Pressure Analysis

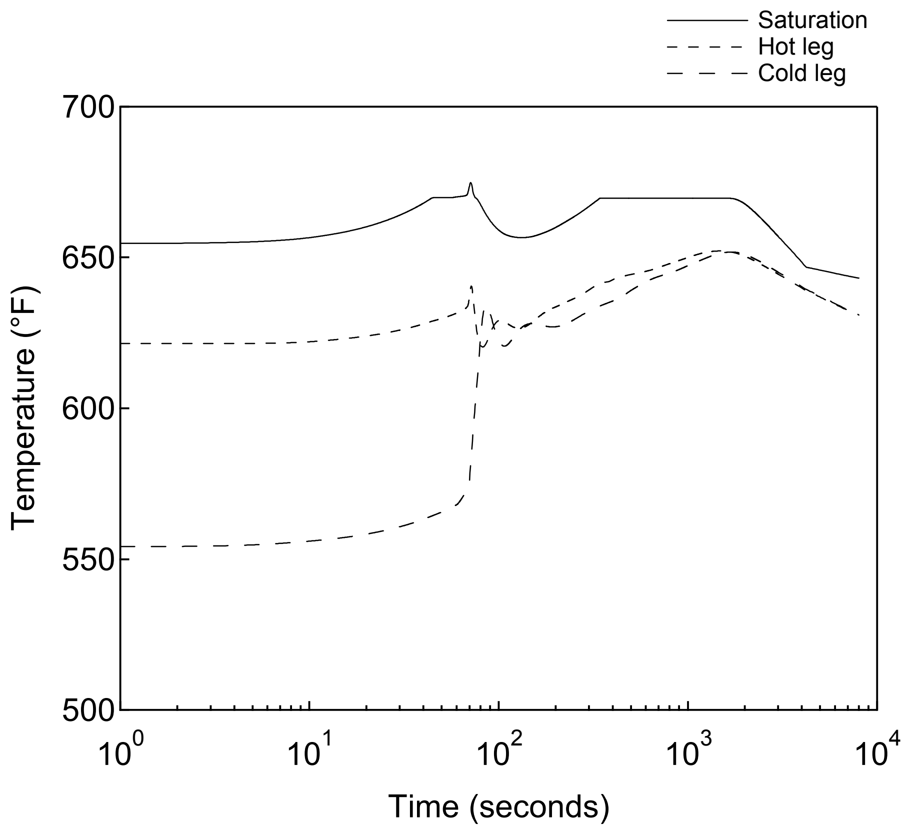


Figure 15.2.8-7 Temperature of Faulted Loop versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

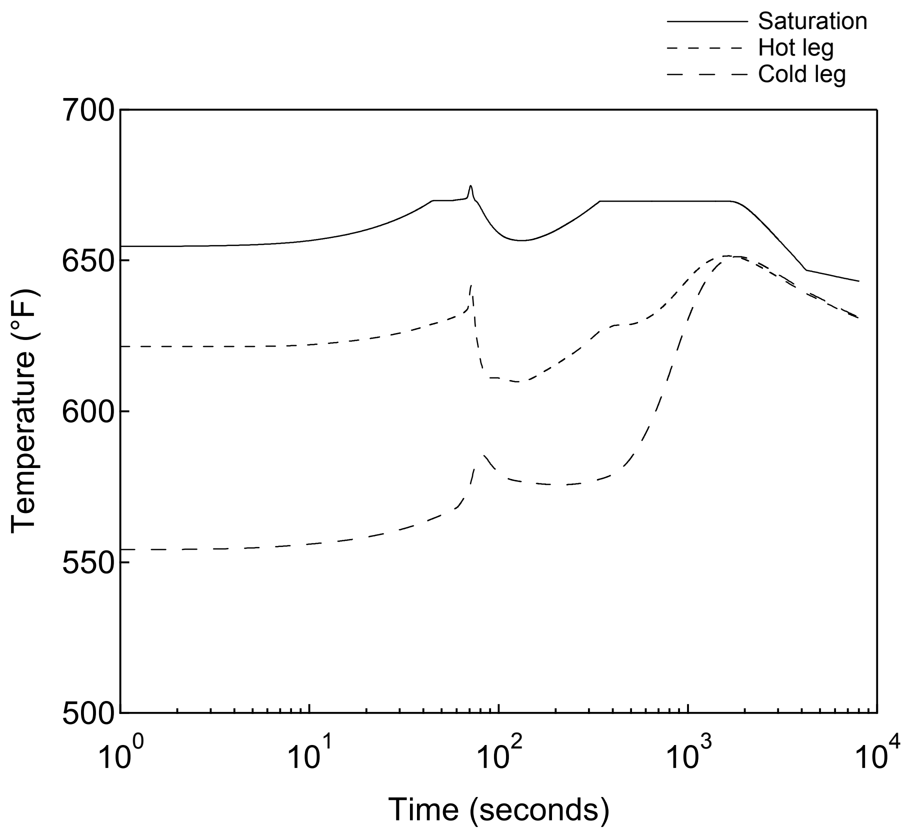


Figure 15.2.8-8 Temperature of Intact Loop without EFW versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

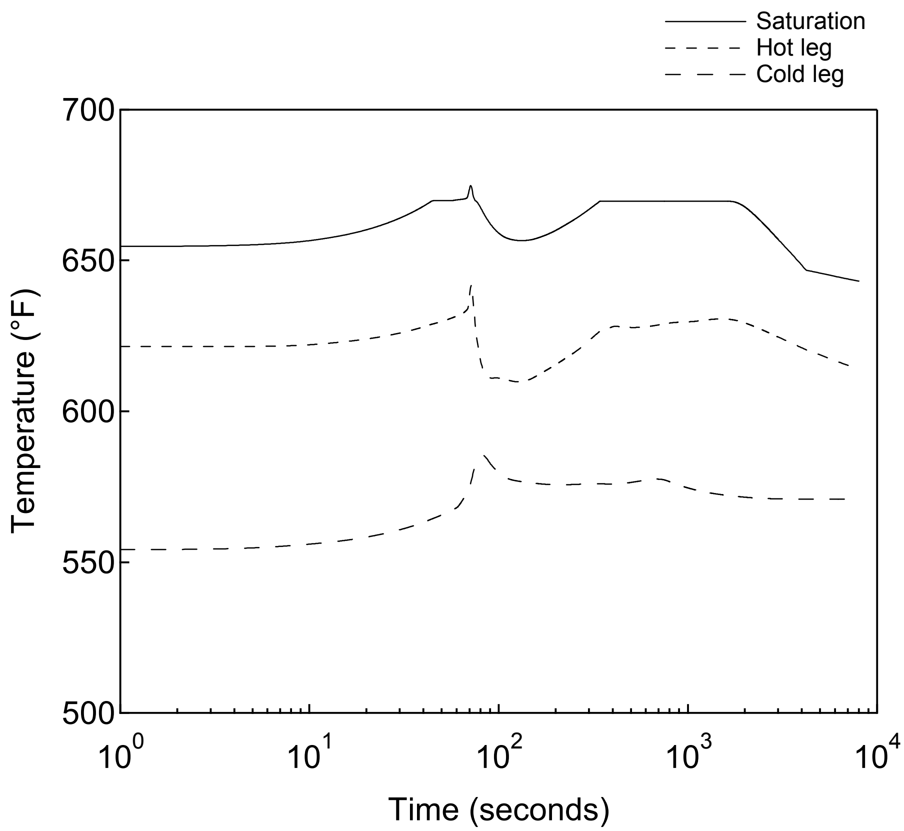


Figure 15.2.8-9 Temperature of Intact Loop with EFW versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

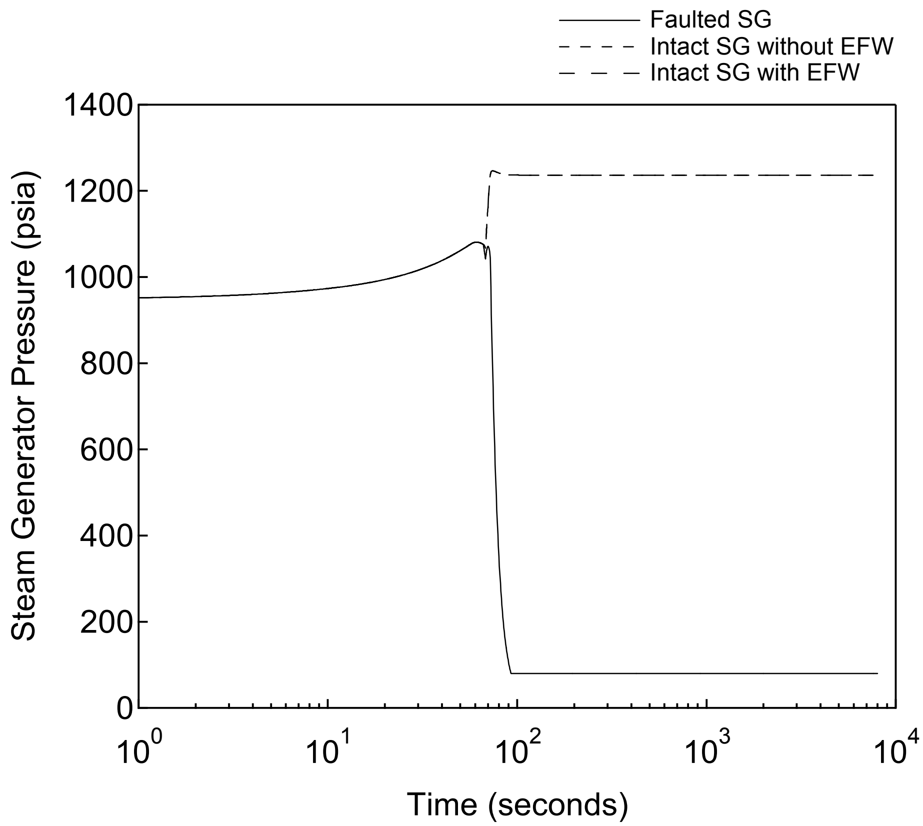


Figure 15.2.8-10 Steam Generator Pressure versus Time

Feedwater System Pipe Break - RCS Pressure Analysis

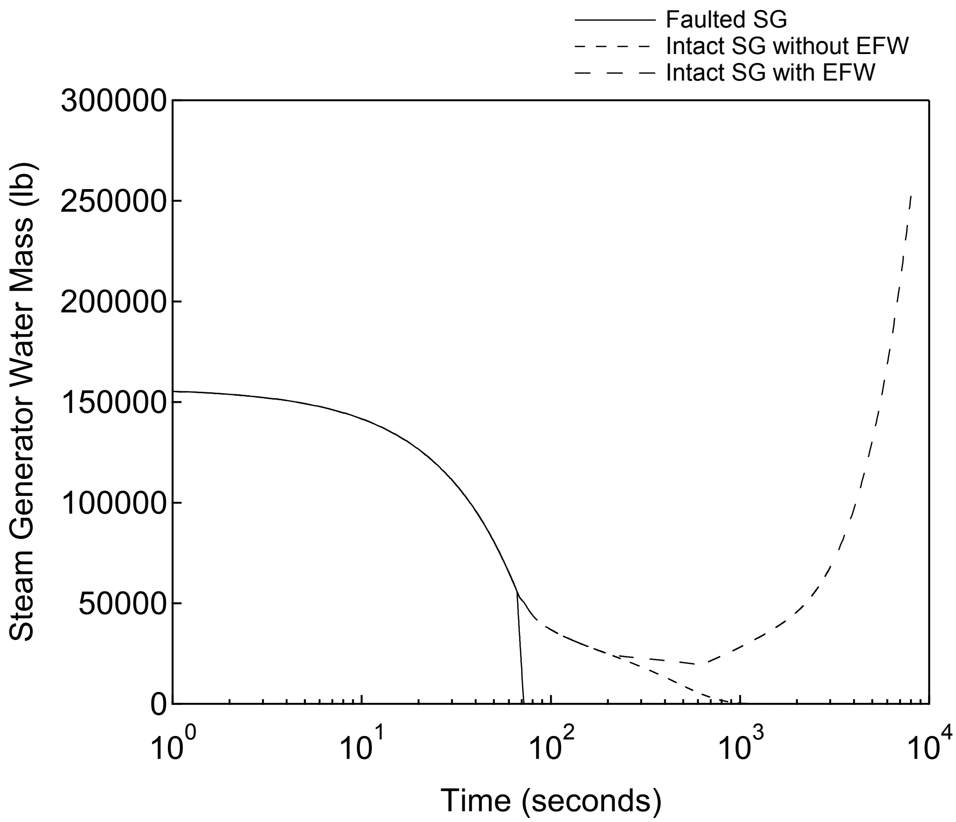


Figure 15.2.8-11 Steam Generator Water Mass versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

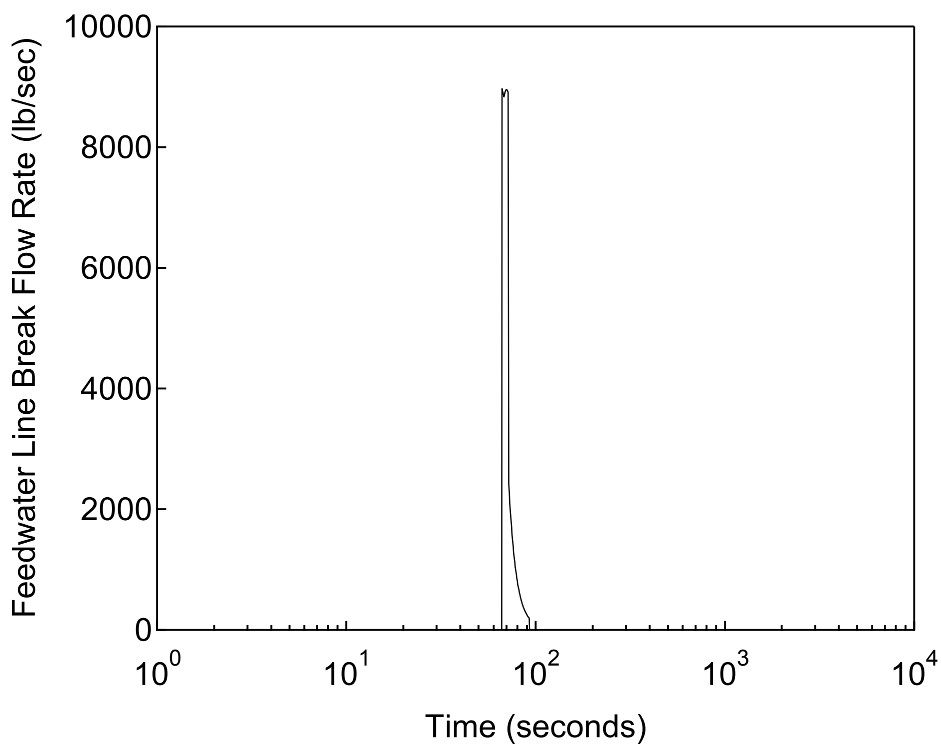


Figure 15.2.8-12 Feedwater Line Break Flow Rate versus Time

Feedwater System Pipe Break
- RCS Pressure Analysis

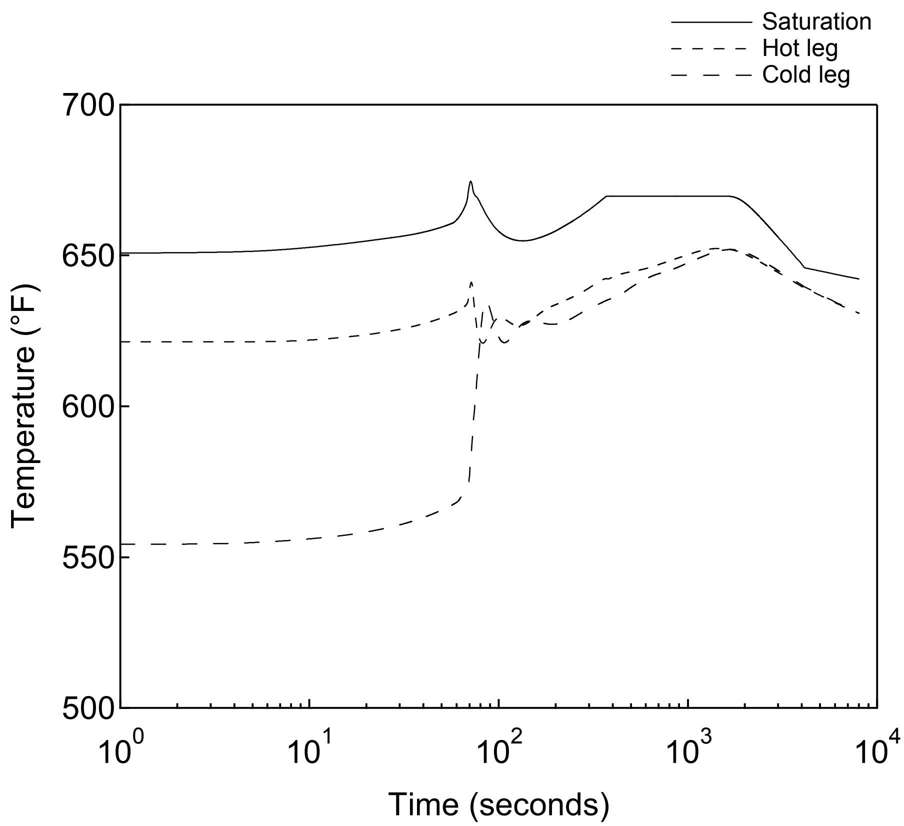


Figure 15.2.8-13 Temperature of Faulted Loop versus Time

Feedwater System Pipe Break
- Hot Leg Boiling Analysis

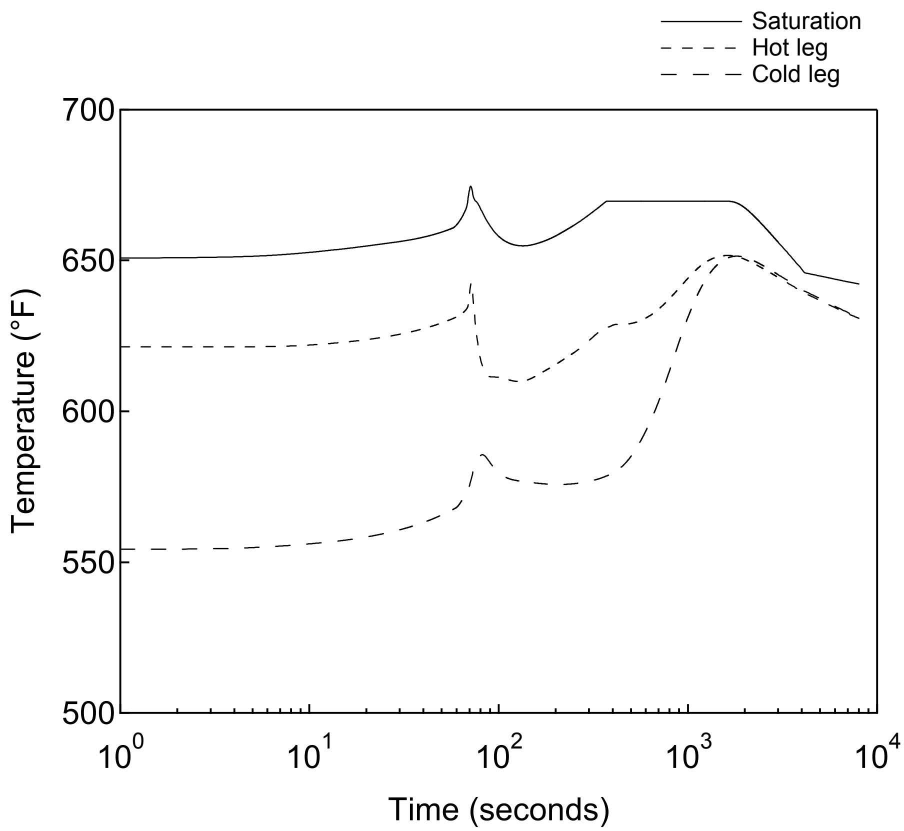


Figure 15.2.8-14 Temperature of Intact Loop without EFW versus Time

Feedwater System Pipe Break
- Hot Leg Boiling Analysis

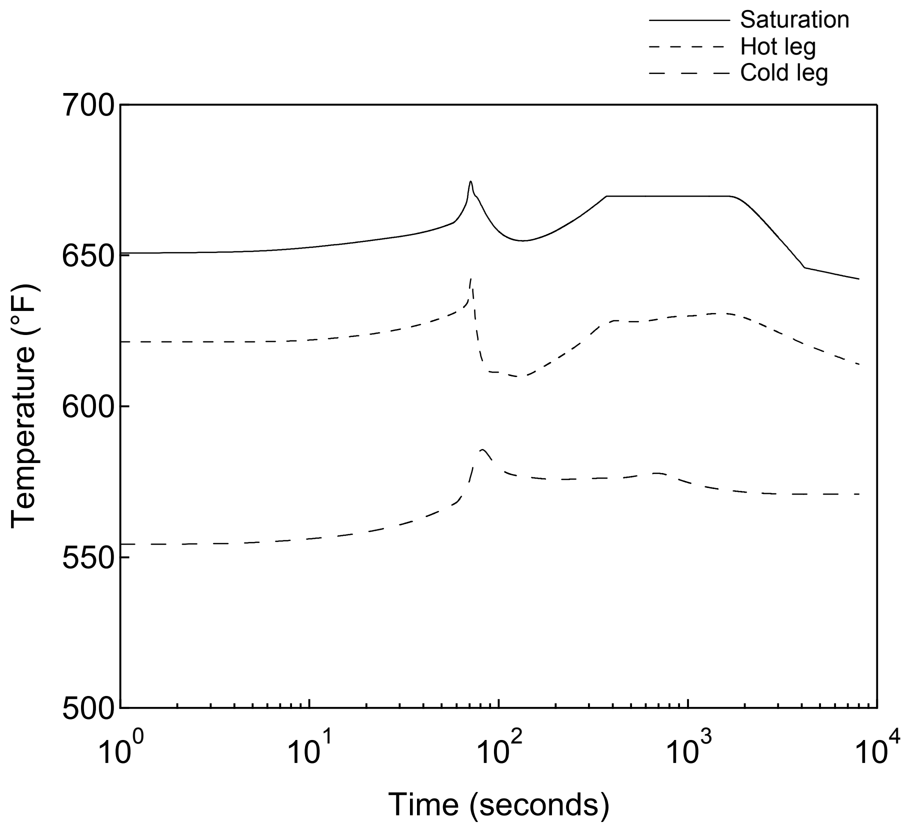


Figure 15.2.8-15 Temperature of Intact Loop with EFW versus Time

Feedwater System Pipe Break
- Hot Leg Boiling Analysis

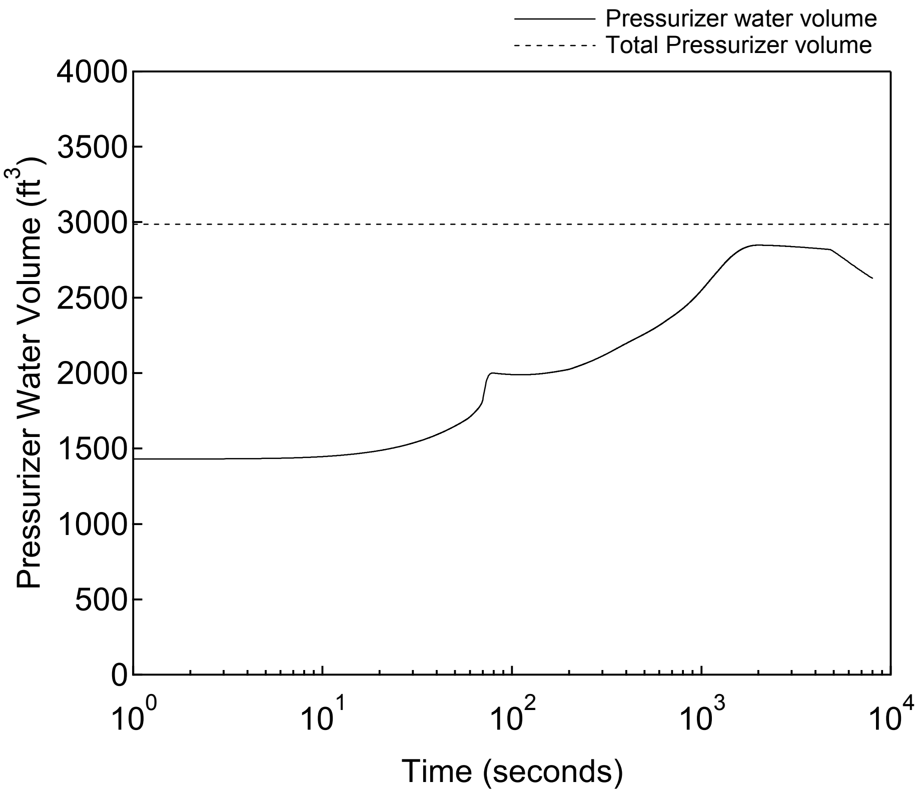


Figure 15.2.8-16 Pressurizer Water Volume versus Time

Feedwater System Pipe Break
- Pressurizer Water Volume Analysis

15.2.9 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.2.10 References

15.2-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.2-2 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.2-3 American Nuclear Society (ANS) 5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, Approved August 29, 1979.

15.2-4 Non-LOCA Methodology, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.3 Decrease in Reactor Coolant System Flow Rate

This section describes analyses that have been performed for events that could result in a decrease in reactor coolant system (RCS) flow rate, which, in turn, can lead to an increase in the primary coolant temperature.

Analyses of the following events are described in this section:

- Partial Loss of Forced Reactor Coolant Flow
- Complete Loss of Forced Reactor Coolant Flow
- Flow Controller Malfunctions (not applicable to US-APWR)
- Reactor Coolant Pump Rotor Seizure (Locked Rotor)
- Reactor Coolant Pump Shaft Break

These events are considered anticipated operational occurrences (AOOs) as defined in Section 15.0.0.1, except for the reactor coolant pump rotor seizure (locked rotor) and shaft break events which are classified as postulated accidents (PAs).

The results of these analyses determined that the reactor coolant pump (RCP) rotor seizure transient has the most severe radiological consequences of the transients listed above. Radiological consequences are reported in this section for only the RCP rotor seizure transient.

15.3.1 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor

15.3.1.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1.1 Identification of Causes and Frequency Classification

Loss of forced reactor coolant flow events can result from a mechanical or electrical failure in one or more RCPs or from a fault in the power supply to the pump motor. A partial loss of forced reactor coolant flow accident results from a simultaneous loss of electrical supply to one or more of the four reactor coolant pump motors. If the reactor is at power at the time of the transient, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature and a decrease in minimum DNBR. This transient is terminated by the low reactor coolant flow trip, which prevents DNB occurrence.

The trip of a single RCP and the simultaneous (or near simultaneous) trip of multiple RCPs are classified as anticipated operational occurrences (AOOs). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.3-1). Event frequency conditions are described in Section 15.0.0.1.

15.3.1.1.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the partial loss of flow event is described in the results section.

Stoppage of one, two, or three of the RCPs during power operation causes a partial loss of reactor coolant flow, which causes a reduction in the core cooling capabilities. This, in turn, can cause an increase in both the reactor fuel temperature and coolant temperature.

The individual RCPs are connected to separate plant buses so that a failure in one plant bus does not cause two or more pumps to stop at the same time. During reactor operation, the buses are supplied with power from the generator. If the power from the generator is cut off, the buses are supplied with power from an offsite transmission line.

Although the US-APWR is configured such that each RCP has its own source of electrical power, the partial loss of flow analysis conservatively assumes that two pumps trip at the same time in order to conservatively bound the case where the pumps might be installed such that two pumps share a common source of electrical power.

The RCPs are equipped with flywheels. The inertia of these flywheels prevents extremely rapid reductions in reactor coolant flow (and heat removal capacity) in the event of an RCP trip.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could cause a loss of offsite power, which could, in turn, cause an RCP coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection for this event:

- Low reactor coolant pump speed (two or more loops)
- Low reactor coolant flow (any one loop)

In this analysis, the transient is terminated by the low reactor coolant flow trip to prevent DNB occurrence. The evaluation of long term cooling is not performed for this event since reactor trip occurs immediately (discussed in Section 15.0.0.8).

The availability and adequacy of instrumentation and control is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event as discussed in Section 15.0.0.5.

15.3.1.1.3 Core and System Performance

15.3.1.1.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of various parameters following a loss of coolant flow. The model simulates the RCS including the RCS piping, RCPs, reactor vessel, core, pressurizer and surge line, the steam generator primary and secondary sides, control and protection systems, as well as pressurizer safety valves and main steam relief and safety valves. The MARVEL-M code includes a dynamic RCP and flow transient model that solves the fundamental flow transient equations based on a momentum balance around each reactor coolant loop and across the reactor vessel, flow continuity, and the RCP characteristics with or without electrical power to supply the pump motors. The multi-loop capability of the MARVEL-M code allows assuming each of the loops behaves independently, allowing the analysis of the partial loss of flow event. Although the analysis of this event is terminated shortly after the reactor trip, the model demonstrates the establishment of reverse flow that bypasses the core in the loops with RCP coastdowns. The MARVEL-M code generates an interface file that includes the time-dependent histories of the reactor power and core inlet flow rate for use in the VIPRE-01M code.

The VIPRE-01M code (Ref. 15.3-2) calculates the minimum DNBR during the transient using this interface as a boundary condition assuming a constant design power distribution. The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

Additional details concerning this methodology, including the RCP model and model validation, are provided in Reference 15.3-1. The MARVEL-M and VIPRE-01M codes are described in Section 15.0.2.2.1 and 15.0.2.2.2, respectively.

A single bounding case assuming the trip of two out of four RCPs is evaluated.

15.3.1.1.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for a partial loss of forced reactor coolant flow event.

- Consistent with use of the RTDP, the assumed initial values of reactor power, reactor coolant average temperature, reactor coolant flow rate, and RCS pressure are to be the nominal values without uncertainties provided in Table 15.0-3.

- Inlet coolant temperature and RCS pressure are held constant at conservative values for the DNBR calculations.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to have the maximum feedback limit shown in Figure 15.0-2. Use of these values maximizes the heat flux in the initial stage of the transient. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. Rod cluster control assembly insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- Conservative axial power profile and radial power distributions are assumed in the analysis as described in Section 15.0.0.2.3.

- The reactor is assumed to be automatically tripped by the low reactor coolant flow signal when the flow rate drops to 87% of rated flow for any reactor coolant loop. Table 15.0-4 summarizes the trip setpoint and signal delay time assumed in the analysis.

- The moment of inertia of the RCP, which is used to calculate the reactor coolant flow coast-down curve, is conservatively assumed to be 90% of the design inertia .

15.3.1.1.3.3 Results

Table 15.3.1.1-1 lists the key events and the times at which they occur, relative to the initiation of the partial loss of flow due to the trip of two RCPs.

Figures 15.3.1.1-1 through 15.3.1.1-6 are plots of key system parameters versus time from the core and system performance evaluation for this event.

The reduced flow conditions due to the tripped pumps are partially offset by the inertia of the two tripped RCPs as shown in Figure 15.3.1.1-1. The remaining two reactor coolant pumps continue running as long as offsite power remains available. This means that the RCS heat removal capacity does not decrease suddenly. The reactor coolant flow remains sufficiently high until the reactor is tripped, which causes power (shown in Figure 15.3.1.1-2) to drop more rapidly than the flow is dropping. As a result, the minimum DNBR (shown in Figure 15.3.1.1-6) remains above the 95/95 limit, and fuel integrity is not degraded.

The normalized core total flow transient is identical to the normalized RCS total flow. For consistency with the MARVEL-M output, the normalized RCS total flow is provided instead of the normalized core flow. Similarly, the average RCS temperature is provided instead of the core average temperature. Inlet coolant temperature does not change during the short duration of the event and is assumed constant in the DNBR calculations; therefore a plot of this parameter is not provided. Additionally, the hot channel heat flux is more closely related to DNBR; therefore, this parameter is provided instead of the average channel heat flux. Because there is significant core subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided; these are not key parameters for this event. Pressurizer safety valve flow is not reported for this event because RCS pressure remains below the safety valve set pressure. A plot of steam line pressure is not provided because it is bounded by the reactor coolant pump rotor seizure event discussed in Section 15.3.3.

15.3.1.1.4 Barrier Performance

The partial loss of forced reactor coolant flow is initiated by malfunctions that cause the loss of electrical power to single or multiple RCPs during power operation, resulting in a reduction in core cooling capabilities. Although the reduction in core cooling capability could cause an increase in both the reactor fuel temperature and reactor coolant temperature, the DNB limit is the primary design concern due to the combination of core temperature and core flow decrease. Therefore, a separate case for barrier performance is not presented.

The partial loss of forced reactor coolant flow event does not result in exceeding any reactor coolant pressure boundary or containment volume fission product barrier design limits. The results of the Core and System Evaluation case demonstrate that the RCS pressure remains well below 110% of system design pressure. In addition, the main steam pressure cannot challenge the main steam system pressure design limit, as discussed in Section 15.3.1.1.3.3. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.3.1.1.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the reactor coolant pump rotor seizure evaluated in Section 15.3.3.5.

15.3.1.1.6 Conclusions

A partial loss of forced reactor coolant flow will cause a reduction in core cooling capability. However, the resulting transient, even under conservatively assumed conditions, does not cause the minimum DNBR to decrease below the 95/95 limit and, therefore, no fuel failures are predicted.

The RCS pressure and main steam pressure remain below 110% of their respective system design pressures, so the integrity of the reactor coolant pressure boundary and the integrity of the main steam system pressure boundary are maintained.

This event does not lead to a more serious fault condition.

Table 15.3.1.1-1
Time Sequence of Events for Partial Loss of Forced Reactor Coolant Flow

Event	Time (sec)
RCP coastdown begins	0.0
Low reactor coolant flow analytical limit reached	1.5
Reactor trip initiated (rod motion begins)	3.3
Minimum DNBR occurs	5.0

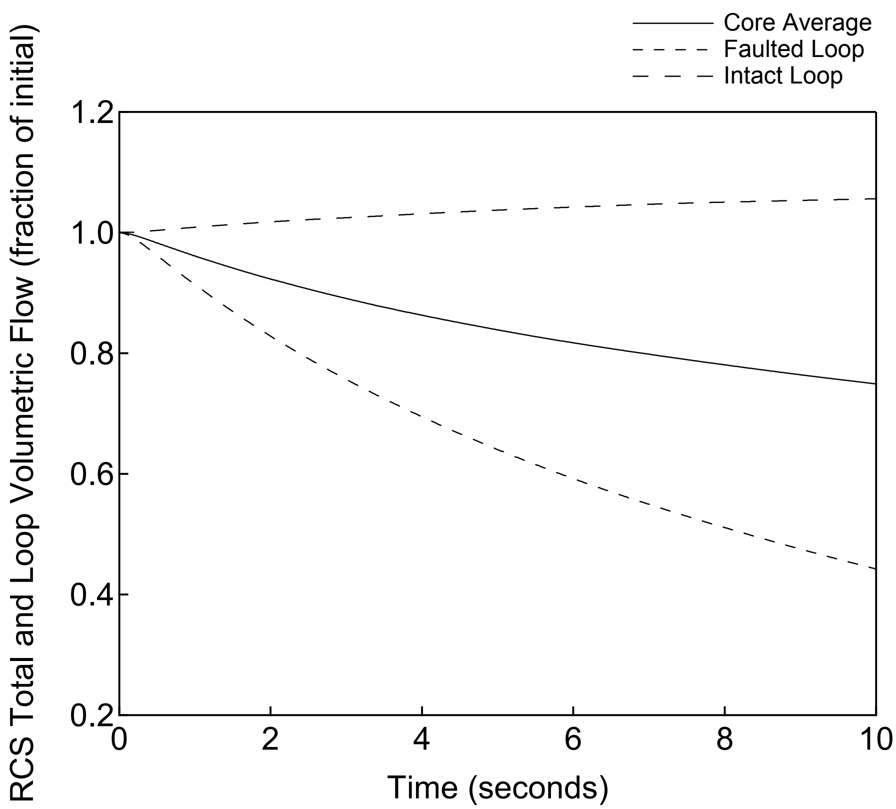


Figure 15.3.1.1-1 RCS Total and Loop Volumetric Flow versus Time

Partial Loss of Forced Reactor Coolant Flow

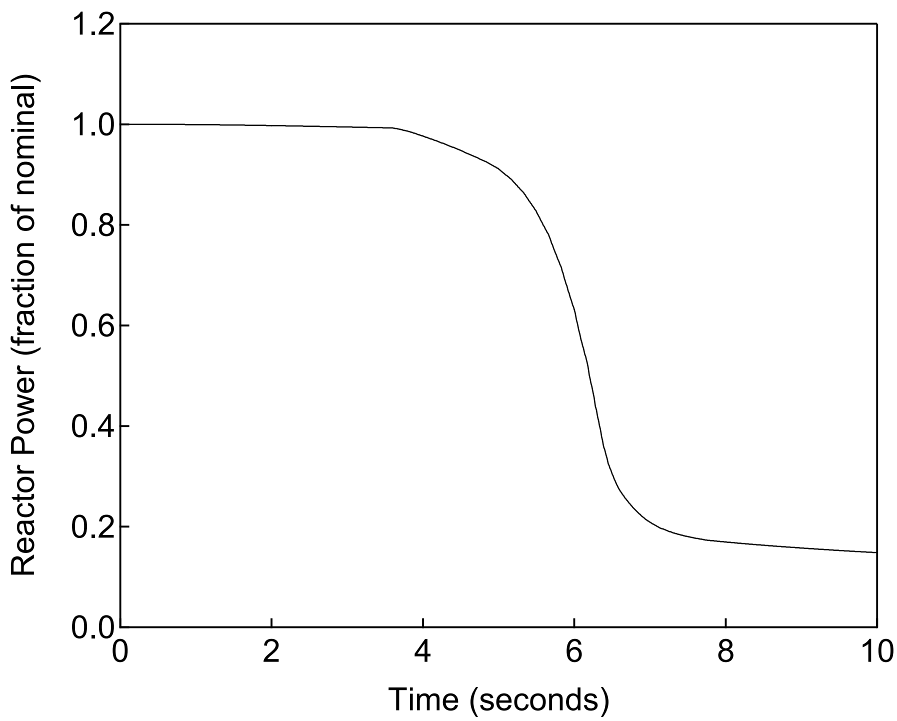


Figure 15.3.1.1-2 Reactor Power versus Time

Partial Loss of Forced Reactor Coolant Flow

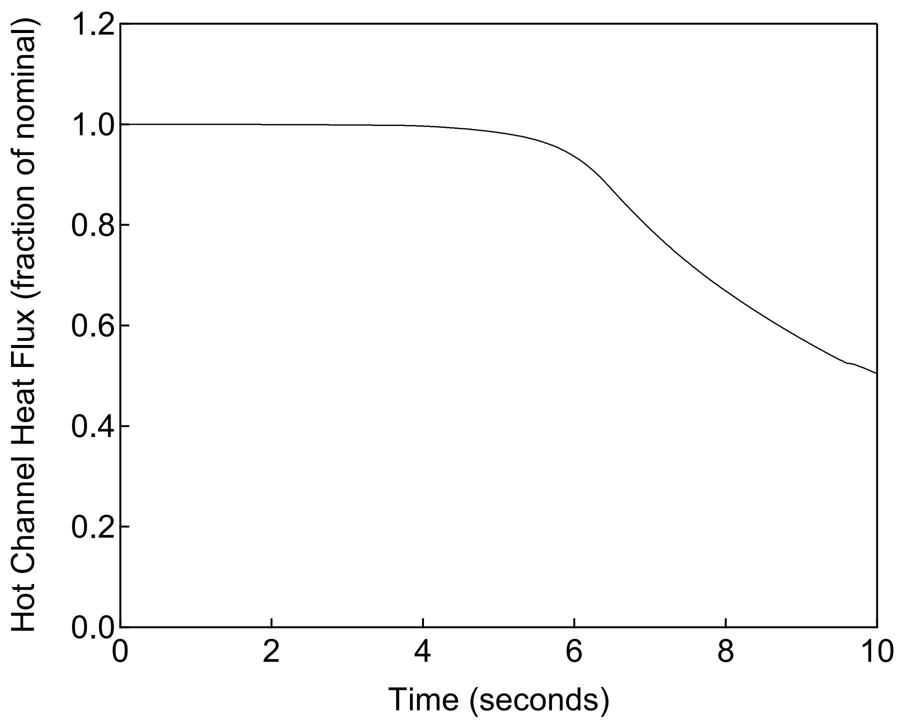


Figure 15.3.1.1-3 Hot Channel Heat Flux versus Time

Partial Loss of Forced Reactor Coolant Flow

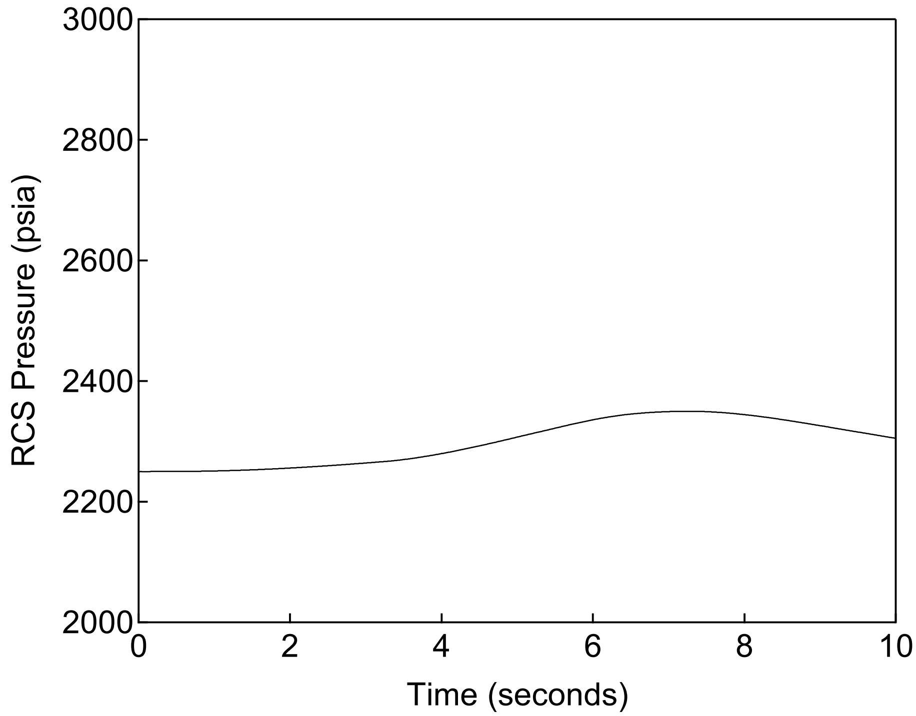


Figure 15.3.1.1-4 RCS Pressure versus Time

Partial Loss of Forced Reactor Coolant Flow

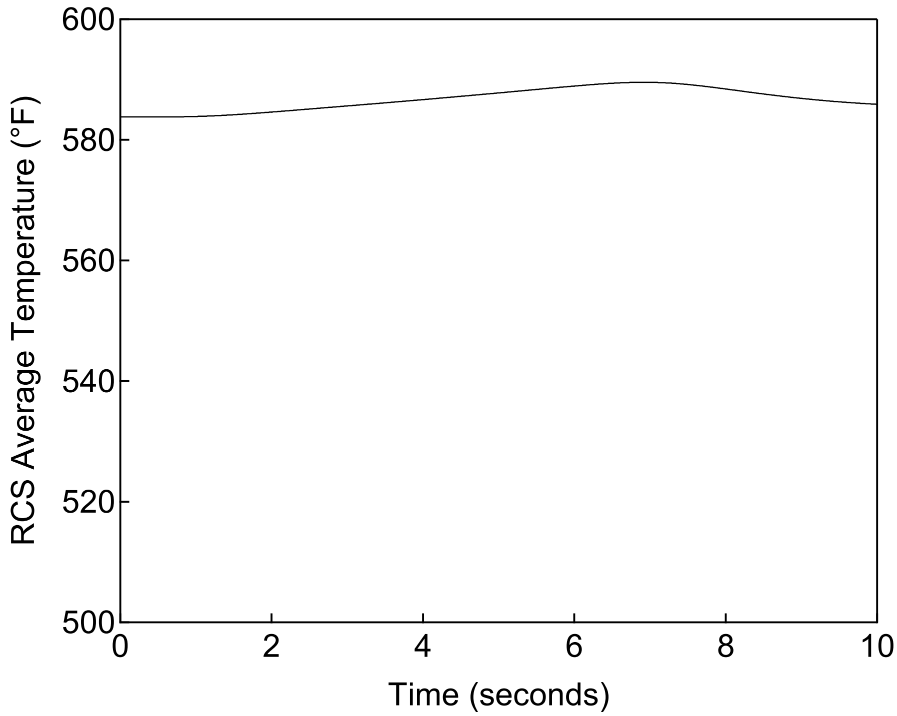


Figure 15.3.1.1-5 RCS Average Temperature versus Time

Partial Loss of Forced Reactor Coolant Flow

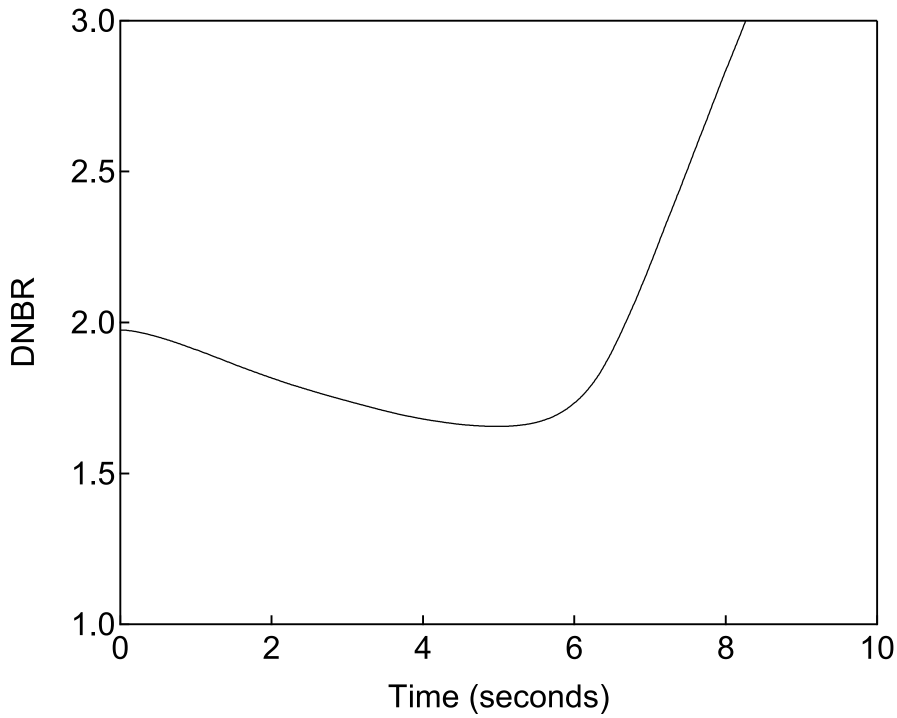


Figure 15.3.1.1-6 DNBR versus Time

Partial Loss of Forced Reactor Coolant Flow

15.3.1.2 Complete Loss of Forced Reactor Coolant Flow

15.3.1.2.1 Identification of Causes and Frequency Classification

The complete loss of forced reactor coolant flow is initiated by malfunctions that cause the loss of electrical power or the decrease of offsite power frequency to all four reactor coolant pumps during power operation, resulting in a reduction in the core cooling capabilities. If the reactor is at power at the time of the transient, the immediate effect of a complete loss of coolant flow is a rapid increase in coolant temperature and decrease in minimum DNBR. This transient is terminated by the low reactor coolant pump speed trip, which prevents DNB occurrence.

The simultaneous (or near simultaneous) trip of all four reactor coolant pumps (RCPs) is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition III infrequent event as defined in ANSI N18.2 (Ref. 15.3-3). Event frequency conditions are described in Section 15.0.0.1.

15.3.1.2.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the complete loss of flow event is described in the results section.

Stoppage or frequency reduction of all four RCPs during power operation causes a complete loss of reactor coolant flow, which causes a reduction in the core cooling capabilities. This, in turn, causes an increase in the reactor fuel temperature and in the reactor coolant temperature.

The RCPs are equipped with flywheels. The inertia of these flywheels prevents extremely rapid reductions in reactor coolant flow (and heat removal capacity) in the event of an RCP trip. In the frequency decay case, flywheel inertia is not available.

During reactor operation, the buses are supplied with power from the generator. If the power from the generator is cut off, the buses are supplied with power from an offsite transmission line.

This evaluation includes the effects of a postulated loss of offsite power because this event can be initiated by a loss of offsite power.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

The following signals are generated which would trip the reactor automatically through the RTS:

- Low reactor coolant pump speed
- Low reactor coolant flow

In this analysis, the transient is terminated by the low reactor coolant pump speed trip to prevent DNB occurrence. The evaluation of long term cooling is not performed for this event because the reactor trip occurs immediately (discussed in Section 15.0.0.8).

The availability and adequacy of instrumentation and control is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event as discussed in Section 15.0.0.5.

15.3.1.2.3 Core and System Performance

15.3.1.2.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses for various parameters following a loss of coolant flow. The model simulates the reactor coolant system including the reactor coolant system (RCS) piping, reactor coolant pumps, reactor vessel, core, pressurizer and surge line, the steam generator primary and secondary sides, control and protection systems, as well as pressurizer safety valves and main steam relief and safety valves. The MARVEL-M code includes a dynamic RCP and flow transient model that solves the fundamental flow transient equations based on a momentum balance around each reactor coolant loop and across the reactor vessel, flow continuity, and the RCP characteristics with or without electrical power to supply the pump motors. The multi-loop capability of the MARVEL-M code allows assuming each of the loops behaves independently, although this event is a uniform coastdown of all the reactor coolant pumps. The MARVEL-M code generates an interface file that includes the time-dependent histories of the reactor power and core inlet flow rate for use in the VIPRE-01M code.

The VIPRE-01M code (Ref. 15.3-2) calculates the minimum DNBR during the transient using this interface as a boundary condition assuming a constant design power distribution. The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

Additional details concerning this methodology, including the RCP model and model validation, are provided in Reference 15.3-1. The MARVEL-M and VIPRE-01M codes are described in Section 15.0.2.2.1 and 15.0.2.2.2, respectively.

Two cases are analyzed for core and system performance. One case evaluates the loss of power supply to all four RCPs resulting in a complete loss of forced reactor coolant flow, where the inertia of the RCP flywheels prevents extremely rapid reductions in reactor coolant flow (and heat removal capacity). The other case evaluates frequency decay of all four RCPs resulting in a complete loss of forced reactor coolant flow, where the RCP speed is forced to decrease at the same linear rate that the frequency decreases.

15.3.1.2.3.2 Input Parameters and Initial Conditions

The following bullets summarize the major input parameters and assumptions used in the loss of power supply case.

- Consistent with use of the RTDP, the assumed initial values of reactor power, reactor coolant average temperature, reactor coolant flow rate, and RCS pressure are the nominal values without uncertainties provided in Table 15.0-3.

- Inlet temperature and RCS pressure are held constant at conservative values for the DNBR calculations.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to have the maximum feedback limit shown in Figure 15.0-2. Use of these values maximizes the heat flux in the initial stage of the transient. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, reactor trip system signal processing delays) are used in the analysis. Rod cluster control assembly insertion characteristics assumed in analysis is described in Section 15.0.0.2.5.

- Conservative axial power profile and radial power distributions are assumed in the analysis as described in Section 15.0.0.2.3.

- The reactor is assumed to be automatically tripped by the low reactor coolant pump speed signal when the speed drops below 95% of its rated value for any reactor coolant loop. Table 15.0-4 summarizes the trip setpoint and signal delay time assumed in the analysis.

- The moment of inertia of the RCP, which is used to calculate the reactor coolant flow coast-down curve, is conservatively assumed to be 90% of the design inertia of the flywheel.

The frequency decay case is similar to the loss of power supply case with the following differences:

- The frequency decay rate is assumed to be 5 Hz/s, which is expected to bound site-specific characteristics.

- The moment of inertia of the RCP is not applicable because the RCP speed is forced to decrease at the same rate that the frequency decreases.

15.3.1.2.3.3 Results

Table 15.3.1.2-1 and Table 15.3.1.2-2 list the key events and the times at which they occur relative to the initiation of the loss of power supply and the frequency decay, respectively.

The transient responses for key parameters are presented in Figures 15.3.1.2-1 through 15.3.1.2-12. The first six figures provide the results for the case that evaluates the loss of power to all four RCPs, while the second set of six figures provides similar results for the frequency decay case.

The loss of power supply transient is initiated by a trip of all four RCPs, while the frequency decay transient is initiated by a decrease of offsite power frequency. As the pumps slow down, a reactor trip signal is generated by the low RCP speed trip in both cases. As shown in Figures 15.3.1.2-1 and 15.3.1.2-7, the rate of change in the flow is less severe for the loss of power supply case due to the inertia of the pump flywheels. In both cases, the power increases and the flow decreases prior to the reactor trip, resulting in a decrease in DNBR. The minimum DNBR occurs shortly after the reactor trip following the sharp decrease in power, as shown in Figures 15.3.1.2-2 and 15.3.1.2-6 for the loss of power supply case and Figures 15.3.1.2-8 and 15.3.1.2-12 in the frequency decay case. Although the minimum DNBR is lower in the frequency decay case, it remains above the 95/95 DNBR design limit for transients using the RTDP.

The normalized core total flow transient is identical to the normalized RCS total flow. For consistency with the MARVEL-M output, the normalized RCS total flow is provided instead of the normalized core flow. Similarly, the average RCS temperature is provided instead of the core average temperature. Inlet coolant temperature does not change during the short duration of the event and is assumed constant in the DNBR calculations; therefore a plot of this parameter is not provided. Additionally, the hot channel heat flux is more closely related to the DNBR; therefore, this parameter is provided instead of the average channel heat flux. Because there is significant core subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided; these are not key parameters for this event. Pressurizer safety valve flow is not reported for this event because RCS pressure remains below the safety valve set pressure. A plot of steam line pressure is not provided because it is bounded by the RCP rotor seizure event discussed in Section 15.3.3.

15.3.1.2.4 Barrier Performance

The complete loss of forced reactor coolant flow events are analyzed for effects on the fuel fission product barrier in Section 15.3.1.2.3. The long-term response of the pressurizer water level to a complete loss of forced reactor coolant flow is bounded by the loss of non-emergency alternating current power event analyzed in Section 15.2.6. Therefore, a separate case for barrier performance is not presented.

This event does not result in exceeding any reactor coolant pressure boundary or containment volume fission product barrier design limits. The results of the Core and System Evaluation case demonstrate that the RCS pressure remains well below 110% of system design pressure. In addition, the main steam pressure cannot challenge the main steam system pressure design limit, as discussed in Section 15.3.1.2.3.3. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.3.1.2.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the reactor coolant pump rotor seizure evaluated in Section 15.3.3.5.

15.3.1.2.6 Conclusions

A complete loss of forced reactor coolant flow will cause a reduction in core cooling capability. However, the resulting transient, even under conservatively assumed conditions, does not cause the minimum DNBR to decrease below the 95/95 limit and, therefore, no fuel failures are predicted.

Additionally, the peak pressures in the RCS and main steam system remain below 110% of their respective design pressures, so the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

This event does not lead to a more serious fault condition.

Table 15.3.1.2-1
Time Sequence of Events for Loss of Power Supply Resulting in a Complete Loss of Forced Reactor Coolant Flow

Event	Time (sec)
RCPs trip (flow coastdown begins)	0.0
Low reactor coolant pump speed analytical limit reached	0.5
Reactor trip initiated (rod motion begins)	1.1
Minimum DNBR occurs	3.5

Table 15.3.1.2-2
Time Sequence of Events for Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

Event	Time (sec)
Decrease of offsite power frequency begins	0.0
Low reactor coolant pump speed analytical limit reached	0.6
Reactor trip initiated (rod motion begins)	1.2
Minimum DNBR occurs	3.7

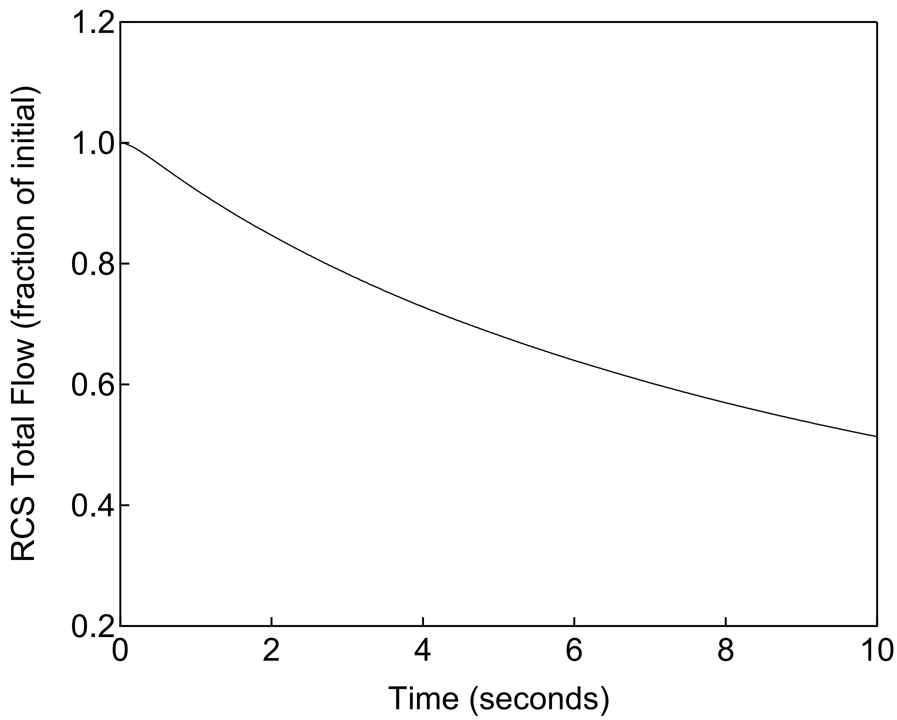


Figure 15.3.1.2-1 RCS Total Flow versus Time

Complete Loss of Forced Reactor Coolant Flow

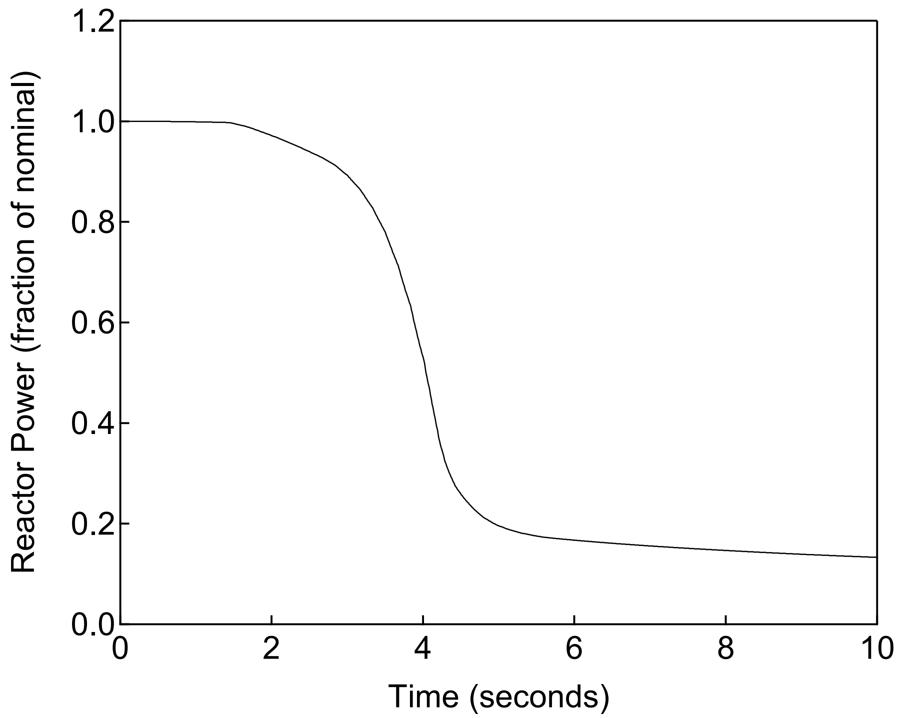


Figure 15.3.1.2-2 Reactor Power versus Time

Complete Loss of Forced Reactor Coolant Flow

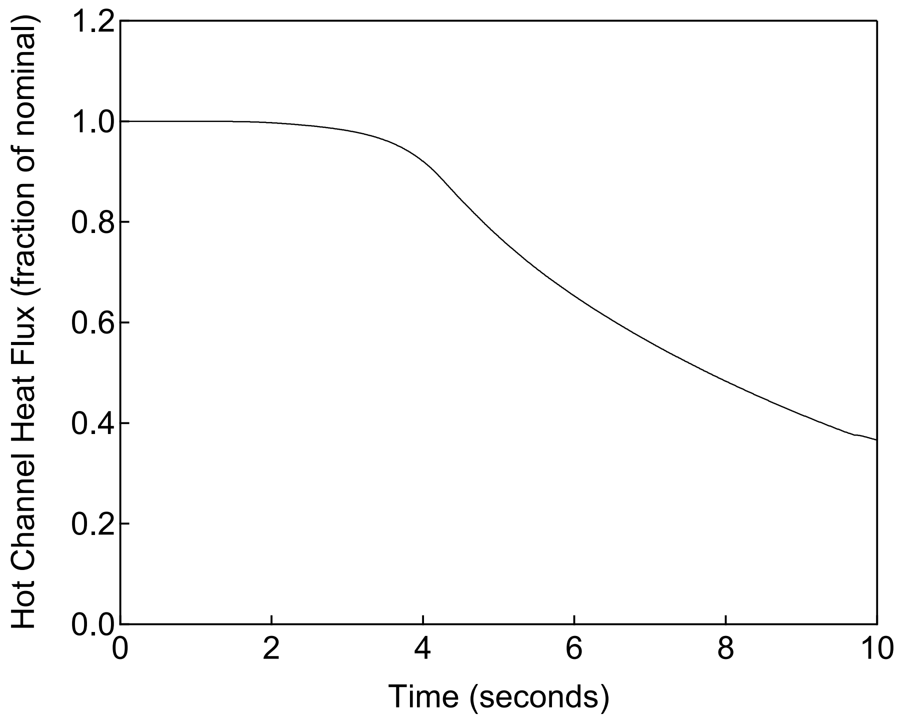


Figure 15.3.1.2-3 Hot Channel Heat Flux versus Time

Complete Loss of Forced Reactor Coolant Flow

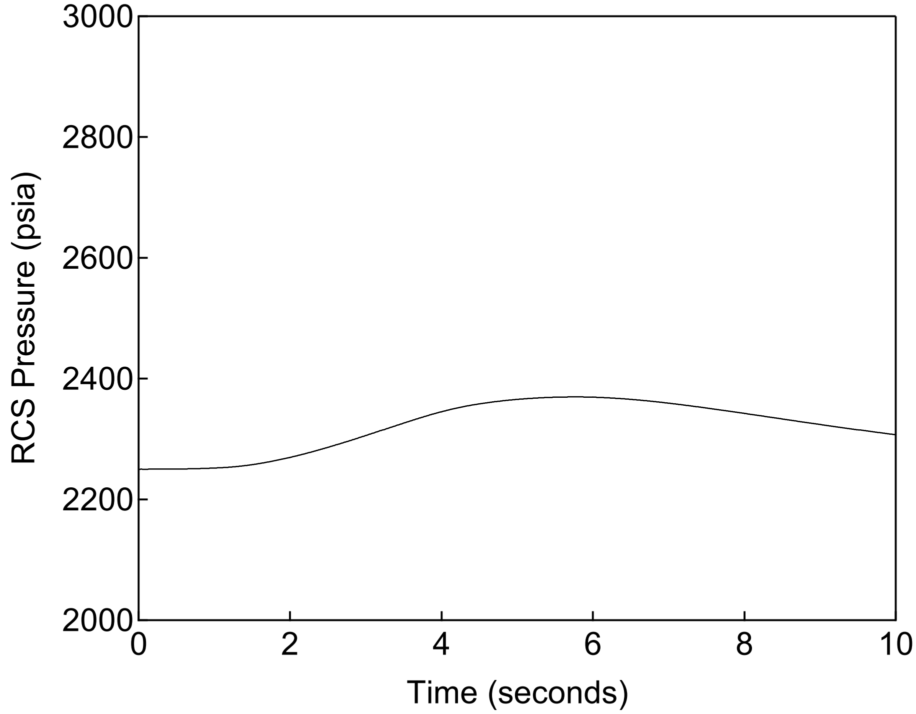


Figure 15.3.1.2-4 RCS Pressure versus Time

Complete Loss of Forced Reactor Coolant Flow

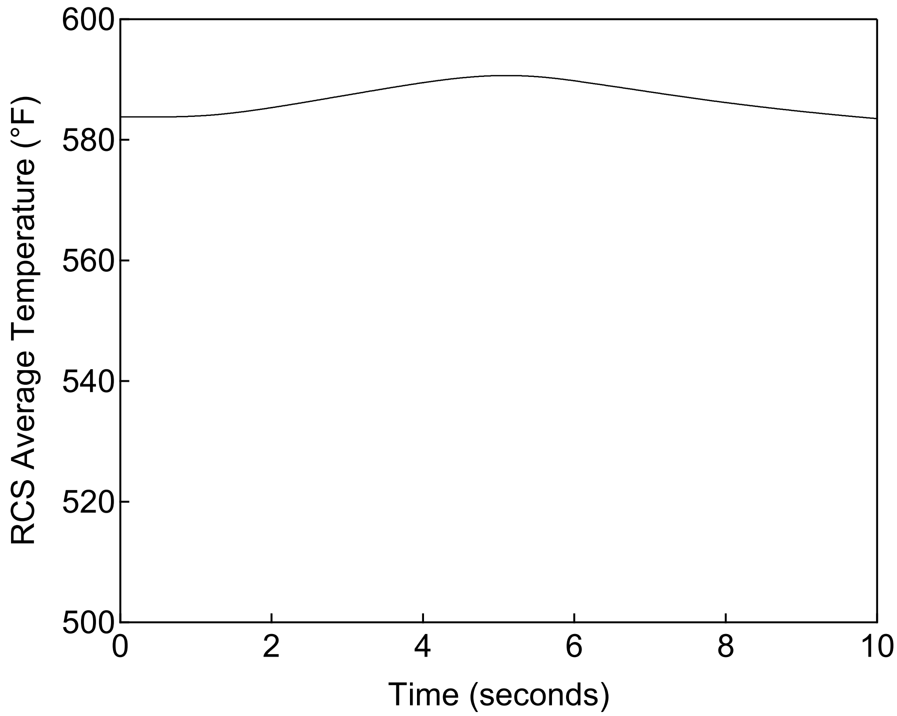


Figure 15.3.1.2-5 RCS Average Temperature versus Time

Complete Loss of Forced Reactor Coolant Flow

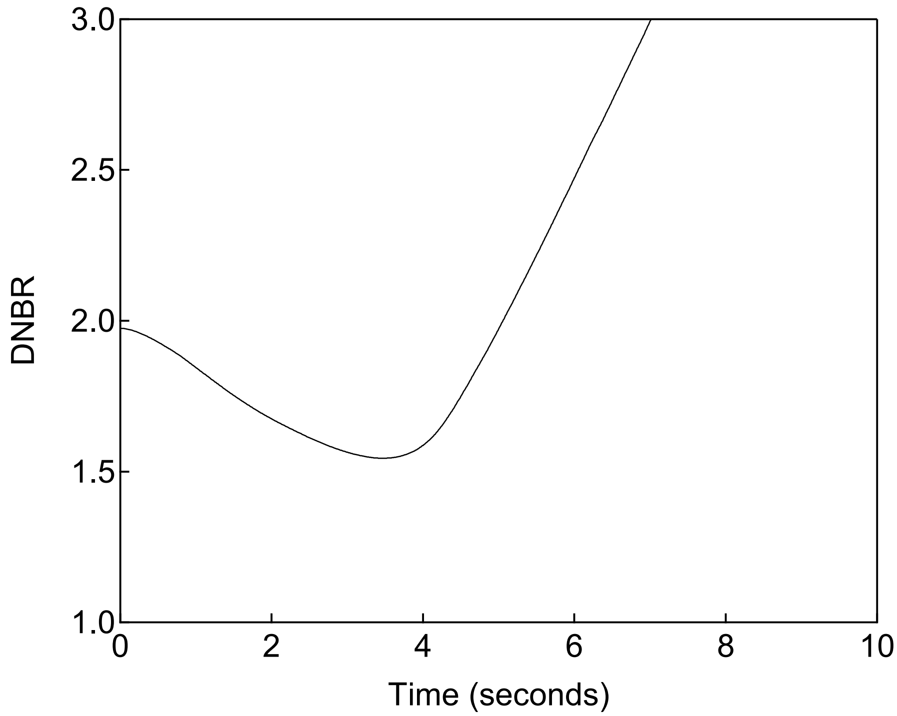


Figure 15.3.1.2-6 DNBR versus Time

Complete Loss of Forced Reactor Coolant Flow

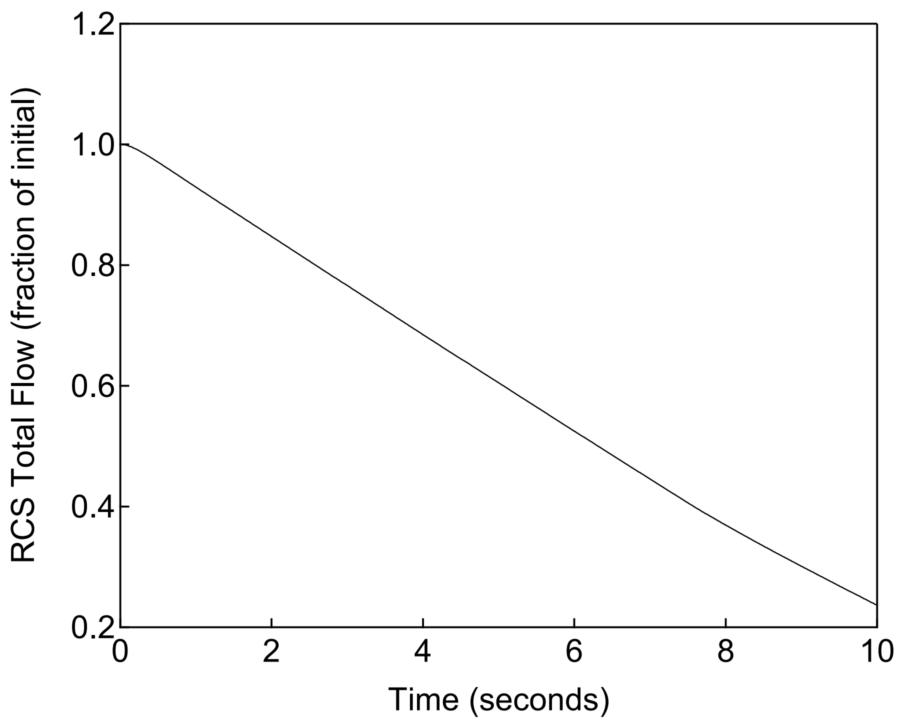


Figure 15.3.1.2-7 RCS Total Flow versus Time

Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

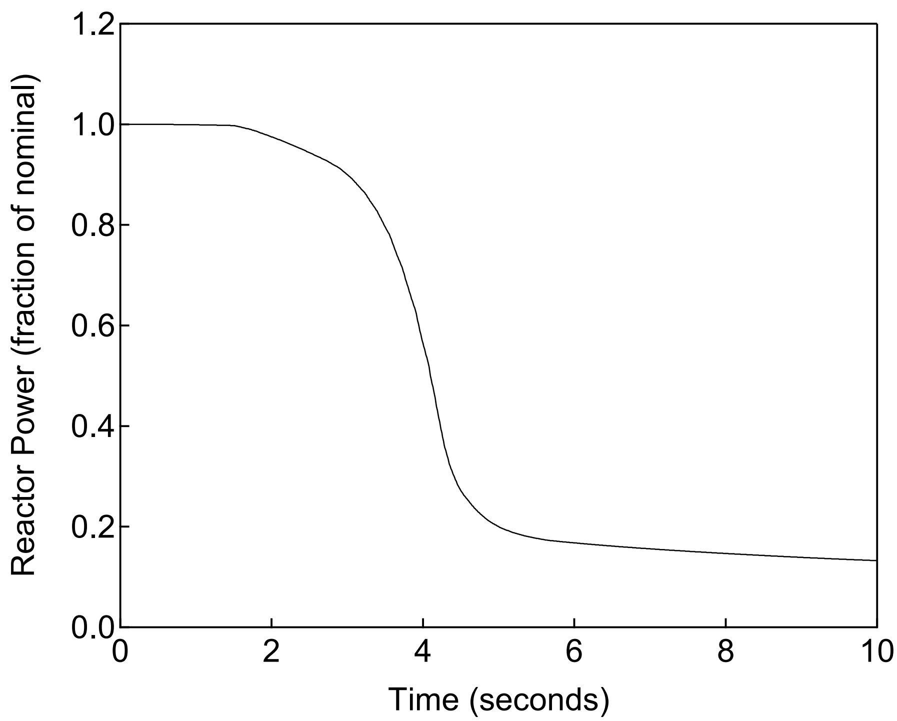


Figure 15.3.1.2-8 Reactor Power versus Time

Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

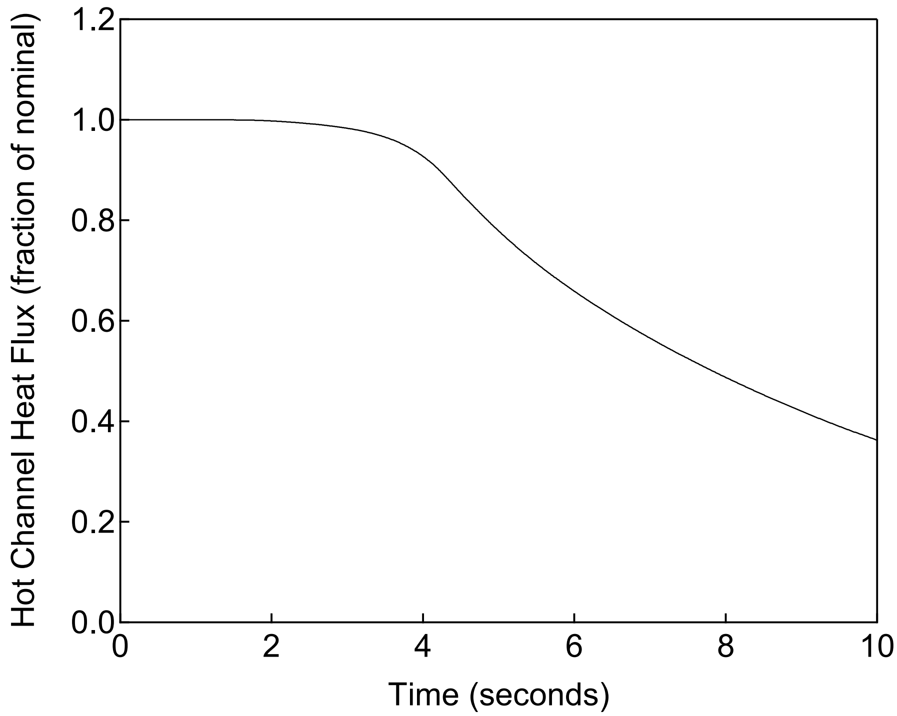


Figure 15.3.1.2-9 Hot Channel Heat Flux versus Time

Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

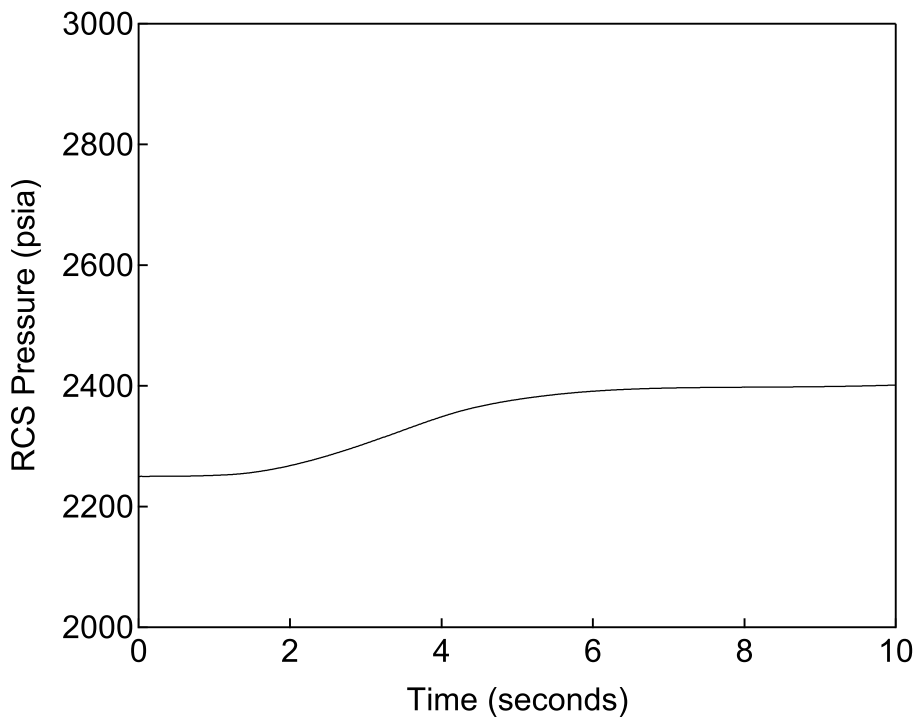


Figure 15.3.1.2-10 RCS Pressure versus Time

Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

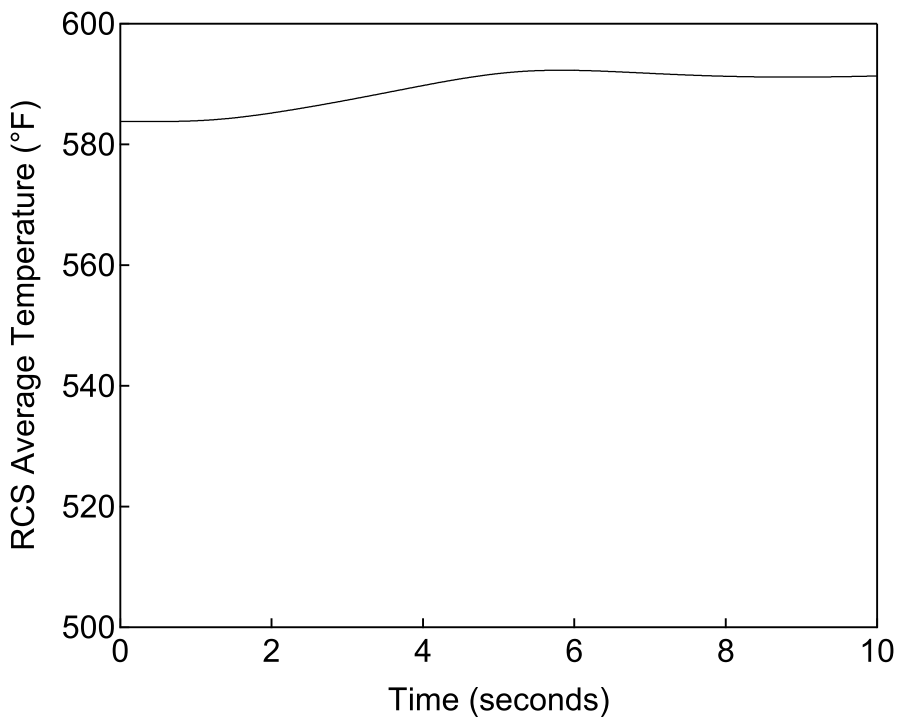


Figure 15.3.1.2-11 RCS Average Temperature versus Time

Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

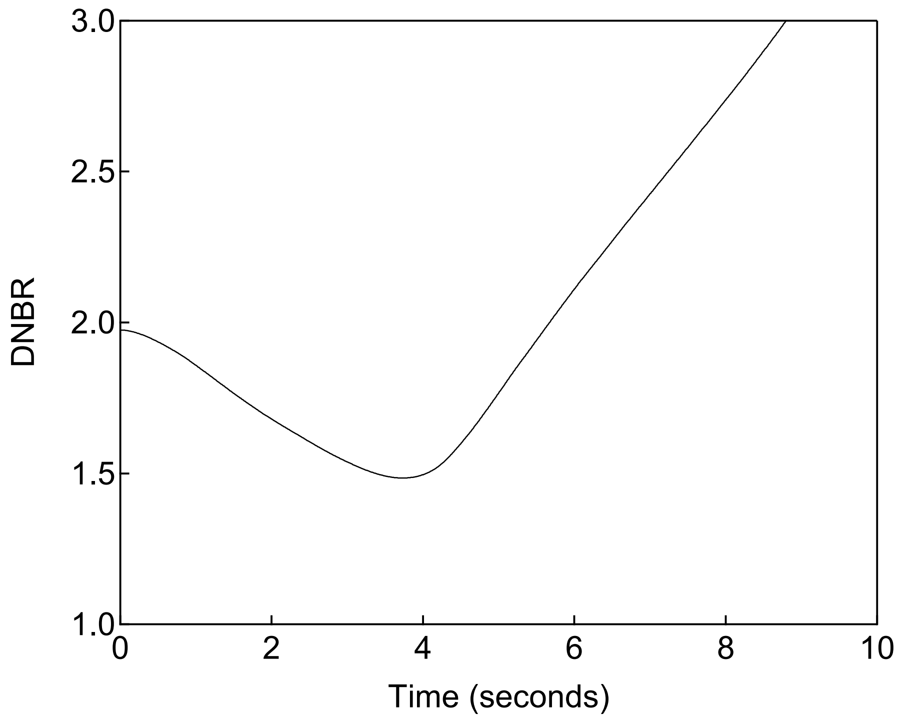


Figure 15.3.1.2-12 DNBR versus Time

Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

15.3.2 Flow Controller Malfunctions

This section is not applicable to the US-APWR, because it does not have reactor coolant system flow controllers.

15.3.3 Reactor Coolant Pump Rotor Seizure

15.3.3.1 Identification of Causes and Frequency Classification

This event is initiated by the instantaneous seizure of one reactor coolant pump (RCP) rotor during power operation. This postulated rotor seizure would cause a rapid reduction in the reactor coolant flow (compared to the coastdown associated with an RCP trip) resulting in a decrease in core cooling capacity. This could, in turn, lead to an increase in the reactor fuel temperature, primary coolant temperature, and reactor pressure. This event is sometimes referred to as a locked pump rotor transient.

Possible causes of a rotor seizure are bearing wear or bearing overheating due to loss of forced cooling or a coolant leak. However, the sudden stoppage of the RCP postulated in this scenario is more consistent with a failure affecting the rotating assembly, which results from a deformation that causes an interference with surrounding RCP components.

The seizure of an RCP rotor is a postulated accident (PA). Historically, this was classified as a Condition IV event in the older plant condition frequency grouping as defined in ANSI N18.2 (Ref. 15.3-3). Event frequency conditions are described in Section 15.0.0.1.

15.3.3.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the RCP rotor seizure (locked rotor) event is described in the results section.

This event is initiated by the instantaneous seizure of one RCP rotor during power operation. This postulated rotor seizure would cause a rapid reduction in the reactor coolant flow and decrease of the core cooling capacity. The flow is reversed in the loop with the locked pump rotor due to the pressure difference between the downcomer and reactor vessel outlet. This in turn leads to an increase in the reactor fuel temperature, primary coolant temperature, and reactor coolant system (RCS) pressure.

Selection of the material, design, manufacture, installation and inspection of the reactor coolant pump must conform to the relevant standards and criteria, and sufficient quality control must be provided. In particular, the bearings must be designed to withstand long-term operation of the reactor coolant pumps without being worn out, thereby minimizing the possibility of pump rotor seizure.

If the bearing lubricating oil level or bearing temperature is abnormal, an oil level low alarm or bearing temperature high control room alarm will result. In accordance with operating procedures, the affected RCP is manually tripped to prevent pump damage (including bearing seizure). The reactor will automatically trip when an RCP is tripped.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could cause a loss of offsite power, which could, in turn, cause an RCP coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

The following signals are assumed to be available to trip the reactor and therefore provide protection from this transient:

- Low reactor coolant flow
- Low reactor coolant pump speed

The reactor is assumed to be automatically tripped by the low reactor coolant flow signal.

The evaluation of long term cooling is not performed for this event since reactor trip occurs immediately (discussed in Section 15.0.0.8).

Emergency feedwater is not assumed to be actuated after all main feedwater pumps trip in this event (discussed in Section 7.3.1.1).

The pump seal cooling with containment isolation is described in Section 15.0.0.9.

The availability and adequacy of instrumentation and control is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event as discussed in Section 15.0.0.5.

Fracture toughness of the reactor coolant pressure boundary and reactor vessel is described in Sections 5.2.3 and 5.3.1.

15.3.3.3 Core and System Performance

Two core and system performance evaluations are performed, one to calculate peak cladding temperature and the second to calculate the number of rods that experience DNB failure as an input to the radiological consequence analysis.

15.3.3.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of various parameters following a reduction in coolant flow due to an RCP failure. The model simulates the RCS including the RCS piping, RCPs, reactor vessel, core, pressurizer and surge line, the steam generator primary and secondary sides, control and protection systems, as well as pressurizer safety valves and main steam relief and safety valves. The MARVEL-M code includes a dynamic RCP and flow transient model that solves the fundamental flow transient equations based on a momentum balance around each reactor coolant loop and across the reactor vessel, flow continuity, and the RCP characteristics with or without electrical power to supply the pump motors. The multi-loop capability of the MARVEL-M code allows assuming each of the loops behaves independently, and has the capability to model the impeller differently for locked rotor and sheared shaft assumptions. The MARVEL-M code generates an interface file that includes the time-dependent histories of the reactor power and core inlet flow rate for use in the VIPRE-01M code.

For DNB failure evaluation, the VIPRE-01M code (Ref. 15.3-2) calculates the minimum DNBR during the transient using this interface as a boundary condition assuming a constant design power distribution. The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation. For peak cladding temperature analysis, the VIPRE-01M code calculates the cladding temperature at the hot spot during the transient.

Additional details concerning this methodology, including the RCP model and model validation, are provided in Reference 15.3-1. The MARVEL-M and VIPRE-01M codes are described in Sections 15.0.2.2.1 and 15.0.2.2.2, respectively. Additional details on the MARVEL-M reactor coolant pump and flow models are provided in Reference 15.3-1.

A limiting case is defined for the locked rotor accident that also bounds the plant response to the reactor coolant pump shaft break event discussed in Section 15.3.4. In both accidents, the pump failure causes a rapid decrease in flow in the affected loop. Reverse flow is then established in the affected loop, which becomes a core bypass path for some of the flow entering the downcomer from the intact loops. The abrupt flow decrease at the beginning of the transient (before loop flow reversal occurs) results in lower core flow for the locked rotor case because the RCP has a higher flow resistance with the impeller locked. Conversely, the loop reverse flow (and total core flow reduction) is greater after flow reversal for the shaft break case since the impeller is free to spin inside the pump casing. The bounding case is defined by assuming the RCP rotor is stopped prior to flow reversal, and that the pump resistance is changed to zero after the flow reverses in the affected loop. The resulting total core flow, as calculated by MARVEL-M and used in the VIPRE-01M DNBR calculations, conservatively bounds the flow transients for both events. The discussion of the analysis beyond this point applies to the bounding case.

15.3.3.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate the rods in DNB for a RCP rotor seizure (locked rotor) event.

- Consistent with use of the RTDP, the assumed initial values of reactor power, reactor coolant average temperature, reactor coolant flow rate, and RCS pressure are to be the nominal values without uncertainties provided in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to have the maximum feedback limit shown in Figure 15.0-2. Use of these values maximizes the heat flux in the initial stage of the transient. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. rod cluster control assembly insertion characteristics assumed in analysis is described in Section 15.0.0.2.5.

- Conservative axial power profile and radial power distributions are assumed in the analysis as described in Section 15.0.0.2.3.

- The reactor is assumed to be automatically tripped by the low reactor coolant flow signal when the flow rate drops to 87% of rated flow for any reactor coolant loop. Table 15.0-4 summarizes the trip setpoint and signal delay time assumed in the analysis.

- The rod power at the hot spot is conservatively assumed to be 2.6 times the core average power (F_q = 2.6) at the initial power level.

The cladding temperature analysis is identical to the rods in DNB analysis with the following differences:

- The initial power level is taken as 102% of the licensed core thermal power level with initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi below the nominal value. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

- The gap heat transfer coefficient between the fuel pellet and fuel cladding is conservatively assumed to be 14,000 BTU/ft^2-h-°F (the maximum value) during the transient.

- The film boiling coefficient is calculated based on the Bishop-Sandberg-Tong film boiling heat transfer correlation equation. DNB is conservatively assumed to start at the beginning of the accident.

- When the fuel cladding temperature reaches about 1800°F, the zirconium-water reaction becomes significant. The Baker-Just formula is used for the calculation of the zirconium-water reaction rate as described in Section 6.6 of Ref. 15.3-2.

15.3.3.3.3 Results

This event is initiated by an instantaneous rotor seizure of one of the reactor coolant pumps during power operation. The bounding analysis case is defined by assuming the RCP rotor is stopped (locked) prior to flow reversal, and that the pump resistance is changed to zero (similar to shaft break) after the flow reverses in the affected loop.

Figures 15.3.3-1 through 15.3.3-4 are plots of system parameters versus time for the rods in DNB analysis of the bounding RCP rotor seizure/shaft break transient with offsite power available. As depicted in Figure 15.3.3-1 flow through the reactor coolant loop with the affected RCP is rapidly reduced resulting in a reactor trip on the low reactor coolant flow signal. The resulting reduction in heat removal capability is reflected as in increase in RCS average temperature as shown in Figure 15.3.3-4.

The analysis confirmed that the number of rods predicted to be in DNB is less than 10% of the core, which is the value used in the radiological consequence analysis. All rods not meeting the 95/95 limits are assumed to fail. This analysis uses only the minimum DNBR during the event, and a DNBR plot is not provided.

For the rods in DNB analysis case, the normalized core total flow transient is identical to the normalized RCS total flow. For consistency with the MARVEL-M output, the normalized RCS total flow is provided instead of the normalized core flow. Similarly, the average RCS temperature is provided instead of the core average temperature. Inlet coolant temperature does not change during the short duration of the event and is assumed constant in the DNBR calculations; therefore a plot of this parameter is not provided. Additionally, the hot channel heat flux is more closely related to DNBR; therefore, this parameter is provided instead of the average channel heat flux. Plots for average and hot channel exit temperatures, steam fractions, and centerline temperature are not provided since any rod experiencing DNB is assumed to fail. Peak cladding temperature is presented in a separate case. Pressurizer safety valve flow is not reported for this event because RCS pressure remains below the safety valve set pressure. A plot of steam line pressure is not provided because it is bounded by the barrier performance analysis case discussed in Section 15.3.3.4. Containment parameters are not reported for this event because there are no releases directly from the RCS or steam generators inside containment.

For the peak cladding temperature analysis case, the RCS parameter transient responses are similar to those for the rods in DNB analysis case. Therefore, only a plot showing the cladding temperature versus time is provided for this analysis case.

Table 15.3.3-1 lists the key events and the times at which they occur, relative to the initiation of the bounding RCP rotor seizure/shaft break transient for the peak cladding temperature analysis case. Figure 15.3.3-5 is a plot of the peak cladding temperature versus time.

Table 15.3.3-3 summarizes the primary results of this evaluation, including the peak local cladding temperature and oxidation fraction. The maximum fuel cladding temperature is about 2082°F, which is significantly below the limit of 2200°F. The cladding oxidation fraction from the zirconium-water reaction at the hottest point is well below 1%. Due to the small fraction of cladding oxidation, hydrogen generation is negligible. Applying the core coolability criteria used for the loss-of-coolant accident (LOCA) (as described in Section 15.0.0.1.2) the GDC 27 and 28 requirements related to core coolability are met.

15.3.3.4 Barrier Performance

15.3.3.4.1 Evaluation Model

The barrier performance evaluation for peak RCS pressure employs the same MARVEL-M evaluation model as in the core and system performance analysis described in Section 15.3.3.3.1. Input parameters and initial conditions are modified in order to maximize the calculated RCS pressure.

No DNBR calculations (employing the VIPRE-01M computer code) are performed.

15.3.3.4.2 Input Parameters and Initial Conditions

The same input parameters are used as in the DNB failure analysis in Section 15.3.3.3.2 with the exception of the items discussed below.

- The initial power level is taken as 102% of the licensed core thermal power level with initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi above the nominal value. This combination of initial condition uncertainties maximizes peak pressurizer water volume. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

15.3.3.4.3 Results

Table 15.3.3-2 lists the key events and the times at which they occur, relative to the initiation of the transient, for the barrier performance evaluation of the bounding RCP rotor seizure/shaft break transient with offsite power available. Figures 15.3.3-6 and 15.3.3-10 are plots of key system parameters versus time for the peak RCS pressure evaluation for this analysis case.

For this analysis case, the primary parameters of interest are related to the peak RCS and main steam pressures. Following the reactor trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant temperature to rise (shown in Figure 15.3.3-9) and to expand. The expansion of the coolant, combined with the reduced heat transfer in the steam generator, causes an insurge into the pressurizer resulting in a pressure increase throughout the RCS. Figure 15.3.3-8 demonstrates that the peak RCP outlet pressure (provided in Table 15.3.3-3) remains well below 110% of the design pressure.

Steam generator pressure is provided in Figure 15.3.3-10 (in place of steam line pressure) for the purpose of demonstrating that the main steam system pressure meets the acceptance limit. The RCP outlet pressure is the highest pressure in the RCS and is presented in place of RCS pressure for the purpose of confirming the RCS pressure boundary limits are not exceeded. The transient parameters in the DNBR core response analysis are representative of this RCS pressure analysis.

15.3.3.5 Radiological Consequences

The radiological consequences evaluation for this event is based on the alternative source term (AST) guidance documented in Reference 15.3-4.

The radiological consequences evaluation assumes the reactor has been operating at the maximum allowable limit for reactor coolant activity and leaking steam generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. The equilibrium concentrations assumed in the analysis is based on Technical Specification coolant concentration limits.

The radiological consequences evaluation also assumes that 10% of the fuel rods experience local clad temperatures that exceed limits and fail during the postulated RCP rotor seizure transient. The iodine, alkali metal, and noble gas inventories in the gap between the pellets and the clad in these fuel rods are assumed to be released into the reactor coolant. The fraction of fuel rod inventory that is in the gap is described in Table 15.0-8.

The analysis documented in this section conservatively assumes the reactor is cooled by releasing steam from loops via main steam relief or safety valves. Some of the fission products assumed to be released into the reactor coolant during the transient (from the failed fuel rods) can migrate to the secondary coolant during the transient via the assumed leaking steam generator tubes.

Radioisotopes, except noble gases are assumed to be mixed with the secondary coolant and partitioned between the generator liquid and steam before release to the environment. Noble gases entering the secondary are assumed to be released directly to the environment.

The radiological assessment corresponds to the case with offsite power unavailable.

15.3.3.5.1 Evaluation Model

Mathematical models used in the analysis are described in the following sections:

- The offsite and onsite doses are calculated with the RADTRAD code.
- The atmospheric dispersion factors (χ/Q values) used in the analysis are described in Section 15.0.3.3.
- The total effective dose equivalent (TEDE) doses to a receptor at the exclusion area boundary (EAB) and outer boundary of the low-population zone (LPZ) are analyzed using the models described in Section 15.0.3.1 and Appendix 15A.

Figure 15A-4 depicts the leakage sources to the environment modeled in the dose computation.

For evaluating the radiological consequences due to a postulated RCP rotor seizure, the radioactivity released from the steam generator is assumed to be released directly to the environment. The steam generators are assumed to continually discharge steam and entrained activity up to the time of the residual heat removal (RHR) system initiation.

All radioactivity is released to the environment with no consideration given to radioactive decay or cloud depletion by ground deposition during transport to the EAB and LPZ.

15.3.3.5.2 Input Parameters and Initial Conditions

The major assumptions and parameters used in the analysis are itemized in Table 15.3.3-4, Tables 15.0-8 through 15.0-10 and Tables 15.0-12 through 15.0-14.

The concentrations of radionuclides entering the primary and secondary system during the transient are based on the following:

- The radiological consequences evaluation assumes that 10% of the fuel rods experience local clad temperatures that exceed limits and fail during the postulated RCP rotor seizure transient. The fuel rod gap inventory of iodine, alkali metals, and noble gases from these fuel rods are assumed to be released into the reactor coolant. The fuel rod gap fraction is described in Table 15.0-8 for iodine, alkali metals, and noble gases, respectively.

The concentrations of radionuclides in the primary and secondary system, prior to the transient are determined as follows:

- Reactor coolant activities are based on the Technical Specification limit of 1.0 μCi/g dose equivalent DE I-131.

- The noble gas concentrations in the reactor coolant are based on the Technical Specification limit of 300 μCi/g DE Xe-133. Additionally, the pre-accident alkali metal concentrations in the reactor coolant are based on 1% fuel defect.

- The secondary coolant iodine and alkali concentration is 10% of the reactor coolant concentration.

- A 600 gallon per day (gpd) steam generator primary-to-secondary leakage is assumed, which is the Technical Specification limit.

- The activity released from the fuel is assumed to be released instantaneously and homogeneously through the reactor coolant.

- The chemical form of radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. Iodine releases from the steam generators to the environment are assumed to be 97%

elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during iodine spiking.

The only filtration system considered in the analysis which limits the consequences of the RCP rotor seizure transient is the main control room (MCR) heating, ventilation, and air conditioning (HVAC) system.

The χ/Q values and breathing rates are listed in Table 15.0-13. The breathing rates are obtained from NRC Regulatory Guide 1.183 (Ref. 15.3-4).

Other assumptions relating to the transport, reduction, and release of radioactive material to the environment are those covered in Appendix G of RG 1.183 (Ref. 15.3-4).

15.3.3.5.3 Results

The calculated TEDE doses for the limiting 2-hour period at the EAB and for the duration of the transient at the LPZ outer boundary are listed in Table 15.3.3-5.

As shown in Table 15.3.3-5, the TEDE doses for the limiting 2-hour period are calculated to be 0.49 rem at the EAB and 0.70 rem at the LPZ outer boundary. These doses are less than 10% of the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34.

The doses for the MCR for the RCP rotor seizure event are bounded by the doses calculated for the LOCA event described in Section 15.6.5.5. Consequently, no doses are provided for the RCP rotor seizure event.

15.3.3.6 Conclusions

For a seizure of a single reactor coolant pump rotor, the analysis demonstrates that the number of rods predicted to be in DNB is less that the number assumed for the radiological dose analysis. Additionally, the maximum hot spot cladding temperature remains below the limit of 2200°F and the fraction of local cladding oxidation due to the zirconium-water reaction rate is significantly below the limit of 17%.

The peak RCS pressure remains below 110% of the RCS design pressure. Therefore, the integrity of the reactor coolant pressure boundary is maintained. Similarly, because the steam generator pressure does not exceed 110% of the main steam system design pressure, the integrity of the main steam system is maintained.

The resultant doses are well within the guideline values of 10 CFR 50.34.

In addition, this event does not lead to a more serious fault condition.

Table 15.3.3-1
Time Sequence of Events for RCP Rotor Seizure - Cladding Temperature Analysis

Event	Time (sec)
Rotor on one pump locks	0.0
Low reactor coolant flow analytical limit reached	0.1
Reactor trip initiated (rod motion begins)	1.9
Maximum cladding inside temperature occurs	4.6

Table 15.3.3-2
Time Sequence of Events for RCP Rotor Seizure - RCS Pressure Analysis

Event	Time (sec)
Rotor on one pump locks	0.0
Low reactor coolant flow analytical limit reached	0.1
Reactor trip initiated (rod motion begins)	1.9
Maximum reactor coolant pump outlet pressure occurs	5.2

Table 15.3.3-3
Summary of Results for RCP Rotor Seizure

Maximum reactor coolant pump outlet pressure (psia)	2509
Maximum cladding inside temperature, core hot spot (°F)	2082
Zr-$H_2$0 reaction, core hot spot (percentage by weight)	< 1%

Table 15.3.3-4
Parameters Used in Evaluating the Radiological Consequences of RCP Rotor Seizure

Parameter	Value
Initial reactor coolant activity (from rods leaking prior to transient)	
Core thermal power level (MWt)	4540 (2% above the design core thermal power)
Iodine	Initial concentration equal to the 1.0 μCi/g DE I-131 in the reactor coolant. (See Table 15.0-10.)
Alkali metals	Based on 1% fuel defect (See Table 11.1-2.)
Noble gas	300 μCi/g DE Xe-133 (See Table 15.0-12.)
Iodine chemical form	elemental:97%, organic:3%
Initial secondary coolant activity (from rods leaking prior to transient)	
Secondary system initial iodine and alkali metals concentration	Based on 10% of reactor coolant concentration
Source term	
Core activity	See Table 15.0-14.
Fraction of core inventory assumed released into reactor coolant (from rods that fail during RCP Rotor Seizure transient)	
Fraction of fuel rods assumed to fail during transient (%)	10
Radial power peaking factor (to calculate fraction of total inventory in failed rods)	1.78
Iodine fission product gap fraction Alkali metal fission product gap fraction Noble gas fission product gap fraction	See Table 15.0-8. See Table 15.0-8. See Table 15.0-8.
RCS and steam system parameters	
Total steam generator tube leakage (gpd)	600
Reactor coolant mass (lb)	646,000
Secondary coolant mass, 4 steam generators (lb)	456,000
Primary-to-secondary leakage duration (h)	14
Steam released (lb)	
0 to 8 h	1,540,000
8 to 14 h	1,540,000
Iodine partition coefficient	100
Particulate partition coefficient for moisture carryover in the steam generators	1000
Offsite power	Lost after trip
Radiological dose parameters	
χ/Q	See Table 15.0-13.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.3.3-5
Radiological Consequences of a RCP Rotor Seizure

Dose Location	TEDE Dose (rem)
EAB (10 to 12 hours)	0.49
LPZ outer boundary	0.70

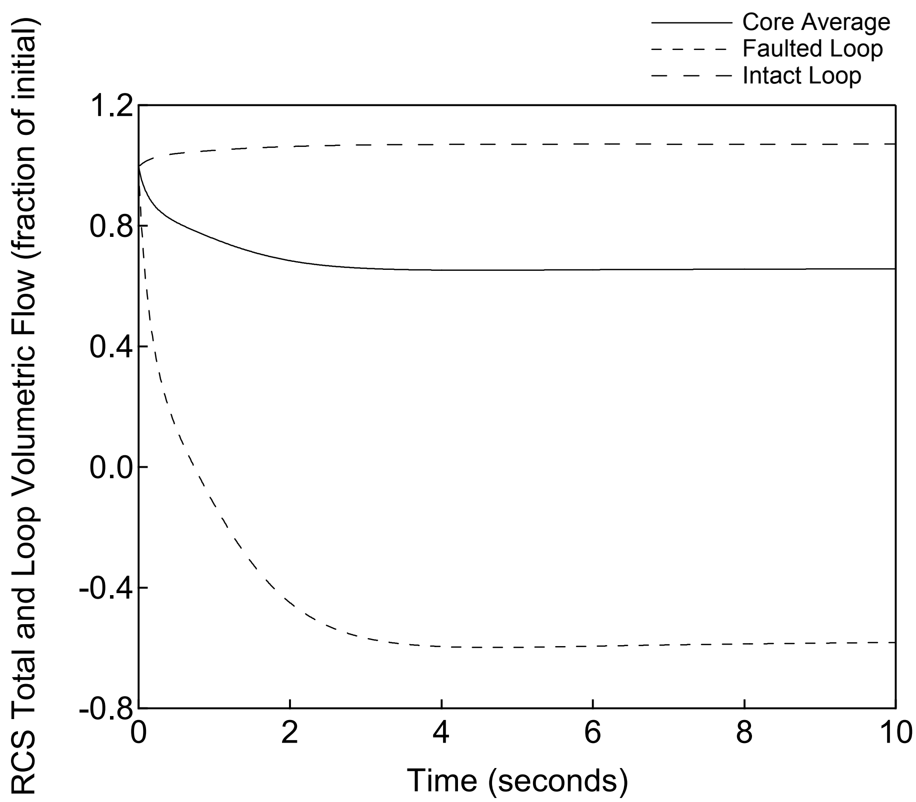


Figure 15.3.3-1 RCS Total and Loop Volumetric Flow versus Time

RCP Rotor Seizure
- Rods in DNB Analysis

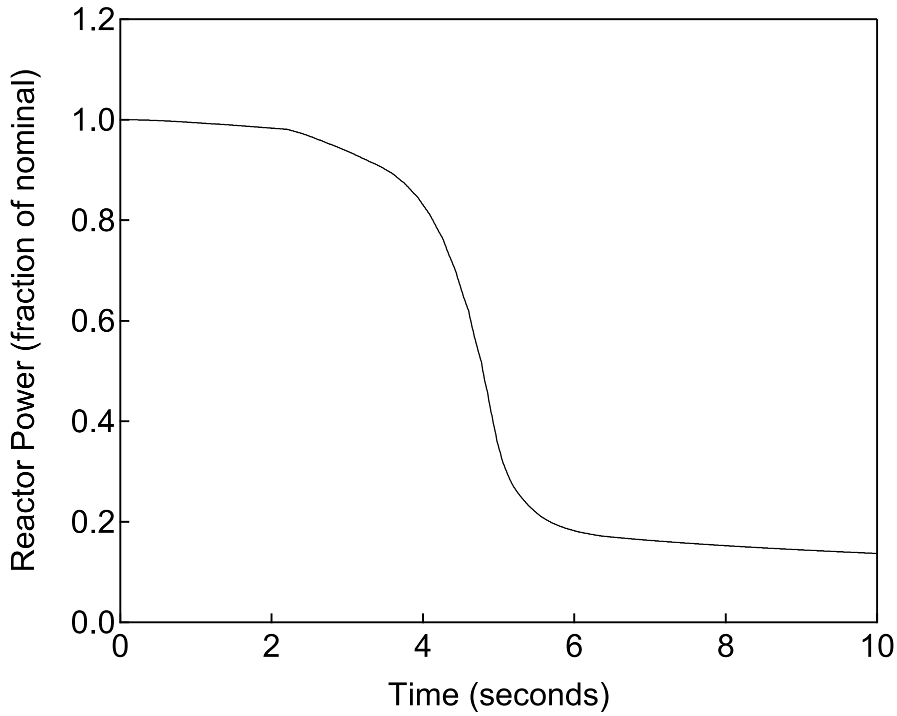


Figure 15.3.3-2 Reactor Power versus Time

RCP Rotor Seizure
- Rods in DNB Analysis

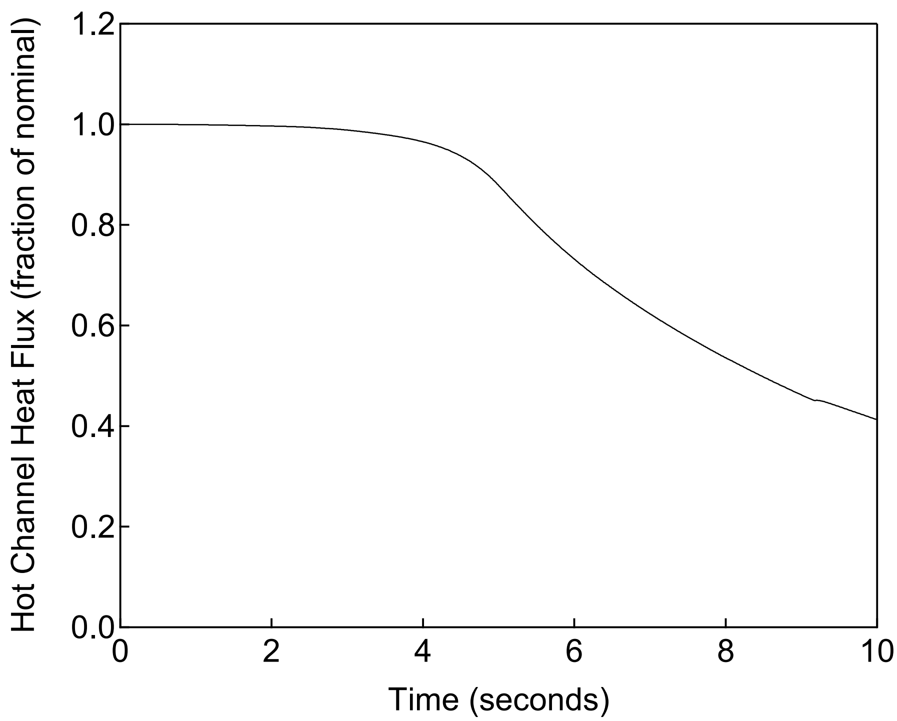


Figure 15.3.3-3 Hot Channel Heat Flux versus Time

RCP Rotor Seizure
– Rods in DNB Analysis

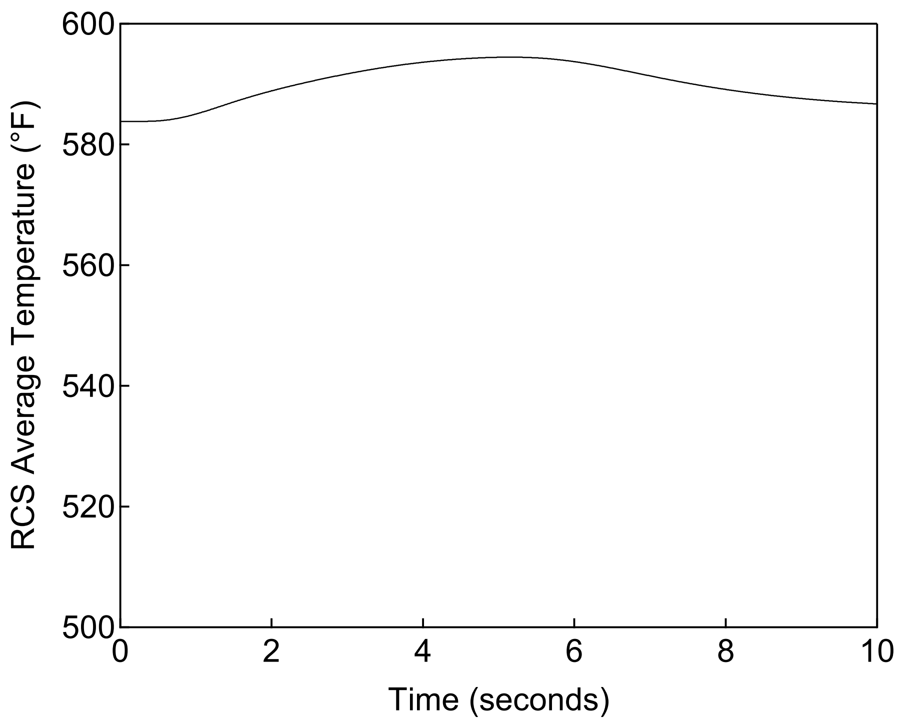


Figure 15.3.3-4 RCS Average Temperature versus Time

RCP Rotor Seizure
- Rods in DNB Analysis

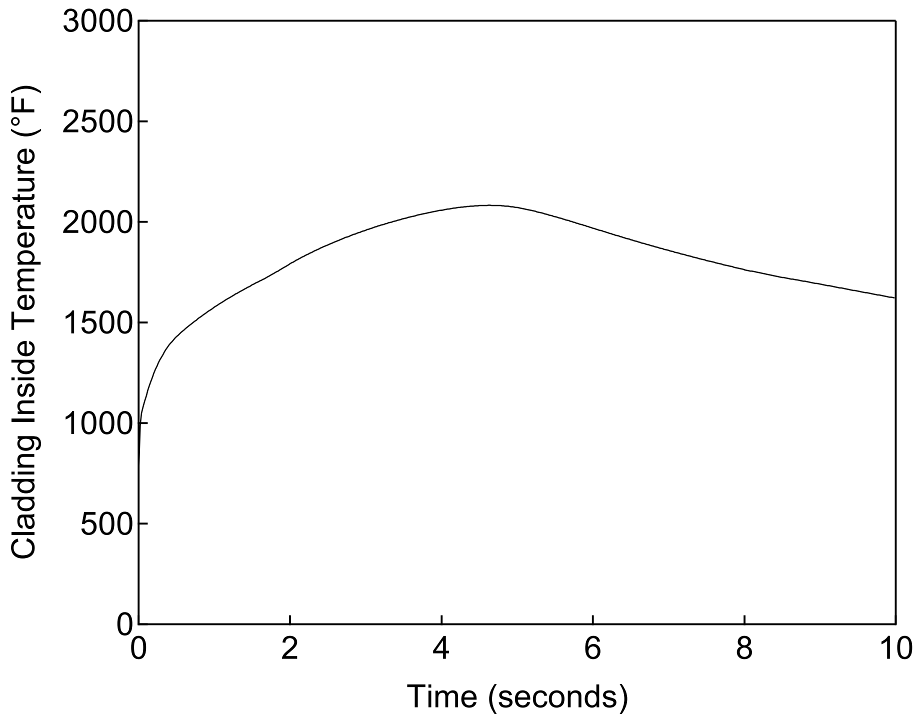


Figure 15.3.3-5 Cladding Inside Temperature versus Time

RCP Rotor Seizure
- Cladding Temperature Analysis

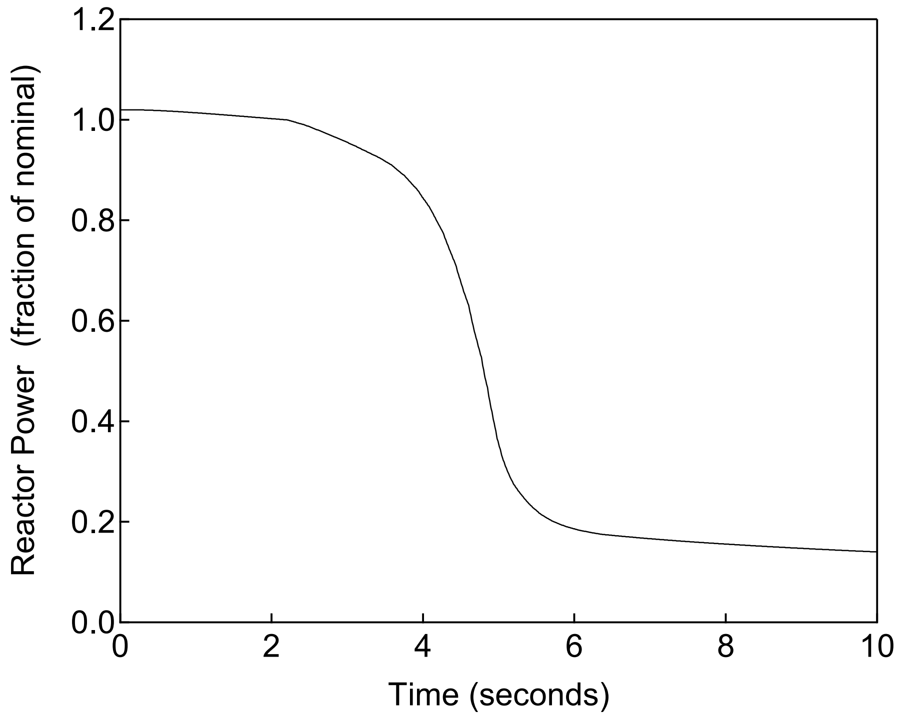


Figure 15.3.3-6 Reactor Power versus Time

RCP Rotor Seizure
- RCS Pressure Analysis

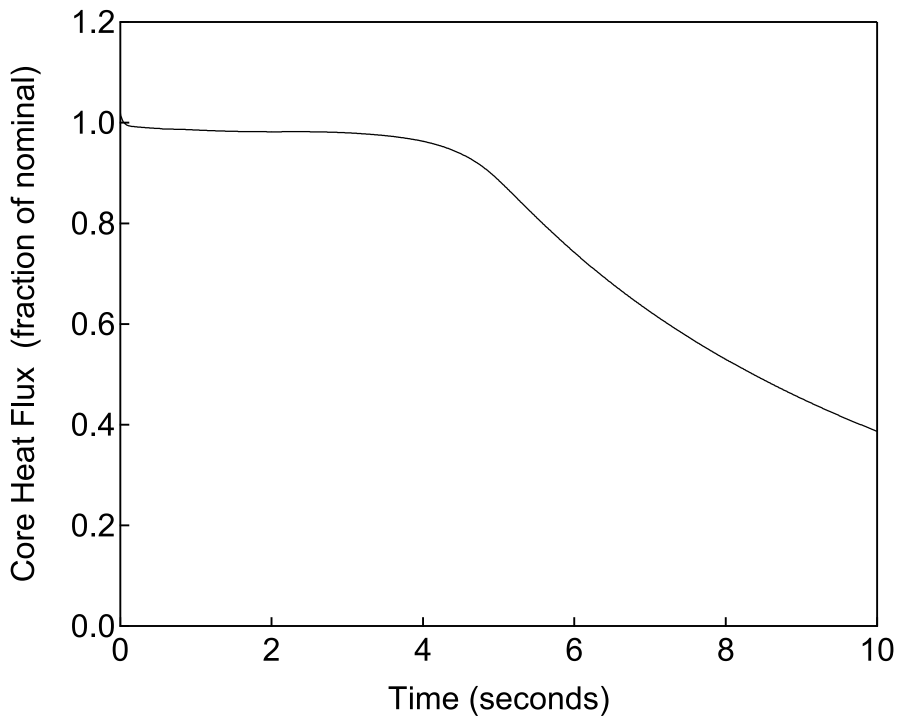


Figure 15.3.3-7 Core Heat Flux versus Time

RCP Rotor Seizure
- RCS Pressure Analysis

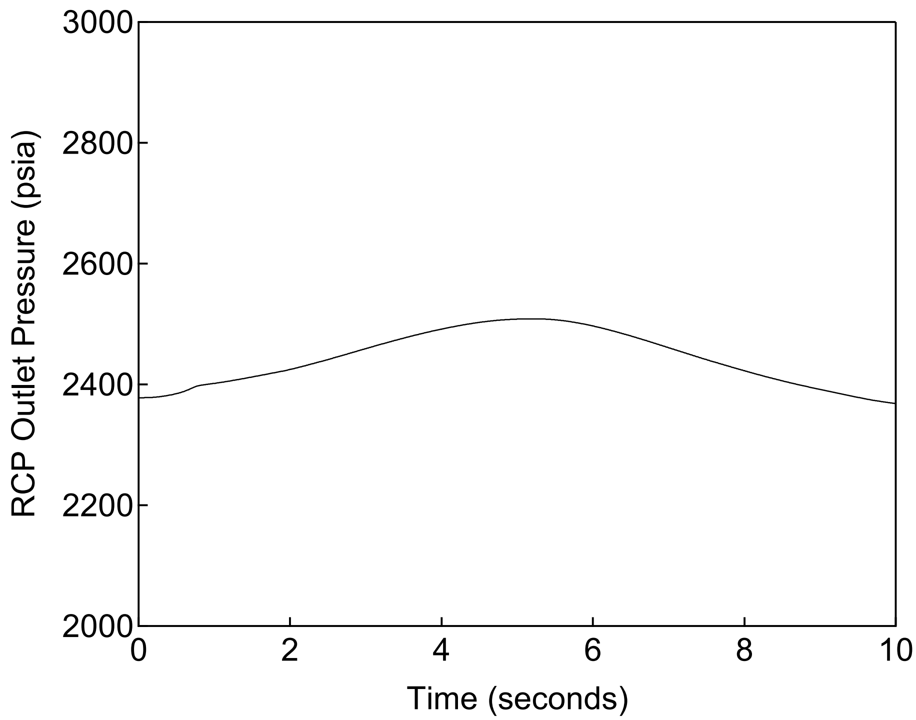


Figure 15.3.3-8 RCP Outlet Pressure versus Time

RCP Rotor Seizure
- RCS Pressure Analysis

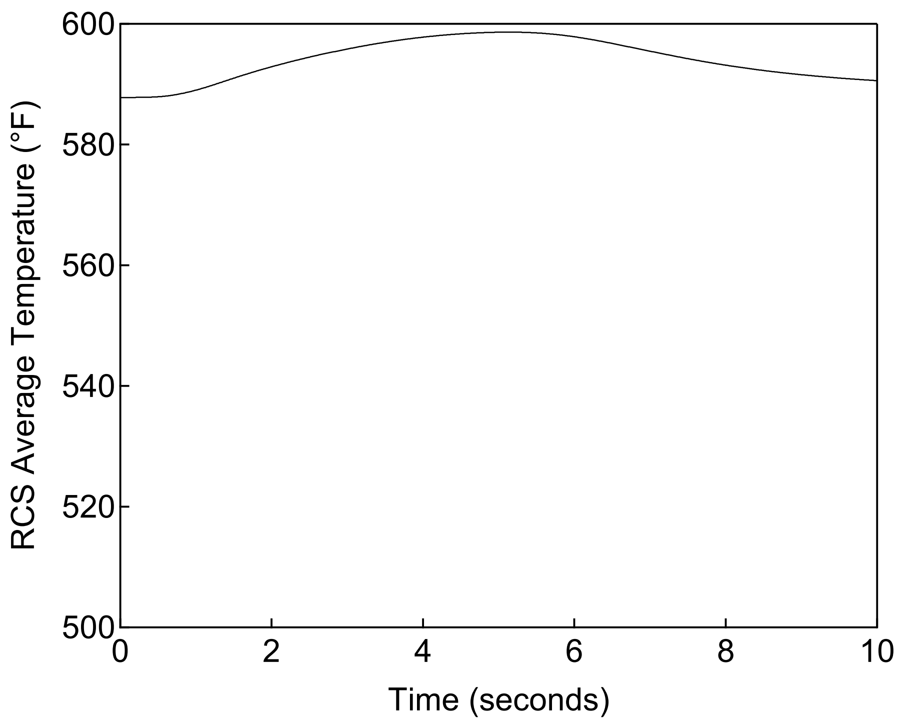


Figure 15.3.3-9 RCS Average Temperature versus Time

RCP Rotor Seizure
- RCS Pressure Analysis

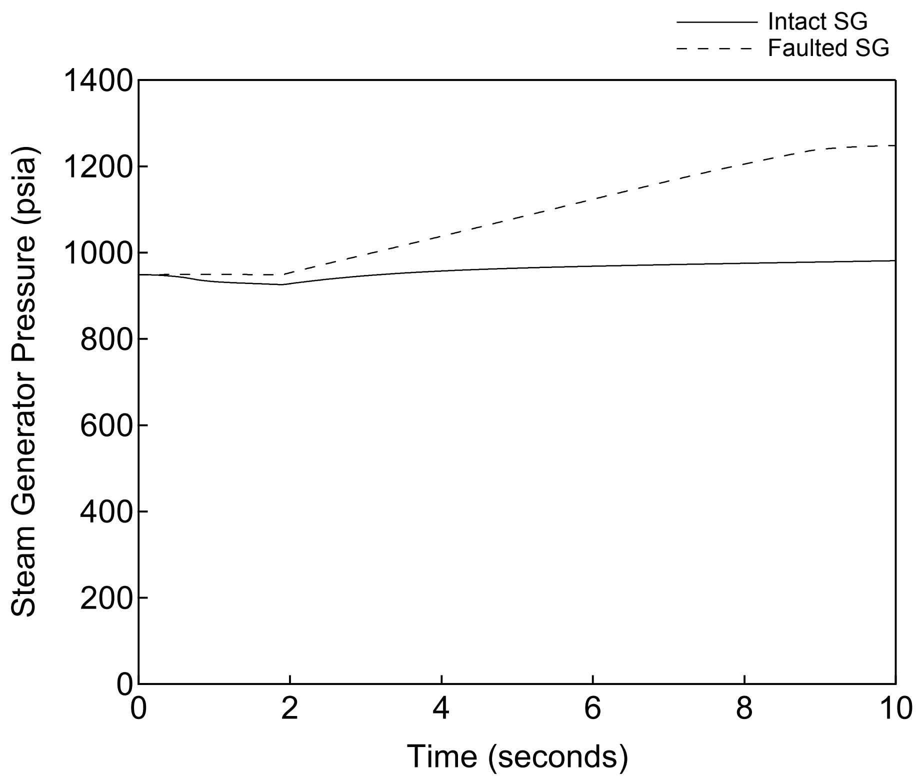


Figure 15.3.3-10 Steam Generator Pressure versus Time

RCP Rotor Seizure
- RCS Pressure Analysis

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Frequency Classification

This event is initiated by the instantaneous break (failure, or fracture and separation) of one of the reactor coolant pump (RCP) shafts during power operation. This postulated shaft break would cause a reduction in the reactor coolant flow and decrease the core cooling capacity. This could, in turn, lead to an increase in the reactor fuel temperature, primary coolant temperature, and reactor pressure.

Possible causes of this event are an undetected flaw in the shaft, or stresses caused by vibration or nonuniform temperatures.

The break of an RCP shaft is a postulated accident (PA). Historically, this was classified as a Condition IV event in the older plant condition frequency grouping as defined in ANSI N18.2 (Ref. 15.3-3). Event frequency conditions are described in Section 15.0.0.1.

As described in Section 15.3.3, a conservative bounding analysis was performed for the reactor coolant pump rotor seizure that bounds the response and results for the reactor coolant pump shaft break. As a result, no analysis is performed or described for this event.

15.3.4.2 Sequence of Events and Systems Operation

The analysis performed for the RCP rotor seizure transient (Section 15.3.3) bounds the response and results for the reactor coolant pump shaft break.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

15.3.4.3 Core and System Performance

The analysis performed for the RCP rotor seizure transient (Section 15.3.3) bounds the response and results for the reactor coolant pump shaft break.

15.3.4.4 Barrier Performance

The analysis performed for the RCP rotor seizure transient (Section 15.3.3) bounds the response and results for the reactor coolant pump shaft break.

15.3.4.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the reactor coolant pump rotor seizure evaluated in Section 15.3.3.5.

15.3.4.6 Conclusions

The analysis performed for the RCP rotor seizure transient (Section 15.3.3) bounds the response and results for the RCP shaft break.

15.3.5 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.3.6 References

15.3-1 Non-LOCA Methodology, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.3-2 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.3-3 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.3-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.4 Reactivity and Power Distribution Anomalies

A number of postulated faults can result in reactivity and power distribution anomalies. Reactivity changes can be caused by control rod movement, control rod ejection, boron concentration changes, or addition of cold water to the reactor coolant system. Power distribution changes can be caused by control rod movement, control rod misalignment, control rod ejection, or fuel assembly mislocation. These events are discussed in this section. Analysis results for the limiting cases are also presented.

Analyses of the following events are described in this section:

- Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition
- Uncontrolled control rod assembly withdrawal at power
- Control Rod Misoperation
  - One or more dropped rod cluster control assembles (RCCAs) within a group or bank
  - One or more misaligned RCCAs (relative to their bank)
  - Uncontrolled withdrawal of a single RCCA
- Startup of an inactive loop or recirculation loop at an incorrect temperature (not applicable to the US-APWR)
- Flow controller malfunction causing an increase in BWR core flow rate (not applicable to the US-APWR)
- Inadvertent decrease in boron concentration in the reactor coolant system
- Inadvertent loading and operation of a fuel assembly in an improper position
- Spectrum of rod ejection accidents

Except for the uncontrolled withdrawal of a single RCCA and the spectrum of rod ejection accidents, these events are classified as anticipated operational occurrences (AOOs) as defined in Section 15.0.0.1. The uncontrolled withdrawal of a single RCCA and the spectrum of rod ejection are classified as postulated accidents (PAs).

The applicable transients in this section have been analyzed. Radiological consequences are only reported for the single RCCA withdrawal and rod ejection events. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing (rod ejection) as discussed in Section 15.4.8.

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

15.4.1.1 Identification of Causes and Frequency Classification

A rod cluster control assembly (RCCA) withdrawal incident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCA banks, which results in a power increase. The occurrence of such a transient can be caused by a malfunction of the reactor control system or the control rod drive system. This incident could occur with the reactor subcritical, at hot zero power, or at power. In this section, the transient event analyzed is for the reactor at hot zero power. The at-power case is discussed in Section 15.4.2.

This event is classified as an anticipated operational occurrence (AOO). Historically, this event has been classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.4-1). Event frequency conditions are described in Section 15.0.0.1.

In addition to the acceptance criteria generally applicable to AOOs as described in Section 15.0.0.1, SRP 15.4.1 requires that for this event that (1) the thermal margin limits of SRP 4.4 (DNBR) are met and (2) the fuel centerline temperature does not exceed the melting point as specified in SRP 4.2.

15.4.1.2 Sequence of Events and System Operation

The sequence and timing of major events for the control rod assembly withdrawal from subcritical event is described in the results section.

RCCA bank withdrawal adds reactivity at a prescribed and controlled rate to bring the reactor from a subcritical condition to a low power level during startup. Although the initial approach to criticality uses the method of boron dilution, a normal startup starts with RCCA bank withdrawal prior to the boron dilution. RCCA bank movement can deliver much faster changes in reactivity than boron concentration changes.

The control rod drive mechanisms are grouped into pre-selected bank configurations. The circuit design prevents the RCCA banks from being withdrawn in any manner other than their proper withdrawal sequence. Power supplied to the RCCA banks is controlled such that no more than two banks are withdrawn at a time. The RCCA drive mechanisms are the magnetic latch type, and coil actuation sequencing provides variable speed travel. The maximum reactivity insertion rate is based on the simultaneous withdrawal of two sequential RCCA banks resulting in the maximum combined rod worth at maximum speed.

The neutron flux response to the continuous reactivity insertion due to RCCA movement is a self-limiting power excursion that is characterized by a rapid rise that is terminated by the reactivity feedback of the negative Doppler coefficient. The Doppler feedback effect is faster than the control rod control system response, which limits the power increase to an acceptable level during the time between the trip signal and the completion of post-trip rod motion. Should a continuous RCCA withdrawal occur, the following automatic trip signals are assumed to be available to provide protection from this transient:

- High source range neutron flux
- High intermediate range neutron flux
- High power range neutron flux (low setpoint)
- High power range neutron flux (high setpoint)
- High power range neutron flux rate

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

15.4.1.3 Core and System Performance

15.4.1.3.1 Evaluation Model

The TWINKLE-M code is used to calculate the core transient including core average and local power behavior following an uncontrolled control rod assembly bank withdrawal. The increase in local power and the Doppler feedback due to the increased effective fuel temperature are evaluated in each spatial mesh using a one-dimensional method in the axial direction. Changes made by MHI to increase the number of meshes and the use of TWINKLE-M for one-dimensional transient calculations are further described in Reference 15.4-2.

The VIPRE-01M code (Ref. 15.4-3) calculates the DNBR and the fuel temperature at the hot spot during the transient using an interface file created by the TWINKLE-M code.

15.4.1.3.2 Input Parameters and Initial Conditions

Table 15.0-1 summarizes the initial conditions used to evaluate this event. To ensure conservative results for a startup incident, the following assumptions are used in the analysis:

- The Doppler feedback is conservatively estimated by multiplying the fast absorption cross section for the given change in the calculated fuel effective temperature by a conservative multiplier. In the MHI one-dimensional methodology, the Doppler weighting factor for radial power distribution effect is conservatively ignored (assumed to be 1.0).

- Moderator reactivity feedback has a relatively minor contribution during the initial phase of the transient. The reason is that the heat transfer between the fuel and moderator takes much longer than the neutron response time. However, after the initial neutron flux peak occurs, the moderator reactivity feedback slows the decrease of neutron power. The analysis conservatively assumes the most positive moderator temperature coefficient value as defined in Table 15.0-1 to assure a conservative power transient during all stages of the transient.

- The analysis assumes that the reactor is at zero power. This assumption leads to the use of a higher (and thus conservative) initial reactor coolant system (RCS) temperature, which is more conservative than using a lower initial temperature since the higher initial temperature leads to a greater fuel-to-water heat transfer coefficient, a greater fuel-specific heat, and a smaller Doppler (i.e., less negative) coefficient. The combination of these parameters tends to reduce the Doppler feedback effect and thus maximize the neutron flux peak. NOTE: In TWINKLE-M appropriate cross section data is selected to assure minimum Doppler feedback conditions. In addition, the analysis assumes the effective multiplication factor (k_{eff}) to be one, which also maximizes the neutron flux peak.

- The initial values of reactor coolant average temperature and RCS pressure are assumed to be 4°F above and 30 psi below the values corresponding to hot standby conditions. This combination of initial condition uncertainties minimizes DNBR and maximizes fuel temperature calculated by VIPRE-01M.

- The positive reactivity insertion rate of 75 pcm/sec used in the analysis is greater than that for the simultaneous withdrawal of two sequential RCCA banks with the highest worth at the maximum speed (45 inches per minute). A greater value is used because of uncertainties in the RCCA bank worth calculation.

- Conservative assumptions for the trip simulation (trip reactivity, rod drop time, reactor trip system signal processing delays) are used in the analysis. The reactor trip is simulated by dropping fully withdrawn rod banks into the core. Maximum time delay from reactor trip signal to rod motion and a conservative RCCA insertion curve are assumed as described in Table 15.0-4 and Section 15.0.0.2.5, respectively. The trip reactivity used is the design limit, which is -2%Δk/k.

- The reactor is assumed to be automatically tripped by the high power range neutron flux (low setpoint) signal. Table 15.0-4 summarizes the reactor trip setpoints assumed in the analysis.

- The most limiting axial and radial power shapes with the two highest combined worth banks is used for the DNBR and fuel temperature calculation.

- The analysis assumes the initial power level to be below that of any shutdown condition (10^{-13} of nominal power level). The low initial power, in conjunction of the maximum reactivity insertion, yields the highest peak heat flux.

- The analysis assumes that four reactor coolant pumps are in operation.

15.4.1.3.3 Results

The predicted results for the uncontrolled control rod assembly bank withdrawal from a subcritical condition are shown in Figures 15.4.1-1 through 15.4.1-4. Figure 15.4.1-1 shows the reactor power transient. Although the reactor is immediately tripped at the analytical limit of 35% of nominal power, the Doppler power feedback limits the power increase. Figure 15.4.1-2 shows the heat flux response. The effect of the fuel time constant on smoothing the heat flux response to reactor power can be seen by comparing the first two figures. Figure 15.4.1-4 shows the peak fuel centerline temperature. It can be seen that the peak fuel temperature is well below its respective design limit. The DNBR transient is shown in Figure 15.4.1-3, which indicates that the minimum DNBR remains above the 95/95 design limit at all times.

The sequence of events for this accident is shown in Table 15.4.1-1. With the reactor tripped, the plant returns to a stable condition and may be subsequently brought to cold shutdown by the appropriate normal plant shutdown procedures.

A plot for peak fuel rod power is not presented. For DNBR calculations, the hot channel heat flux is provided. Because core inlet temperature is assumed constant for the short duration of the event, a transient plot is not provided. The DNBR remains well above the 95/95 limit, so average and hot channel exit temperatures are not presented. The pressurizer does not fill, so all safety valve flow is steam, which limits pressure to below 110% of the RCS design pressure. No releases to the containment vessel are predicted for this event, so containment parameters are not presented.

15.4.1.4 Barrier Performance

15.4.1.4.1 Evaluation Model

The evaluation model used for the peak RCS pressure analysis is similar to the model used for the DNBR analysis described in Section 15.4.1.3.1. The TWINKLE-M code is used to analyze the core average power histories following the uncontrolled control rod assembly bank withdrawal from subcritical event. The VIPRE-01M code generates a time-dependent core total void fraction and core heat flux interface file which is used by the MARVEL-M code to calculate the RCS pressure transient. Additional details regarding this methodology are provided in Reference 15.4-2.

15.4.1.4.2 Input Parameters and Initial Conditions

The barrier performance case for peak RCS pressure is similar to the DNBR analysis described in Section 15.4.1.3.2 with the following differences:

- The analysis assumes that the reactor is at zero power with initial reactor coolant temperature 4°F above and the pressurizer pressure 30 psi above the values corresponding to hot standby conditions. This combination of initial condition uncertainties maximizes RCS pressure calculated by MARVEL-M.

- Pressurizer spray is not assumed.

15.4.1.4.3 Results

The reactor coolant pump (RCP) outlet pressure is the highest pressure in the RCS and is presented in place of RCS pressure for the purpose of confirming the reactor coolant pressure boundary limits are not exceeded. Figure 15.4.1-5 indicates that the reactor coolant system pressure remains well below 110% of the system design pressure, so the integrity of the reactor coolant pressure boundary is maintained. The main steam system pressure is not challenged by this event.

The other transient parameters of interest for this event are bounded by the results discussed in the DNBR core response analysis in Section 15.4.1.3.3, so only the RCP outlet pressure plot is provided for the peak RCS pressure case.

15.4.1.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the rod ejection accidents evaluated in Section 15.4.8.5.

15.4.1.6 Conclusions

The uncontrolled control rod assembly bank withdrawal from subcritical causes a rapid power excursion that is limited by Doppler feedback. The resulting transient does not cause the minimum DNBR to decrease below the 95/95 limit, and no fuel failures are predicted. Additionally, the peak fuel centerline temperature remains well below its design limit for the entire transient.

The reactor coolant system pressure remains well below 110% of its system design pressure, so the integrity of the reactor coolant pressure boundary is maintained.

This event does not lead to a more serious fault condition.

Table 15.4.1-1
Time Sequence of Events for Uncontrolled Control Rod Assembly Withdrawal from a Subcritical

Event	Time (sec)
Initiation of uncontrolled rod withdrawal from 10^{-13} of nominal power	0.0
High power range neutron flux (low setpoint) analytical limit reached	10.8
Peak reactor power occurs	11.0
Reactor trip initiated (rod motion begins)	11.4
Peak heat flux occurs	12.0
Minimum DNBR occurs	12.0
Peak fuel centerline temperature occurs	14.2

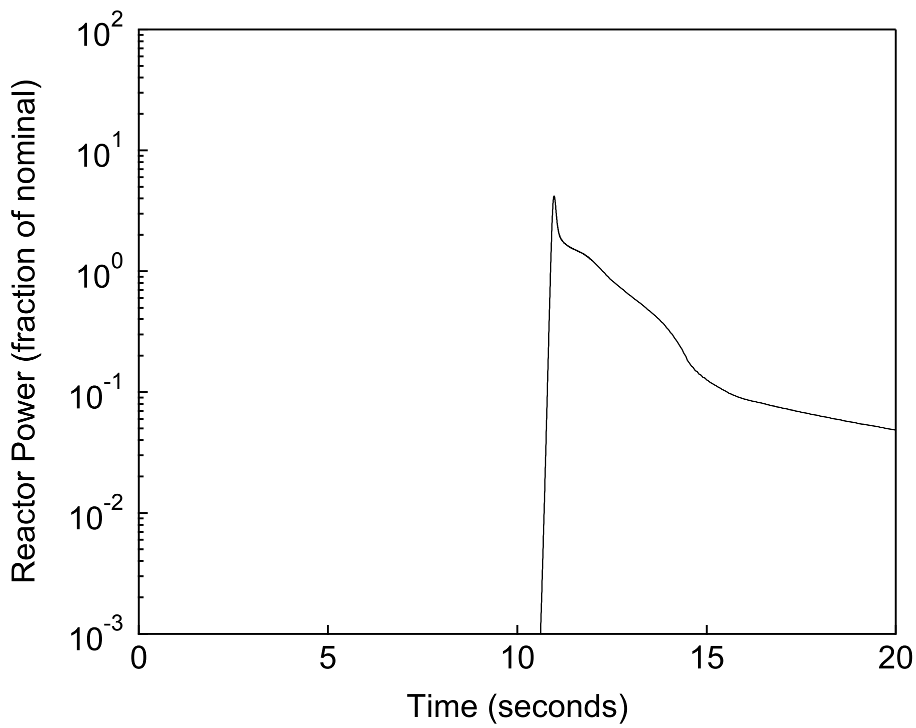


Figure 15.4.1-1 Reactor Power versus Time

Uncontrolled Control Rod Assembly Withdrawal from a Subcritical - DNBR Analysis

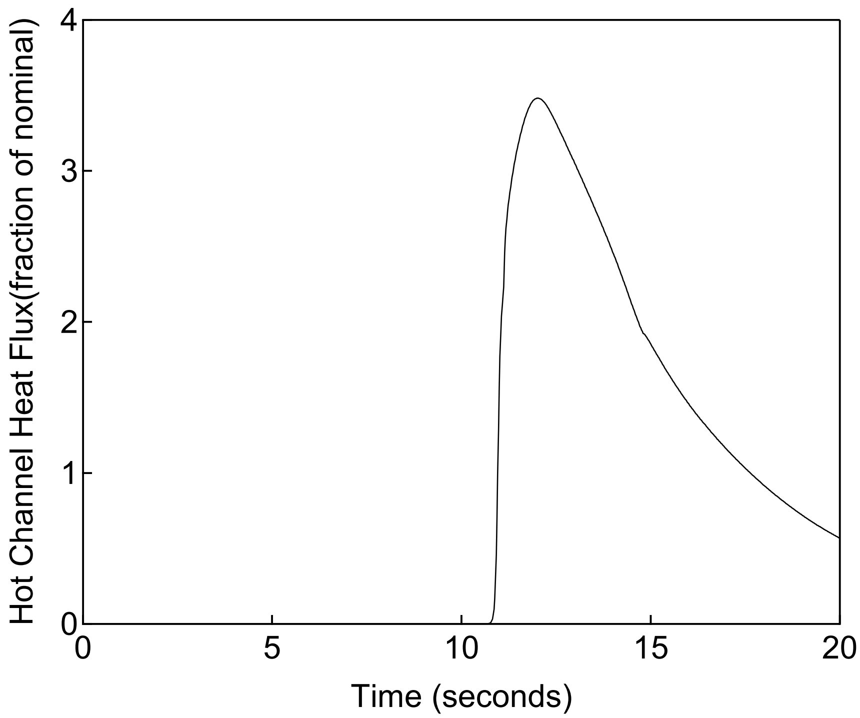


Figure 15.4.1-2 Hot Channel Heat Flux versus Time

Uncontrolled Control Rod Assembly Withdrawal from a Subcritical
- DNBR Analysis

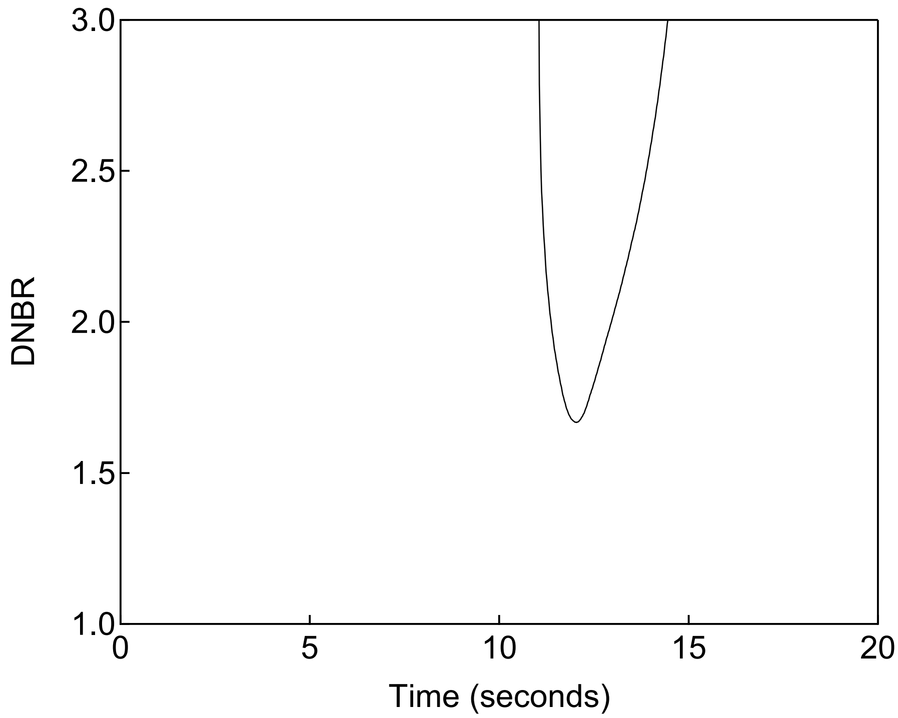


Figure 15.4.1-3 DNBR versus Time

Uncontrolled Control Rod Assembly Withdrawal from a Subcritical
- DNBR Analysis

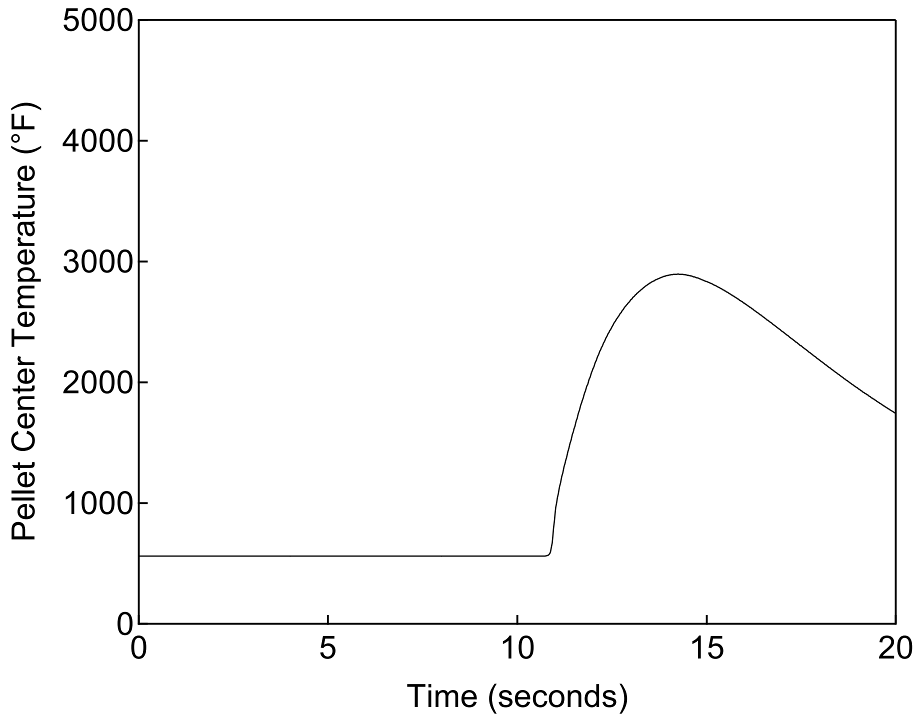


Figure 15.4.1-4 Fuel Temperature versus Time

Uncontrolled Control Rod Assembly Withdrawal from a Subcritical - Fuel Temperature Analysis

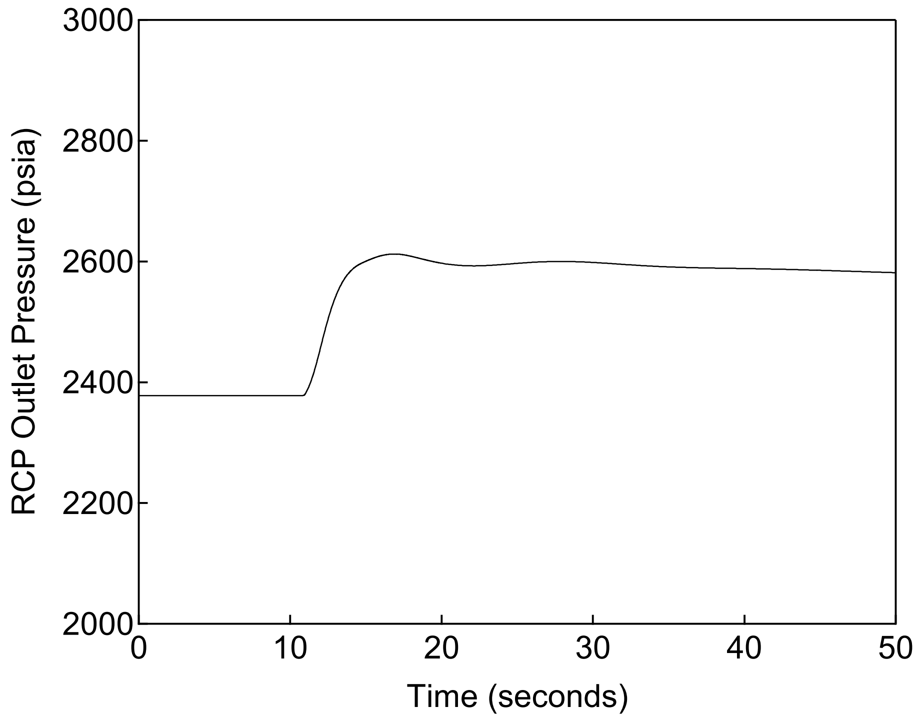


Figure 15.4.1-5 RCP Outlet Pressure versus Time

Uncontrolled Control Rod Assembly Withdrawal from a Subcritical
- RCS Pressure Analysis

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 Identification of Causes and Frequency Classification

The uncontrolled control rod assembly bank withdrawal at power is caused by a control system or rod control system failure that causes a bank withdrawal to occur. The event may occur from various initial power levels and the positive reactivity insertion can vary from zero to 75 pcm/sec depending on rod cluster control assembly (RCCA) withdrawal rate and assumed maximum bank worth.

An uncontrolled control rod assembly bank withdrawal at power results in an increase in core heat flux. Since the heat extracted from the steam generator lags behind the core power until the steam generator pressure reaches the main steam safety valve setpoint, the reactor coolant temperature tends to increase. Without a manual or automatic reactor trip, the power mismatch and the rise of reactor coolant temperature could eventually result in DNB. To prevent damage to the fuel cladding, the reactor trip system (RTS) is designed to terminate the transient before the DNBR reaches the design limit.

This event is classified as an anticipated operational occurrence (AOO). Historically, this event has been classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.4-1). Event frequency conditions are described in Section 15.0.0.1.

15.4.2.2 Sequence of Events and Systems Operation

The sequence and timing of the major events for the uncontrolled control rod assembly bank withdrawal at power event is described in the results section.

RCCA bank withdrawal adds reactivity to the reactor at a prescribed and controlled rate resulting in a power excursion. The following automatic reactor trip signals are assumed to be available to provide protection from this transient:

- High power range neutron flux (high setpoint)
- High power range neutron flux rate
- Over power ΔT
- Over temperature ΔT
- High pressurizer pressure
- High pressurizer water level

The RTS over power and over temperature ΔT trips are designed to provide margin to the core design limits. The details of using the combination of over power ΔT and over temperature ΔT trips to provide reactor protection over the full range of RCS coolant conditions is described in Section 7.2.

Besides the reactor trip, the US-APWR has the following withdrawal blocks to stop any RCCA withdrawal movement:

- Low turbine power (1st stage impulse pressure)
- High intermediate and power range neutron flux
- Over temperature ΔT
- Over power ΔT

Figure 15.0-1 shows allowable reactor coolant average temperature and reactor ΔT for the design power distribution and flow as a function of reactor coolant pressure. The over power ΔT and over temperature ΔT trip lines are shown as “protection lines” in the diagram. These protection lines are drawn to include all adverse instrumentation errors and uncertainty, in order to assure that a trip occurs well within the bounded area.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause an reactor coolant pump (RCP) coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

The limiting single failure for this event is the failure of one train of the RTS. Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

15.4.2.3 Core and System Performance

15.4.2.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, reactor coolant pressure, reactor coolant temperature, hot spot heat flux, pressurizer water volume and minimum DNBR following an uncontrolled RCCA bank withdrawal at power. This evaluation model is described in Section 15.0.2.2.1. Additional details regarding the MARVEL-M code are provided in Reference 15.4-2.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

A positive reactivity insertion into the core prior to trip is simulated by an external reactivity input. A range of cases utilizing different reactivity insertion rates up to a maximum positive reactivity insertion rate of 75 pcm/sec are evaluated. Several cases at both beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions are evaluated to determine the sensitivity of the calculated DNBR to feedback conditions and power level. The focus of this reactivity insertion event is the challenge to the DNB design limit resulting from the power increase. Therefore, only the plant transient responses for the most limiting DNBR cases are provided. (See Section 15.4.2.3.3 for a discussion of the results.)

15.4.2.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for an uncontrolled control rod assembly bank withdrawal at power event:

- Consistent with the use of the RTDP, the initial values of reactor power, reactor coolant temperature, reactor coolant flow rate and reactor coolant pressure are assumed to be the nominal values without uncertainties provided in Table 15.0-3.

- Conservative axial power profile and radial power distributions are assumed in the analysis as described in Section 15.0.0.2.3.

- For the purpose of determining trip reactivity, the highest worth rod assembly is assumed to be stuck at its fully withdrawn position. The trip reactivity is -4% $\Delta k/k$ for the full power and the trip reactivity for reduced power is assumed to be the value that would result in the shutdown margin at hot zero power 1.6 % $\Delta k/k$ that is corresponding to the most restrictive time in the core cycle. See Section 15.0.0.2.5 for additional details.

- The analysis assumes the following reactor trips and uses conservative assumptions for reactor trip setpoint and delay times as described in Table 15.0-4.

  - High power range neutron flux (high setpoint)
  - Over power ΔT
  - Over temperature ΔT
  - High pressurizer pressure

  For slower reactivity insertion rates from hot full power (HFP) conditions, the over temperature ΔT trip provides DNB protection. At higher insertion rates, the power range neutron flux trip provides protection. At lower reactivity insertion rates from reduced power conditions, the over temperature ΔT trip and high pressurizer pressure trip provide protection.

  As described in Section 7.2, the over temperature ΔT trip has an axial offset penalty that reduces the setpoint if the axial power distribution is severe for DNB. This portion of the trip is not modeled in the MARVEL-M point kinetics model.

- Because a failure of the rod control system initiates this event, the rod control system is not otherwise assumed to operate.

- Pressurizer spray is assumed to operate. This minimizes DNBR for any given combination of power and temperature.

- Two cases of reactivity coefficients are considered. Figure 15.4.2-1 provides plots of the minimum DNBR as a function of reactivity insertion rate initiated from hot full power for both the BOC and EOC conditions mentioned below.

Minimum reactivity feedback	BOC conditions use the minimum moderator density coefficient value as defined in Section 15.0.0.2.4 and the Doppler coefficient corresponding to the minimum feedback limit in Figure 15.0-2.
Maximum reactivity feedback	EOC conditions use the maximum moderator density coefficient value as defined in Section 15.0.0.2.4 and the Doppler coefficient corresponding to the maximum feedback limit in Figure 15.0-2.

- Three cases of initial power level are analyzed. Figure 15.4.2-2 provides plots of the minimum DNBR as a function of reactivity insertion rate for BOC conditions for 10%, 75%, and 100% power.

- For the DNBR analysis, various reactivity insertion rates are analyzed at each power level from approximately zero pcm/sec to 75 pcm/sec. The maximum value bounds the maximum possible reactivity insertion rate considering both rod speed and differential rod worth.

15.4.2.3.3 Results

Figure 15.4.2-1 presents minimum DNBR as a function of reactivity insertion rate initiated from HFP conditions assuming both minimum (BOC) and maximum (EOC) reactivity feedback. For the minimum feedback cases, the minimum DNBR shown on Figure 15.4.2-1 occurs at a reactivity insertion rate of 5.0 pcm/sec. Two reactor trip functions - over temperature ΔT and high power range neutron flux - provide protection over the entire range of insertion rates. The over temperature ΔT trip provides protection for slow reactivity insertion rates and the high neutron flux provides protection at higher rates. The minimum DNBR occurs at the breakpoint where these protection lines intersect.

Figure 15.4.2-2 presents the minimum DNBR as a function of reactivity insertion rate for minimum (BOC) reactivity feedback starting at 10%, 75% and 100% power conditions. This shows sensitivity of minimum DNBR to initial power for the limiting minimum feedback case. At initial power levels of 10% and 75%, the reactor is tripped by the high power range neutron flux trip for higher reactivity insertion rates and high pressurizer pressure trip for medium reactivity insertion rates. For slower reactivity insertion rates, the main steam safety valves open and pressurizer pressure drops before the high pressurizer pressure trip setpoint is reached, then the reactor trips by the over temperature ΔT trip. This figure confirms that the limiting DNBR occurs at a reactivity insertion rate of 5.0 pcm/sec for minimum (BOC) feedback conditions for HFP, rather than reduced power conditions. The 100% power curve provided on Figure 15.4.2-2 is identical to the minimum feedback curve shown in Figure 15.4.2-1.

The shape of the curves presented in Figures 15.4.2-1 and 15.4.2-2 is due to the combined effects of the core and coolant system response and the RTS action in initiating a reactor trip. It is also clear that the high power range neutron flux and over temperature ΔT trips provide DNB protection over the entire range of reactivity insertion rates. The minimum DNBR is greater than the 95/95 limit described in Section 4.4 over the entire range of reactivity insertion rates.

The minimum feedback curve also shows that at high insertion rates, approaching 75 pcm/sec, the minimum DNBR has turned down and has a negative slope. The maximum feedback at 100% power is also shown on the same graph. The minimum DNBR at both the breakpoint and at the maximum withdrawal rate is lower for the minimum feedback case.

Transient parameter plots are provided for the uncontrolled bank withdrawal at power with minimum feedback for the rapid RCCA withdrawal (75 pcm/sec) scenario in Figures 15.4.2-3 through 15.4.2-8. Since the reactor trip on high power range neutron flux is rapid with respect to the thermal time constants of the core and the reactor coolant system, the changes in temperature and pressure are relatively small and the DNBR margin is maintained.

The same parameters are provided for the 5.0 pcm/sec insertion rate minimum feedback scenario in Figures 15.4.2-9 through 15.4.2-14. Because of the slow reactivity insertion rate, the reactor is tripped on over temperature ΔT after a longer period and the rise in temperature and pressure is greater than the fast withdrawal case. However, the DNBR margin is maintained throughout the entire transient.

The sequence of events for these specific cases is provided in Table 15.4.2-1 (75 pcm/sec) and Table 15.4.2-2 (5.0 pcm/sec).

A plot of peak fuel centerline temperature is not provided because the radial power distribution resulting from RCCA bank withdrawals are uniform and the DNBRs remain above the 95/95 limit. Because the DNBRs most closely follow hot spot heat flux, a plot for core average heat flux is not presented. In addition, RCS average temperature plots are provided as being representative of various temperatures. Average and hot channel exit temperatures are not key parameters since the DNBRs are above the 95/95 limit. Steam generator pressure is not presented since the RCS temperature cannot cause saturated steam pressures on the secondary side that challenge the main steam system pressure design limits. Releases to containment are not predicted and containment parameters are not presented.

15.4.2.4 Barrier Performance

The uncontrolled control rod assembly bank withdrawal at power event does not result in exceeding any reactor coolant pressure boundary or containment volume fission product barrier design limits. The results of the Core and System Evaluation case demonstrate that the reactor coolant system pressure remains well below 110% of system design pressure. In addition, the main steam pressure cannot challenge the main steam system pressure design limit, as discussed in Section 15.4.2.3.3. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.4.2.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the rod ejection accidents evaluated in Section 15.4.8.5.

15.4.2.6 Conclusions

In summary, the shape of the minimum DNBR curves versus reactivity insertion rate is controlled by the transient response of the reactor power, reactor coolant, and the response of the RTS.

The high power range neutron flux and over temperature ΔT trip functions provide adequate protection over the entire range reactivity insertion rates for the uncontrolled control rod assembly bank withdrawal at power event. The analytical results indicate that the DNBR is always greater than the 95/95 limit and therefore, no fuel failures are predicted.

The reactor coolant pressure remains below 110% of the system design pressure, so the integrity of the pressure boundary is maintained.

This event does not lead to a more serious fault condition.

Table 15.4.2-1
Time Sequence of Events for Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (Minimum Feedback & 75 pcm/sec)

Event	Time (sec)
RCCA bank withdrawal begins	0.0
High power range neutron flux (high setpoint) analytical limit reached	1.4
Reactor trip initiated (rod motion begins)	2.0
Minimum DNBR occurs	3.1
Peak hot spot heat flux occurs	3.2

Table 15.4.2-2
Time Sequence of Events for Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (Minimum Feedback & 5.0 pcm/sec)

Event	Time (sec)
RCCA bank withdrawal begins	0.0
Over temperature ΔT analytical limit reached	19.5
Reactor trip initiated (rod motion begins)	25.5
Minimum DNBR occurs	25.9
Peak hot spot heat flux occurs	26.0

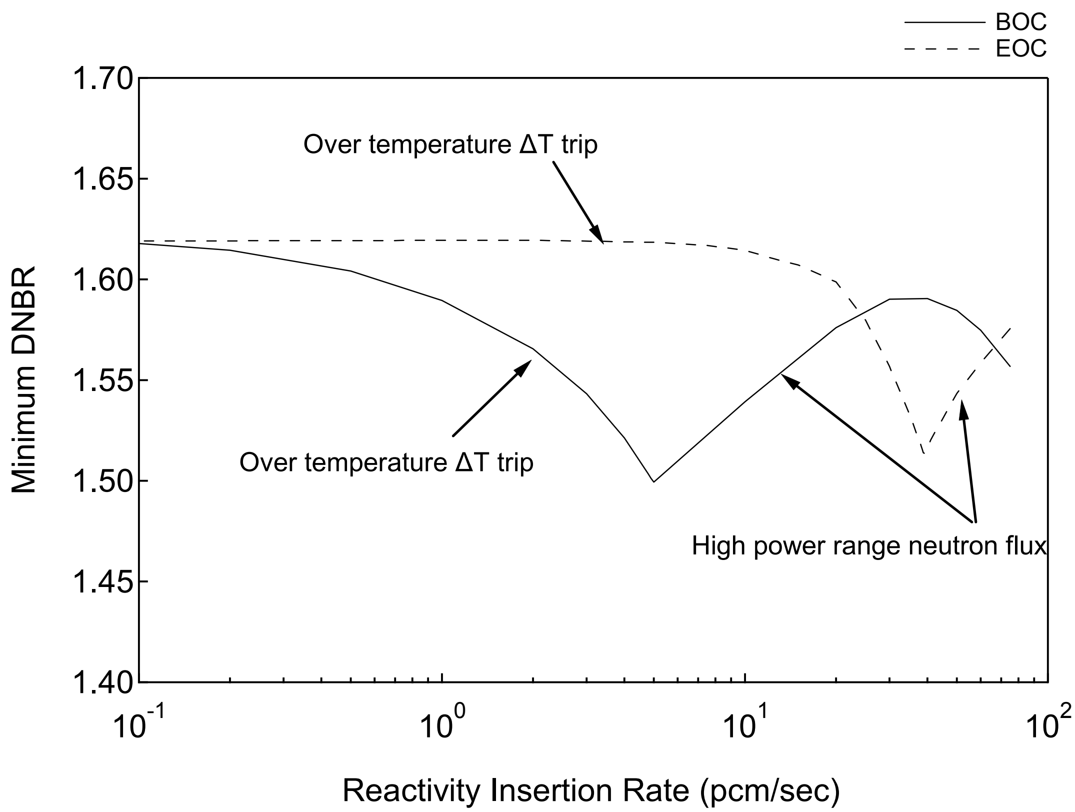


Figure 15.4.2-1 Minimum DNBR versus Reactivity Insertion Rate at HFP Uncontrolled Control Rod Assembly Withdrawal at Power

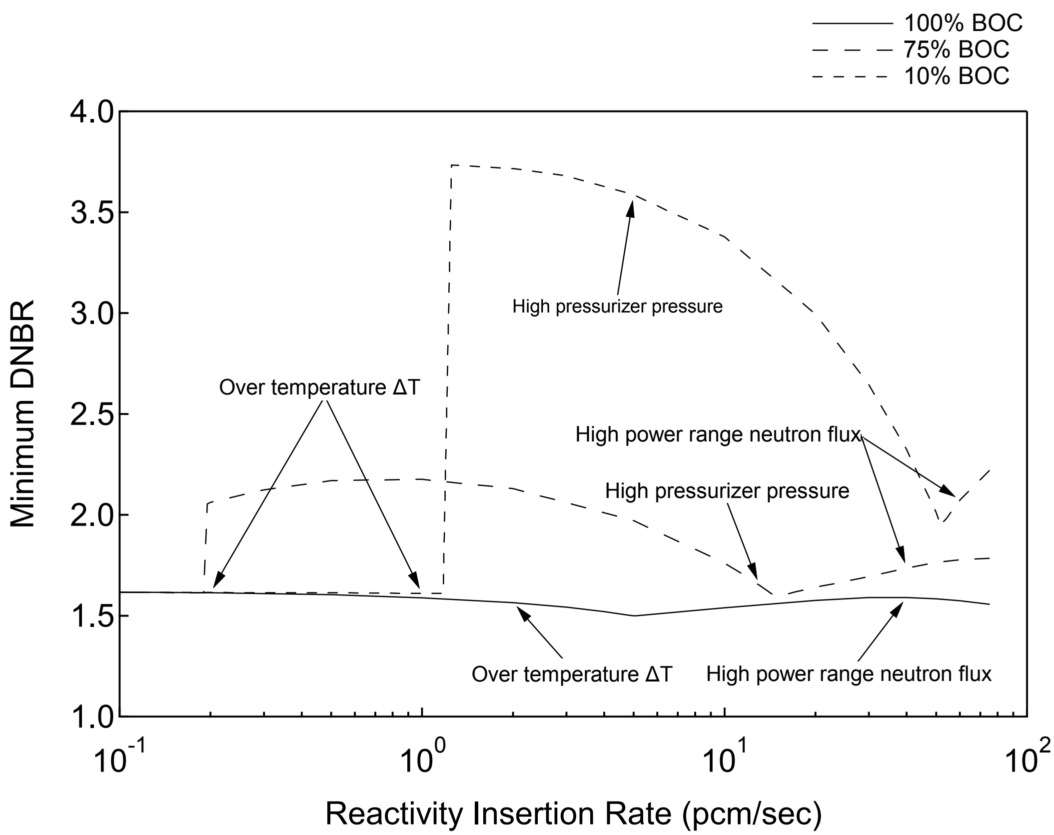


Figure 15.4.2-2 Minimum DNBR versus Reactivity Insertion Rate for Minimum Feedback Conditions for 10%, 75%, and 100% Power

Uncontrolled Control Rod Assembly Withdrawal at Power

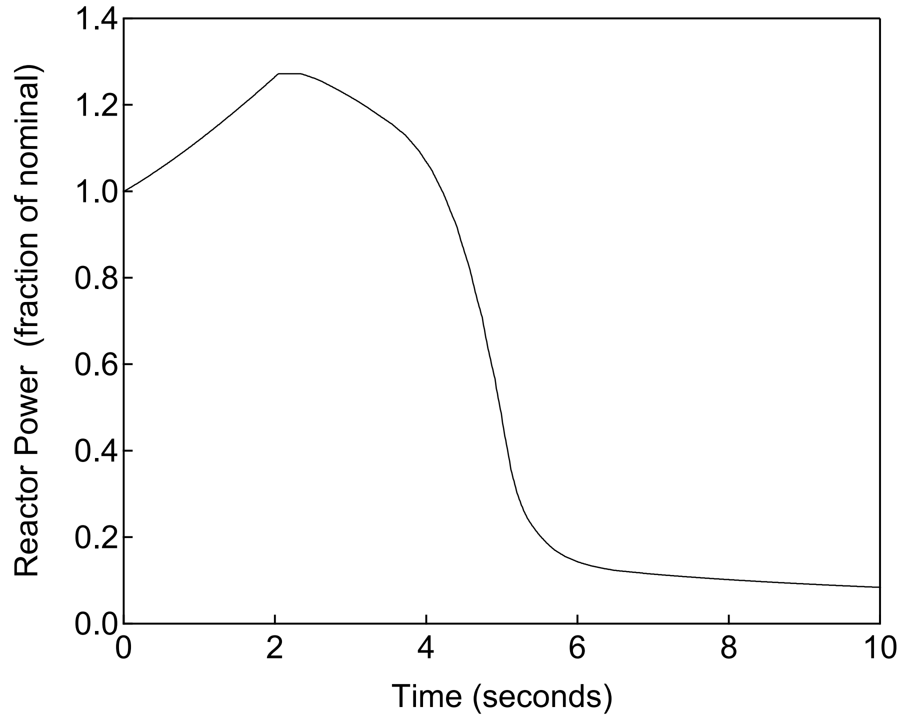


Figure 15.4.2-3 Reactor Power versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)

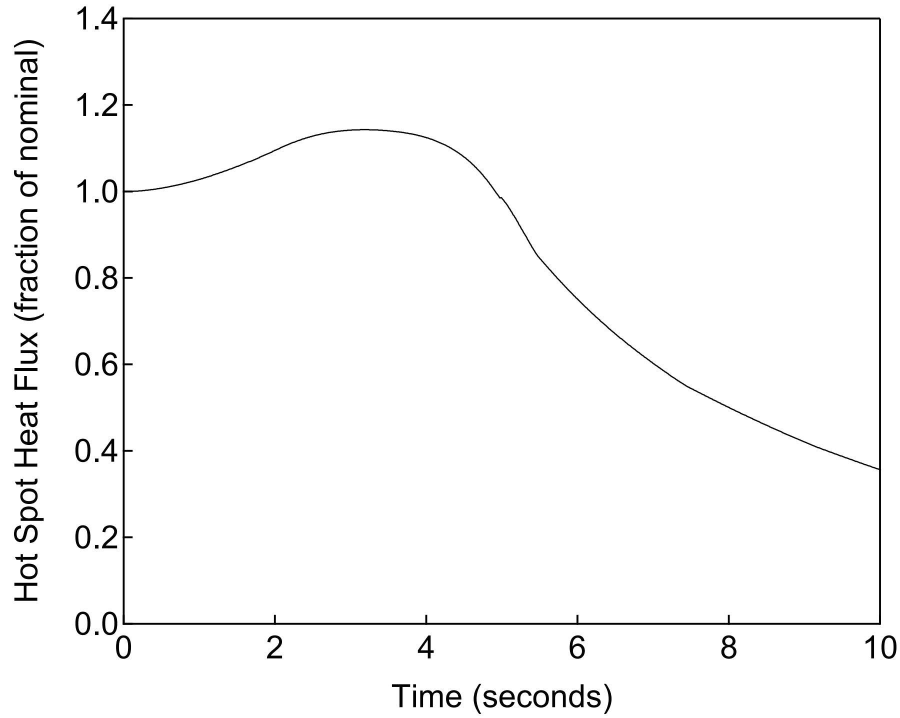


Figure 15.4.2-4 Hot Spot Heat Flux versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)

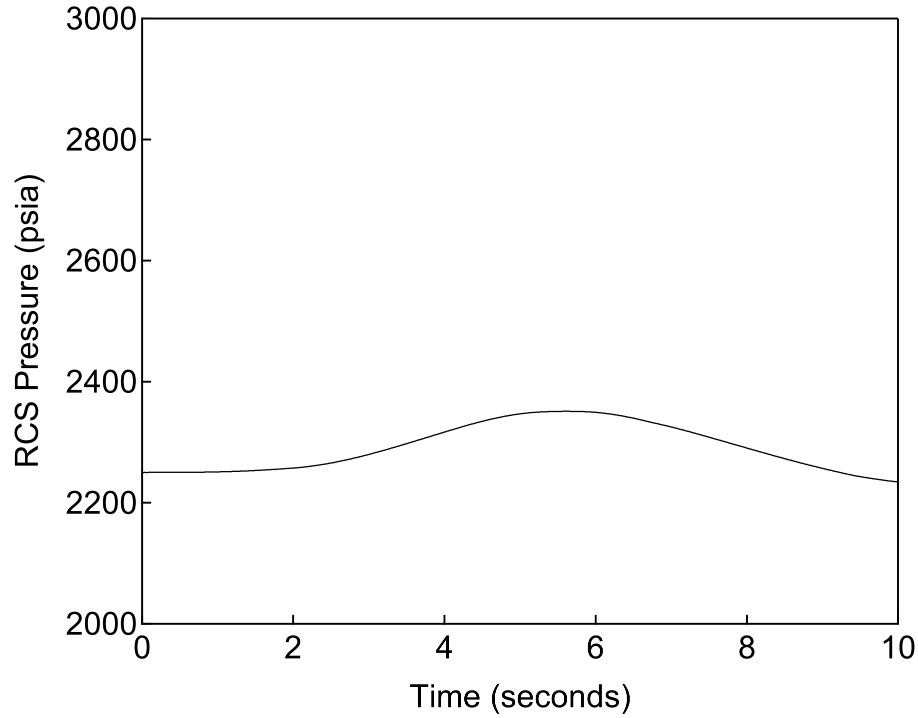


Figure 15.4.2-5 RCS Pressure versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)

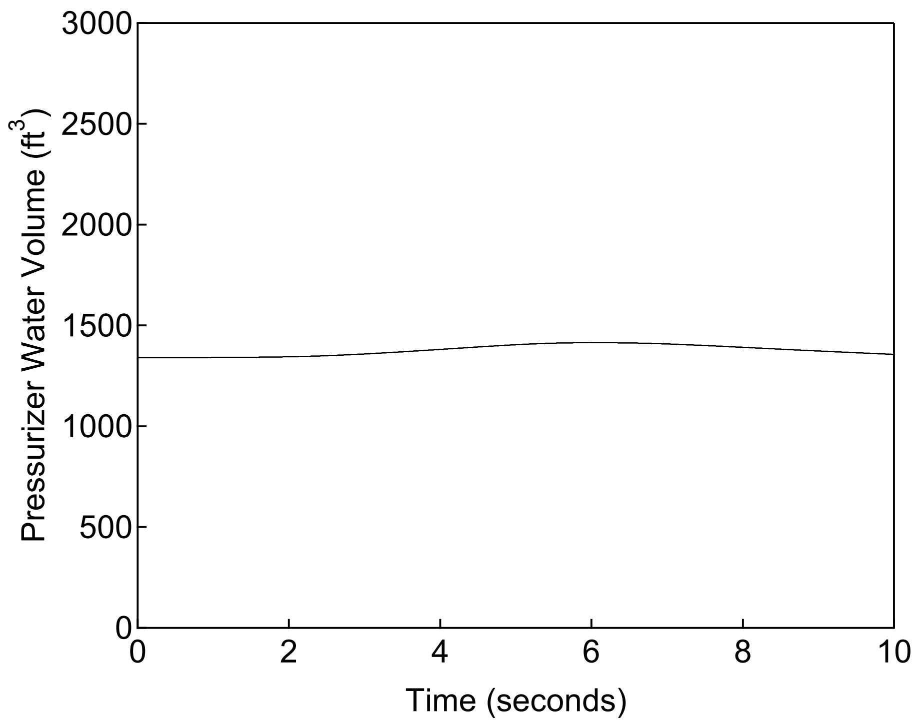


Figure 15.4.2-6 Pressurizer Water Volume versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)

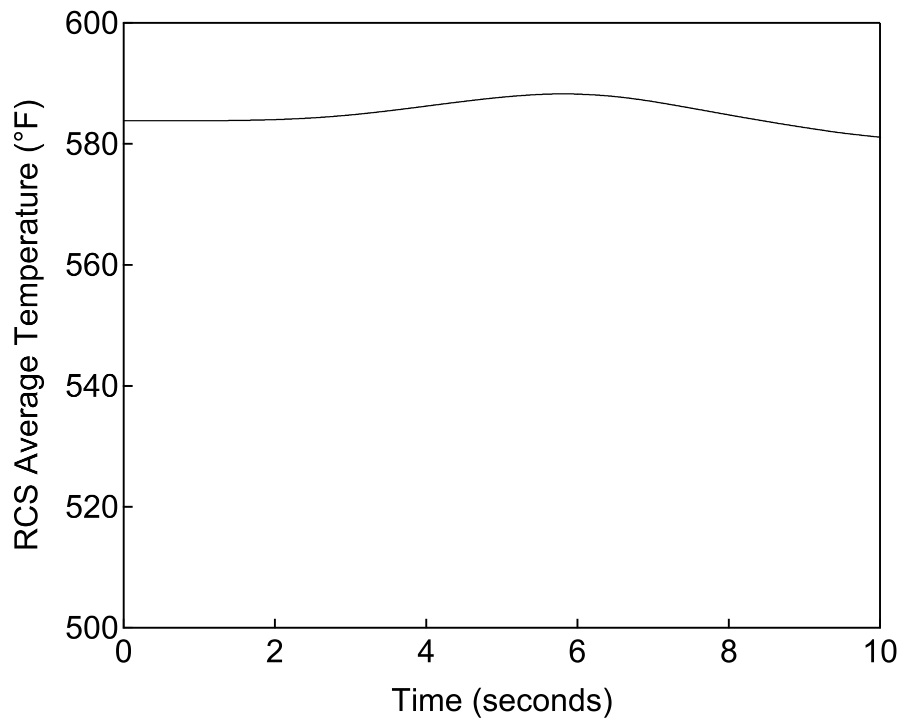


Figure 15.4.2-7 RCS Average Temperature versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)

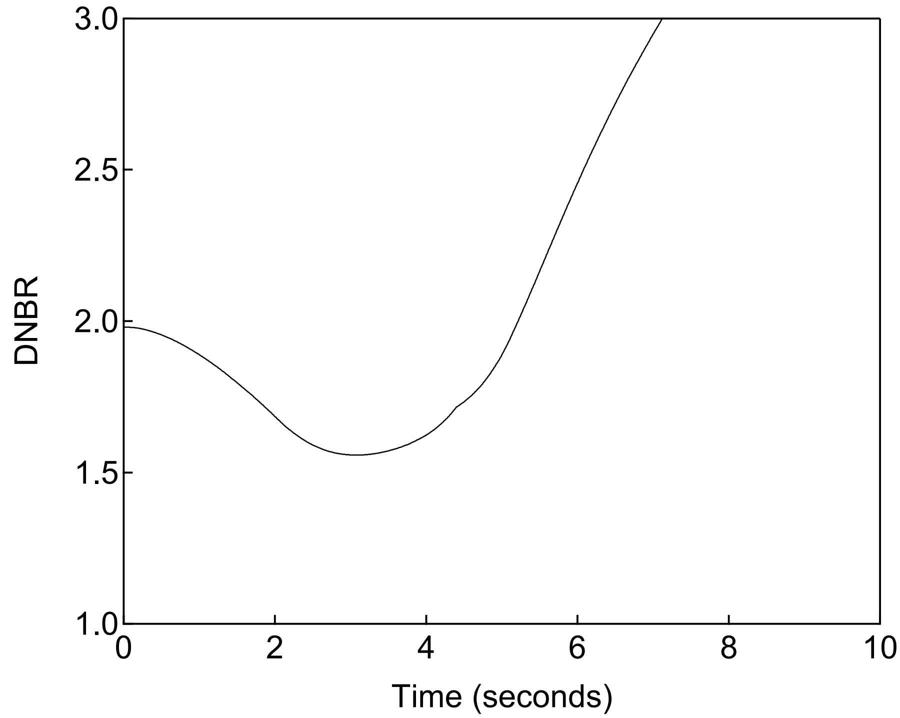


Figure 15.4.2-8 DNBR versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)

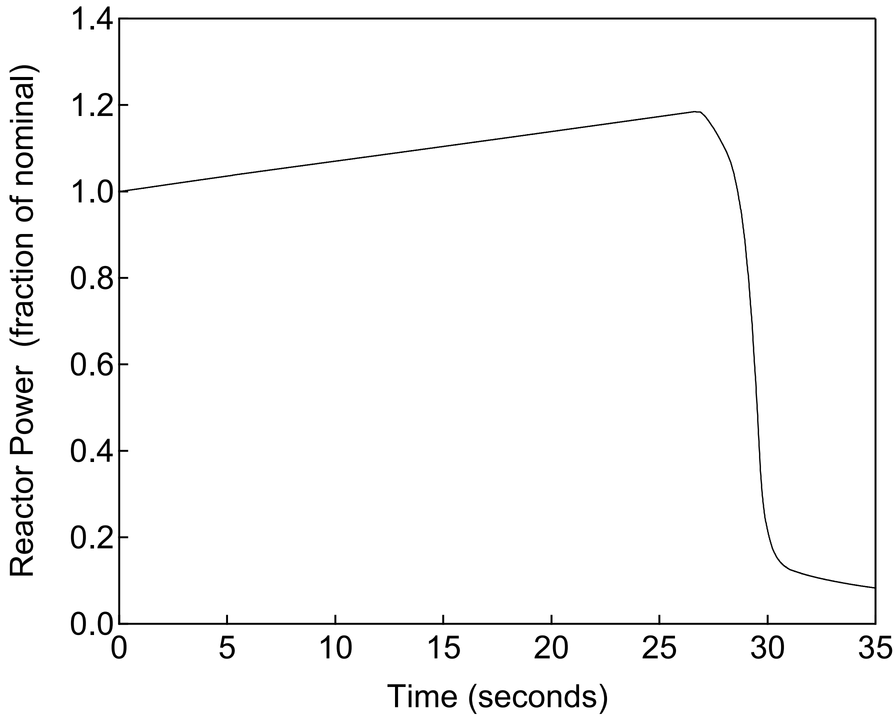


Figure 15.4.2-9 Reactor Power versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

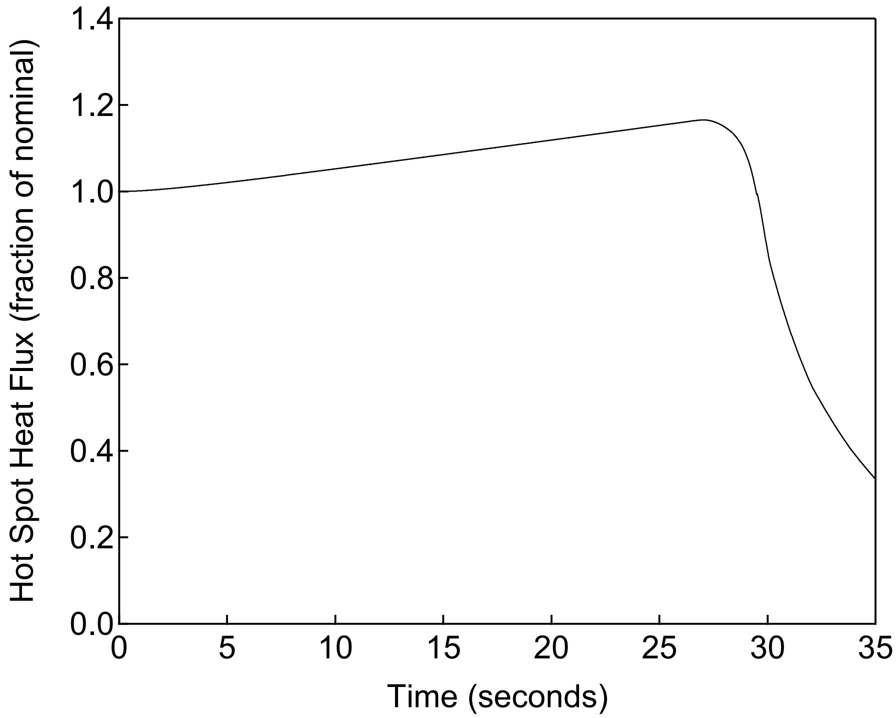


Figure 15.4.2-10 Hot Spot Heat Flux versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

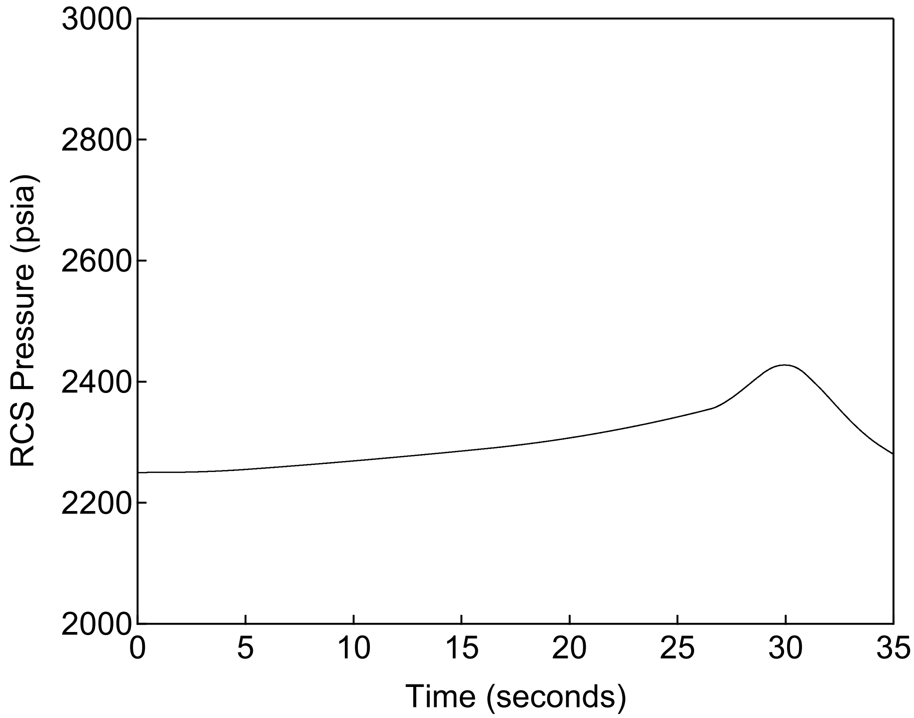


Figure 15.4.2-11 RCS Pressure versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

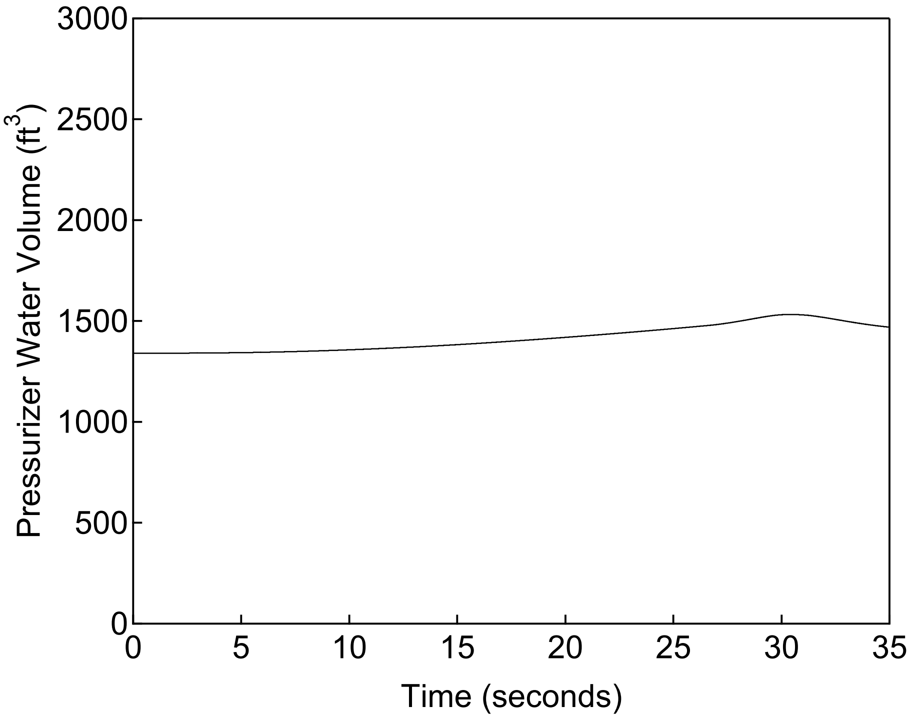


Figure 15.4.2-12 Pressurizer Water Volume versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

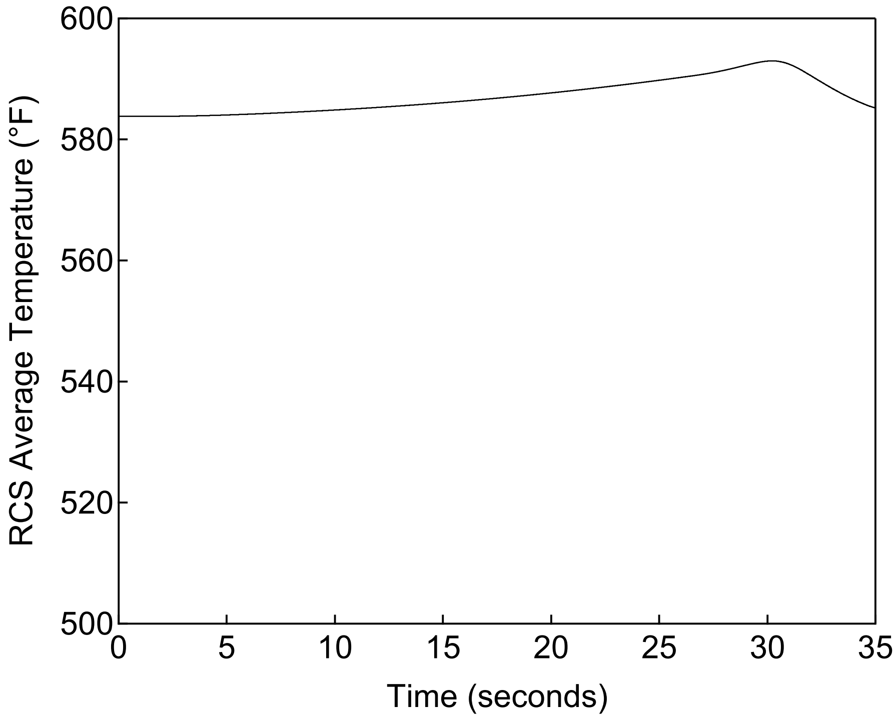


Figure 15.4.2-13 RCS Average Temperature versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

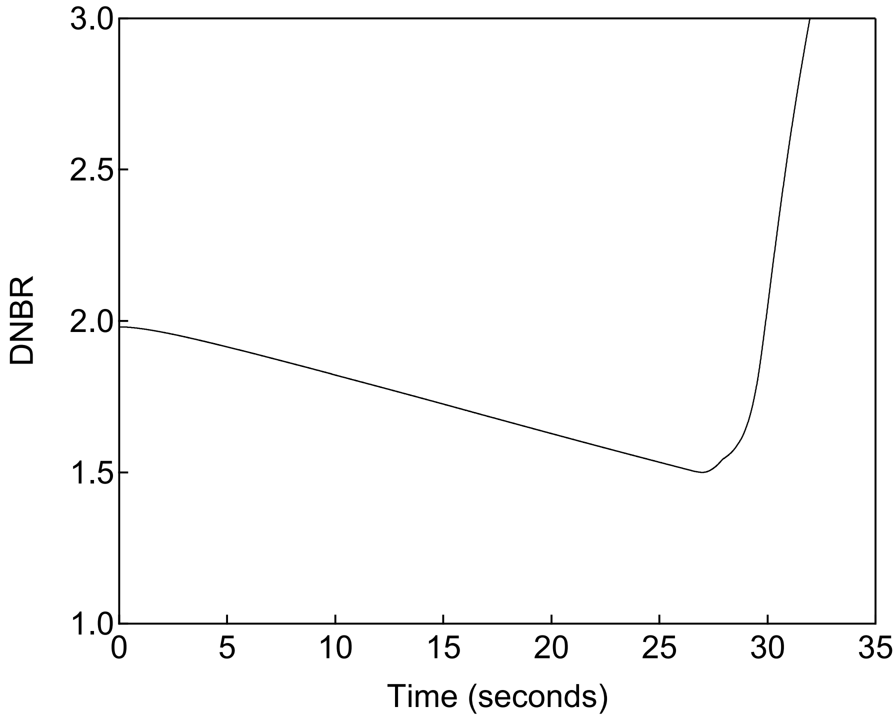


Figure 15.4.2-14 DNBR versus Time

Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Frequency Classification

Control rod misoperation includes:

- One or more dropped rod cluster control assemblies (RCCAs) within a group or bank
- One or more misaligned RCCAs (relative to their bank)
- Uncontrolled withdrawal of a single RCCA

Dropped or misaligned RCCAs could be caused by failures or malfunctions of an RCCA drive mechanism or RCCA drive mechanism control equipment.

Movement of a single RCCA is never performed during normal operations. However, the capability to move a single RCCA exists in order to restore a dropped RCCA to its correct position under strict administrative procedural control. Section 7.7.2.3 describes how no single equipment failure can cause uncontrolled withdrawal of a single RCCA. Each control bank RCCA is assigned to a bank, and all RCCAs in that bank are moved together in a pre-selected sequence such that a single failure cannot cause a single RCCA to withdraw. Given the design of the rod control system, assignment of RCCAs to groups, and strict administrative controls in place for restoration of a single RCCA to its proper insertion step, a single RCCA withdrawal can only be caused by multiple operator errors. This event is therefore considered more adverse and less probable than other anticipated operational occurrences (AOOs), and for this reason, MHI has not classified this event as an AOO. An internal MHI criterion to limit fuel damage to 5% of the core in the associated radiological consequence analysis reflects that higher allowable consequences typically allowed for other postulated accidents (PAs) historically defined as design basis events are not appropriate. The RCCA misoperation transient evaluated in this section involving uncontrolled withdrawal of a single RCCA is withdrawal of a single RCCA outside of the dropped RCCA recovery procedure.

The dropped and misaligned RCCA misoperation events are classified as AOOs. Historically, these events have been classified as Condition II events of moderate frequency as defined in ANSI N18.2 (Ref. 15.4-1). The uncontrolled withdrawal of a single RCCA is classified by MHI as a PA with an accident-specific acceptance limit of 5% failed fuel, reflecting that it is less probable and more consequential than an AOO but more probable and less consequential than a design basis event or limiting PA. Historically, the single RCCA withdrawal classified as a Condition III infrequent event as defined in ANSI N18.2 (Ref. 15.4-1). Event frequency conditions are described in Section 15.0.0.1.

SRP 15.4.3 provides the following additional event-specific acceptance criteria:

- Fuel centerline temperatures do not exceed the melting point.
- Uniform cladding strain as defined in SRP 4.2 does not exceed 1%.

15.4.3.2 Sequence of Events and Systems Operation

The RCCAs are arranged in groups and banks (two groups per bank). The two groups are always at essentially the same position. The banks are withdrawn and inserted in a fixed sequence (with never more than two banks moving at any one time).

An RCCA drop is an occurrence where one or more RCCAs drop from their withdrawn position. An RCCA drop causes a local reactor power reduction due to a neutron flux decrease around that RCCA. This, in turn, causes an operational disturbance of the reactor (a decrease in reactor power and/or a decrease in average reactor coolant temperature).

The sequence and timing of major events for the one or more dropped RCCAs with automatic rod control event is described in the results section.

A number of direct and indirect means are available in the main control room to detect a dropped RCCA. Direct means include a rod position deviation alarm, rod position indication, or if the RCCA is fully inserted, a rod bottom signal. Indirect means include asymmetric axial or radial power indication from the excore detectors, a decrease in reactor power, or sudden unexpected automatic rod motion.

If the dropped RCCAs are not detected and corrective action taken, other RCCAs could be withdrawn to compensate for the reactivity decrease caused by the dropped RCCAs (to restore reactor power and/or average coolant temperature to match the turbine demand). The dropped rod event is therefore analyzed assuming automatic rod control. If RCCAs were withdrawn, the reactor power is restored which results in the increase of the hot channel heat flux relative to the hot channel heat flux prior to the dropped rod. This increase in hot channel heat flux could lead to a reduction in the reactor safety margin.

RCCA misalignment could occur if a fault in the control system causes a single RCCA or the RCCAs in a bank to be moved out of sequence from the pre-programmed sequence. (e.g., not all of the RCCAs in a group move at the same speed or the two groups in a bank do not move at the same speed). Similar to dropped rods, misaligned rods may be directly or indirectly detected in the control room by a rod position deviation alarm, differences in rod positions for RCCAs within a bank, or asymmetric axial or radial indication from the excore detectors. If misaligned RCCAs are not detected and corrective action taken, the core power distribution could exceed the design power distribution, resulting in a reduction of margin to the fuel design limits.

A single RCCA withdrawal can result in a core power response similar to a bank withdrawal at power with a concurrent adverse change in power distribution that can exceed the design power distribution shape and peaking factors, resulting in violating fuel design limits before the event is terminated by the reactor trip system (RTS).

The following automatic reactor trip signals are assumed to be available to provide protection from these transients:

- High power range neutron flux rate
- High intermediate range neutron flux
- High power range neutron flux
- Low pressurizer pressure
- Over temperature ΔT
- Over power ΔT

The limiting single failure for this event is the failure of one train of the RTS. Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

A transient analysis is performed for the dropped RCCA event to verify that the RTS protects the DNBR fuel design limits. The DNBR analysis uses a nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$) that bounds the dropped rod power distribution.

15.4.3.3 Core and System Performance

15.4.3.3.1 One or More Dropped RCCAs within a Group or Bank

15.4.3.3.1.1 Evaluation Model

An evaluation model based on the MARVEL-M plant transient analysis code is used to obtain transient responses of reactor power, reactor coolant pressure, and reactor coolant temperature following a dropped RCCA. The model simulates the RTS trips setpoints and delays, the automatic insertion of trip reactivity, the insertion of negative reactivity prior to trip to simulate the dropped rod, and point reactor kinetics. This evaluation model is described in Section 15.0.2.2.1. Additional details on the MARVEL-M code are provided in Reference 15.4-2.

Various combinations of dropped RCCA locations and rod worths are identified and modeled using standard steady state nuclear design codes to calculate a bounding nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$) for use in the DNBR calculation. The calculated $F_{\Delta H}^{N}$ is used to evaluate the hot spot heat flux, which is used in the transient analysis using MARVEL-M to analyze the minimum DNBR.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

15.4.3.3.1.2 Input Parameters and Initial Conditions

A transient analysis is performed for the dropped RCCA event to verify that the minimum DNBR meets the 95/95 design limit. The following assumptions are made for this analysis:

- Consistent with use of the RTDP, the initial values of reactor power, reactor coolant average temperature, and reactor coolant pressure are assumed to be the nominal full power values without uncertainties provided in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to have the minimum feedback limit shown in Figure 15.0-2. This combination of reactivity coefficients maximizes the heat flux in the initial stage of the transient. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. The inserted reactivity used to simulate the dropped RCCA is 0.25 %Δk/k, which is greater than the maximum reactivity insertion resulting from one RCCA dropping from the fully withdrawn position to the fully inserted position during rated-power operation. It is inserted instantaneously.

- The rod control system is assumed to be in the automatic control mode. This assumption results in rod withdrawal to match turbine load, which maximizes reactor power and core heat flux. Due to this assumption, no reactor trips occur.

- Pressurizer heaters are assumed to be off. This minimizes reactor coolant system (RCS) pressure which is conservative for the DNBR calculation.

- For the minimum DNBR analysis, the nuclear enthalpy rise hot channel factors ($F_{\Delta H}^{N}$) subsequent to the RCCA drop is 1.90. This bounding constant value is used for the entire transient. It is calculated using steady state nuclear analysis design calculations that consider various dropped RCCA locations and associated RCCA worths.

15.4.3.3.1.3 Results

Table 15.4.3-1 lists the key events and the times at which they occur, relative to the initiation of the transient, for the Core and System Performance DNBR Evaluation for the dropped RCCA transient with automatic rod control.

Figures 15.4.3-1 through 15.4.3-6 are plots of key system parameters versus time for the Core and System Performance Evaluation of the dropped RCCA transient with automatic rod control. Following the instantaneous insertion of negative reactivity corresponding to the dropped RCCA, the reactor power, core heat flux, and core average temperature decrease. The automatic rod control system withdraws the control rods to compensate, resulting in a slight power overshoot followed by a return to steady state at the initial power. No reactor trips are predicted for this event; the sequence of events and parameter plots reflect reaching a new steady state condition without a reactor trip. The minimum DNBR slightly decreases below the initial value during the transient, but remains well above the 95/95 limit.

If the control rod control system is in manual control and no operator action is taken, the reactor coolant pressure would continue to decrease until the reactor is tripped automatically on low pressurizer pressure.

The DNBR remains above the 95/95 limit, and no fuel failures are predicted. A plot of fuel centerline temperature is not provided because fuel integrity is verified by demonstrating that the minimum DNBR remains above the 95/95 limit.

15.4.3.3.2 One or More Misaligned RCCAs (relative to their bank)

15.4.3.3.2.1 Evaluation Model

No nuclear steam supply system transient is analyzed for this event, and therefore, the MARVEL-M code is not used.

Various static rod misalignment scenarios (single rods or groups within banks) are identified and modeled using standard steady state nuclear design codes to calculate a bounding nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$) for use in the DNBR calculation.

The VIPRE-01M code is used to determine the DNBR. DNBR is calculated using the RTDP. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties. The VIPRE-01M evaluation model is described in Section 15.0.2.2.2. The Reference 15.4-3 provides additional description of the VIPRE-01M DNBR coolant channel model and its application in calculating DNBR for Chapter 15 events.

15.4.3.3.2.2 Input Parameters and Initial Conditions

- Consistent with use of the RTDP, the initial values of reactor power, reactor coolant average temperature, and reactor coolant pressure are assumed to be the nominal full power values without uncertainties provided in Table 15.0-3.

- Various static rod misalignment scenarios are identified and modeled for the purpose of defining the limiting nuclear enthalpy rise hot channel factor for use in the DNBR channel analysis. Scenarios considered include, but are not limited to, a single RCCA fully inserted, one RCCA fully withdrawn with the remaining bank RCCAs at their insertion limits, and other intermediate misalignment conditions. The limiting RCCA misalignment for the analysis is assumed to be one RCCA fully withdrawn with the remaining RCCAs in the bank at their insertion limit.

- For the minimum DNBR analysis, the nuclear enthalpy rise hot channel factors ($F_{\Delta H}^{N}$) subsequent to the RCCA static misalignment is 1.90.

15.4.3.3.2.3 Results

No transient analysis is performed for this event. The DNBR analysis is performed at the initial condition of nominal power, pressure, and coolant temperature without uncertainties applied consistent with the RTDP methodology using an event-specific $F_{\Delta H}^{N}$. Based on the limiting nuclear enthalpy rise hot channel factor $F_{\Delta H}^{N}$ of 1.90, the minimum DNBR is above the 95/95 limit. The hot spot linear heat generation rate associated with this condition (peak radial peaking factor associated with the statically misaligned RCCA in conjunction with the design F_Z) is well below the threshold value where fuel melting begins. Therefore, no fuel failures are predicted and a plot of fuel centerline temperature is not provided.

15.4.3.3.3 Uncontrolled Withdrawal of a Single RCCA

15.4.3.3.3.1 Evaluation Model

No event-specific nuclear steam supply system transient is analyzed for this event, and therefore, the MARVEL-M code is not used. However, it is assumed that the overall core response is the same as that for the spectrum of RCCA withdrawal transients analyzed in Section 15.4.2, only with a bounding nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$) characteristic of a single RCCA withdrawal event.

Various combinations of single RCCA locations and rod worths are identified and modeled using standard steady state nuclear design codes to calculate a bounding $F_{\Delta H}^{N}$ for use in the DNBR calculation.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

The approach used for the rods in DNB calculation is discussed in Section 15.4.8.

15.4.3.3.3.2 Input Parameters and Initial Conditions

- Consistent with the use of RTDP, the initial values of reactor power, reactor coolant temperature, reactor coolant rate, and reactor coolant pressure are assumed to be the nominal values without uncertainties provided in Table 15.0-3.

- Various combinations of single RCCA locations and rod worths are identified and modeled using standard steady state nuclear design codes to calculate a bounding nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$) for use in the DNBR calculation.

- For the minimum DNBR analysis, the nuclear enthalpy rise hot channel factor

($F_{\Delta H}^{N}$) subsequent to the single RCCA withdrawal is 1.90.

15.4.3.3.3.3 Results

The changes in overall core parameters, (core power, RCS temperature, and RCS pressure) due to this event are bounded by those for the transient associated with an uncontrolled withdrawal of a bank of RCCAs, which is described in Section 15.4.2.3.

The rods in DNB analysis confirmed that the number of rods predicted to be in DNB is less than 5% of the core, the value used in the radiological consequence analysis. In this section, a conservative analysis method is applied, which results in the minimum DNBR being below the 95/95 limit. However, using a more detailed analysis method, the minimum DNBR is above the 95/95 limit, and no fuel failures are predicted.

15.4.3.4 Barrier Performance

15.4.3.4.1. One or More Dropped RCCAs within a Group or Bank

The dropped RCCA event does not result in exceeding any reactor coolant pressure boundary or containment volume fission product barrier design limits. The results of the Core and System Evaluation case demonstrate that the reactor coolant system pressure remains well below 110% of system design pressure. In addition, the main steam pressure cannot challenge the main steam system pressure design limit. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.4.3.4.2 One or More Statically Misaligned RCCAs

This transient involves only a change in the core power distribution. Overall core power, RCS flow, and RCS pressure are not changed. Therefore, the maximum reactor coolant pressure remains well below 110% of the design pressure. The integrity of the reactor coolant pressure boundary is maintained.

15.4.3.4.3 Uncontrolled Withdrawal of a Single RCCA

The effects of this event on core pressure are bounded by those for the RCCA bank withdrawal at power described in Section 15.4.2.4. The maximum reactor coolant pressure for that bounding transient remains well below 110% of the design pressure. Therefore the integrity of the reactor coolant pressure boundary for this single RCCA withdrawal transient is maintained.

15.4.3.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the rod ejection accidents evaluated in Section 15.4.8.5.

15.4.3.6 Conclusions

For the dropped RCCA and statically misaligned RCCA events evaluated in Section 15.4.3.3, no fuel failures are predicted and are no radiological releases. Therefore, these AOOs have no radiological consequences. Since the fuel integrity has been verified by the DNB analysis peak fuel centerline temperature and clad strain are not evaluated.

For the uncontrolled withdrawal of a single RCCA, the Core and System Performance analysis confirms that the number of rods predicted to be in DNB is less than 5% of the core, which is the value used in the radiological consequence analysis. Additionally, the radiological consequences of this event are substantially less than that of the postulated rod ejection accidents described in Section 15.4.8.

The reactor coolant pressure remains below 110% of the reactor coolant system design pressure so the integrity of the reactor coolant pressure boundary is maintained for all of the RCCA misoperation transients described in this section.

The control rod misoperation events do not lead to a more serious fault condition.

Table 15.4.3-1
Time Sequence of Events for One or More Dropped RCCAs with Automatic Rod Control - DNBR Analysis

Event	Time (sec)
Malfunction causes an RCCA drop	0.0
Rod control system initiates rod withdrawal	0.1
Peak reactor power occurs	37.7
Minimum DNBR occurs	38.0

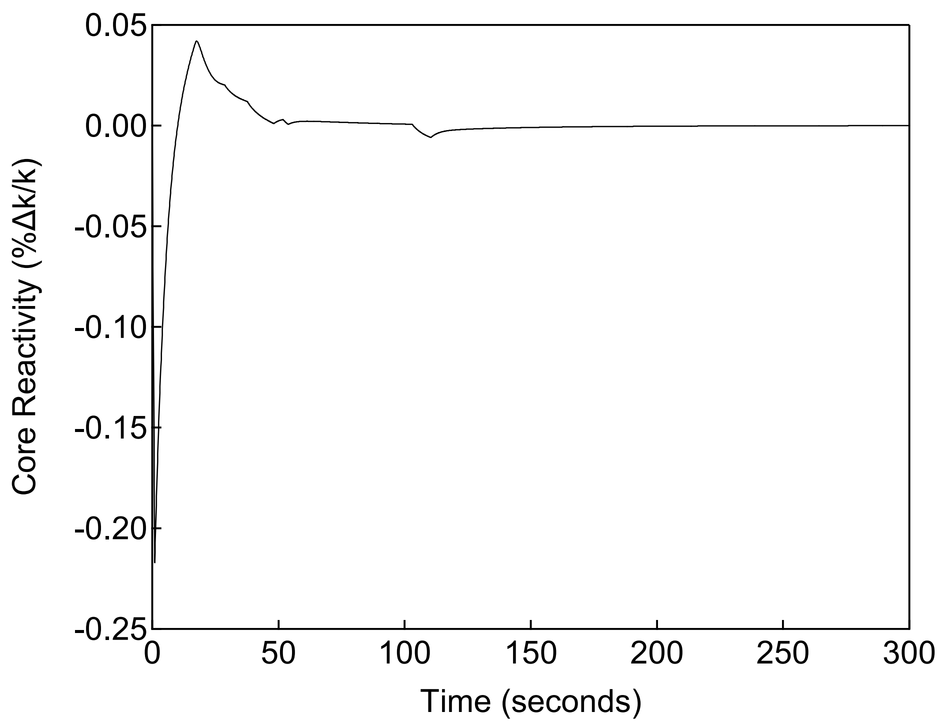


Figure 15.4.3-1 Core Reactivity versus Time

One or More Dropped RCCAs within a Group or Bank

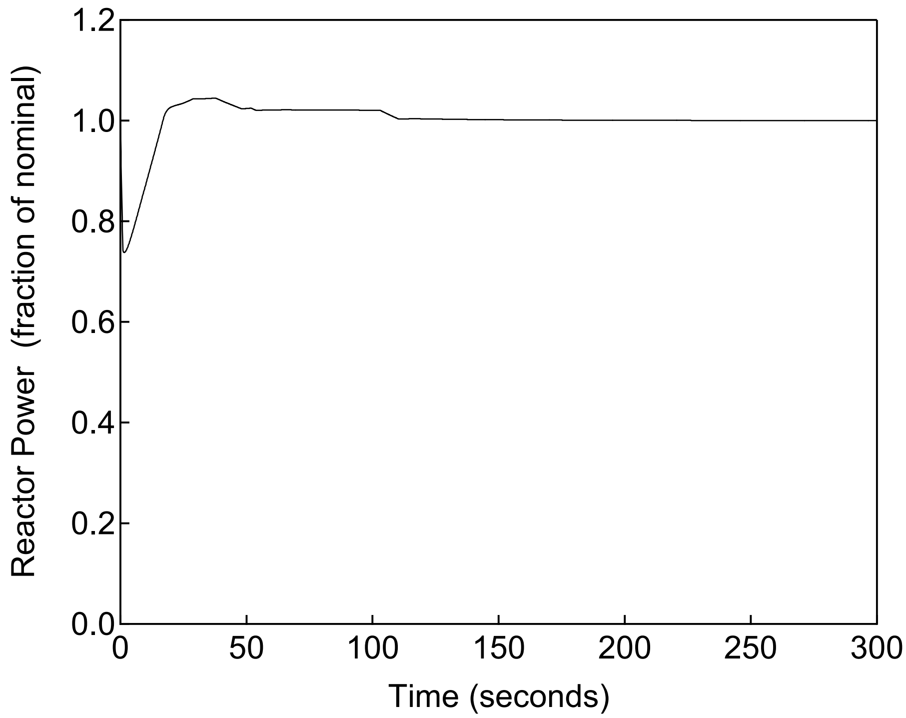


Figure 15.4.3-2 Reactor Power versus Time

One or More Dropped RCCAs within a Group or Bank

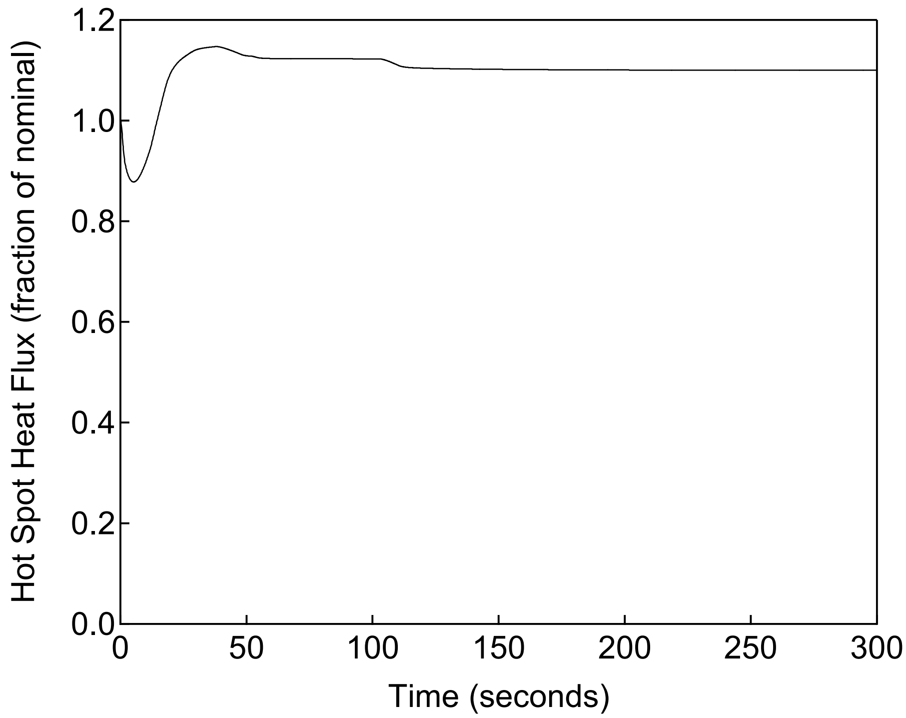


Figure 15.4.3-3 Hot Spot Heat Flux versus Time

One or More Dropped RCCAs within a Group or Bank

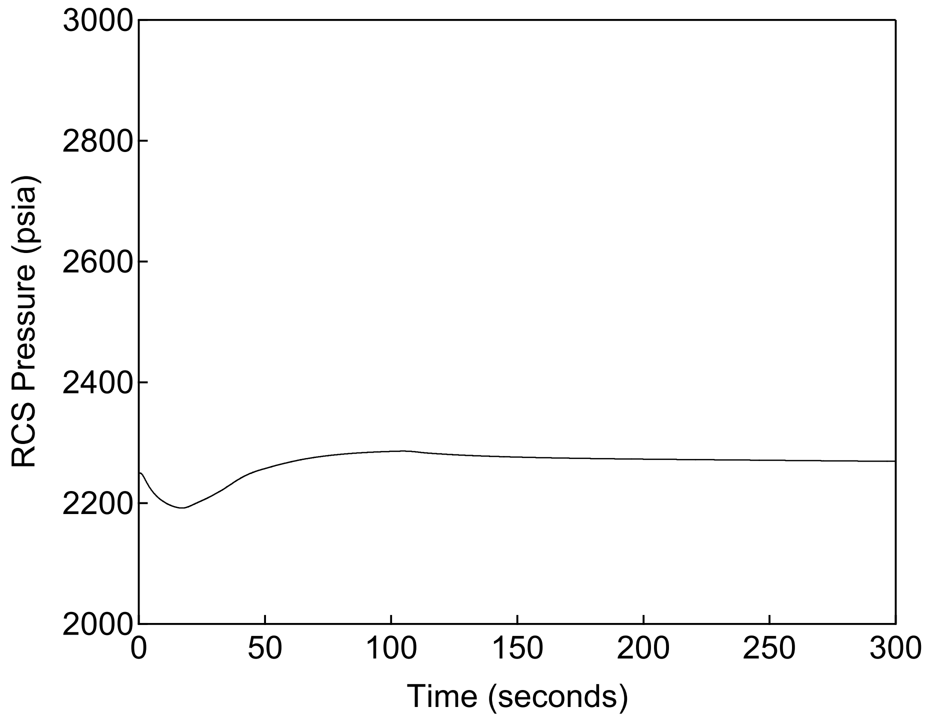


Figure 15.4.3-4 RCS Pressure versus Time

One or More Dropped RCCAs within a Group or Bank

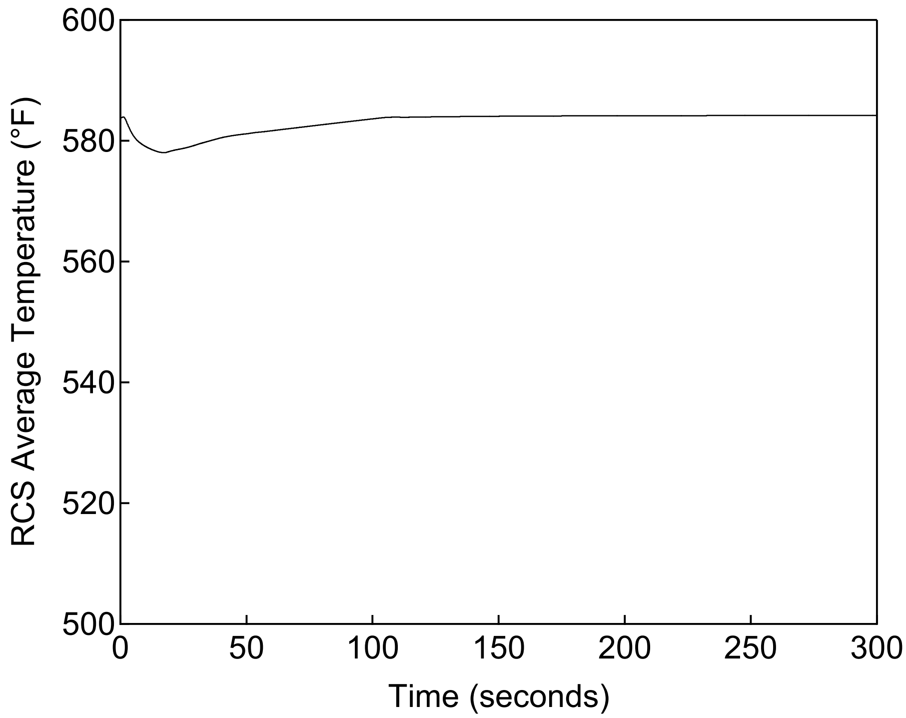


Figure 15.4.3-5 RCS Average Temperature versus Time

One or More Dropped RCCAs within a Group or Bank

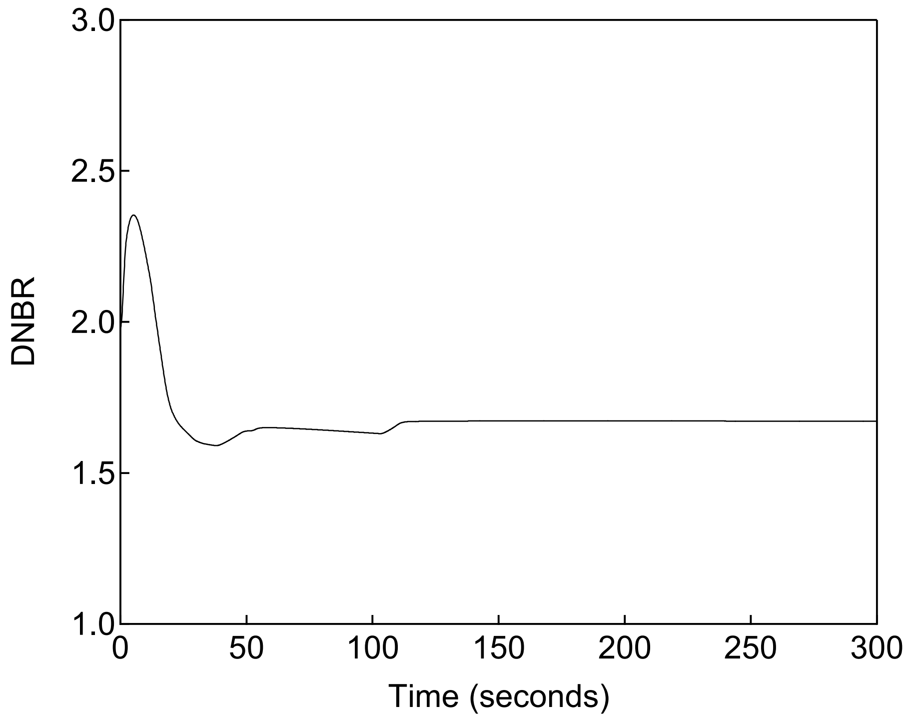


Figure 15.4.3-6 DNBR versus Time

One or More Dropped RCCAs within a Group or Bank

15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature

This section is not applicable to the US-APWR, because power operation with an inactive loop is not allowed by the Technical Specifications.

15.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

This BWR section is only applicable to BWRs and is not applicable to the US-APWR.

15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System

15.4.6.1 Identification of Causes and Frequency Classification

An inadvertent decrease of the boron concentration in the reactor coolant can occur due to the addition of low-boron-concentration water into the reactor coolant due to a malfunction or improper operation of the chemical and volume control system (CVCS). This transient results in a positive reactivity addition to the core.

Inadvertent decrease in boron concentration in the reactor coolant is classified as an anticipated operational occurrence (AOO). Historically, these events have been classified as Condition II events of moderate frequency as defined in ANSI N18.2 (Ref. 15.4-1). Event frequency conditions are described in Section 15.0.0.1.

In addition to the general AOO acceptance criteria described in Section 15.0.0.1.1, SRP 15.4.6 imposes additional guidance that the following minimum time intervals be available for operator actions between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost (criticality):

- Refueling: 30 minutes
- Startup, shutdown, hot standby, and power operation: 15 minutes

15.4.6.2 Sequence of Events and Systems Operation

The inadvertent decrease in reactor coolant boron concentration (i.e., boron dilution) event is evaluated during all modes of operation including refueling conditions, shutdown conditions, the beginning of reactor startup, and power operation.

Table 15.4.6-1 provides a summary of the operating parameters and conditions associated with each mode of operation for the boron dilution event for the US-APWR. As can be seen from the table, the dilution flow and dilution volume are the same for all modes.

If the reactor is operating at power (Mode 1) under automatic rod control when the boron dilution is initiated, rod cluster control assemblies (RCCAs) will be automatically inserted into the core in order to compensate for the positive reactivity addition caused by the boron dilution. Unanticipated rod stepping can be recognized in the control room and confirmed by observing the rod position indications, and in the limiting case, an alarm may actuate when the rods reach their respective rod insertion limit. If the unplanned dilution continues after this alarm, the reactor is tripped and returns to criticality under hot zero power (HZP) conditions.

If the reactor is operating at power (Mode 1) under manual rod control, reactivity added by the decrease in boron concentration causes an increase in the reactor power. This will increase the heat flux and reactor coolant temperature, which will reduce the DNBR margin. The resulting transient is similar to the uncontrolled RCCA bank withdrawal at power transient described in Section 15.4.2 and results in a reactor trip if no action is taken to prevent the uncontrolled dilution. If such a reactor trip occurs, the reactor trip alerts the operator to the unplanned reactivity addition. If the unplanned dilution continues after the reactor is tripped, the reactor may return to criticality under HZP conditions.

If the plant is in a normal start-up operation (Mode 2), the reactivity is added by manual (planned) dilution and manual withdrawal of the control rods. An inadvertent dilution during this operation could result in a power escalation and high source range neutron flux reactor trip before the operator manually blocks the source range reactor trip.

For the cases where the reactor is in hot standby or shut down (Modes 3, 4, & 5), the reactivity added by the decrease in boron concentration could cause a reduction in shutdown margin, leading to criticality.

The potential for boron dilutions during refueling (Mode 6) or during shutdown operation with no reactor coolant pumps (RCPs) running (Modes 4 & 5) does not exist due to strict administrative controls. Therefore, no quantitative analysis is performed.

Boron dilution is carried out by adding a predetermined batch of pure water into the reactor coolant system. When the predetermined amount of pure water has been added, the makeup water valve is automatically closed, so that boron dilution beyond the predetermined limit will not occur. When carrying out a boron dilution, the operator performs two operations: (1) changing from the automatic makeup mode to the dilution mode and (2) operating the start switch. Dilution cannot start unless both of these steps are performed. The requirement for two distinct actions reduces the likelihood of inadvertent dilution caused by operator action. For the US-APWR, planned boron dilutions are under strict administrative control.

The CVCS design inherently limits the maximum boron dilution rate so boron dilution transients proceed relatively slowly. This slow rate, together with the alarms and trips (described above) that provide operator indication ensure that sufficient time exists so that reactivity transients can be terminated by manual action to prevent criticality or a return to criticality. The alarms and trips described above are in addition to other parameters continuously available for monitoring in the control room, such as neutron flux and RCCA bank position.

If a fault in the chemical and volume control system causes the borated water or pure water flow rate to deviate from the setpoint flow rate, the operator is warned by a flow rate high deviation alarm and the water makeup isolation valve is automatically closed to stop makeup to the reactor coolant system (Section 7.6.1.3).

For the case where the event is initiated from at-power or start-up conditions, the following automatic reactor trip signals are assumed to be available to provide protection from this transient:

- High power range neutron flux (high/low setting)

- Over temperature ΔT

For the case where the event is initiated from subcritical conditions, the high source range neutron flux reactor trip signal is assumed to be available to provide protection from this transient:

The plant does not have an automatic system to detect and terminate a boron dilution event. Operator action is required to terminate dilution flow.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause an RCP coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

15.4.6.3 Core and System Performance

15.4.6.3.1 Evaluation Model

No transient calculations are performed for the nuclear steam supply system response using the MARVEL-M code. These transients are evaluated by calculating the minimum time available for the operator to detect the reactivity addition and to take corrective action. These calculations are done using boron and water mass equilibrium equations. Separate calculations are performed for the cases defined by Table 15.4.6-1. Quantitative analysis is not performed for Mode 6 or Modes 4 & 5 with no RCPs running.

The primary focus of the analysis is to confirm the operator will have sufficient control room indication and/or sufficient time to detect and diagnose the inadvertent boron dilution and to take corrective action to prevent criticality.

The acceptance criterion for these calculations is that the minimum calculated time available for operator action is greater than the minimum allowed time interval described in Section 15.4.6.1.

The basic calculation technique is as follows:

(a) Define the initial concentration of boron in the reactor coolant (C1). For the at-power and start-up cases, the initial critical boron concentration is based on the control rods being inserted to the insertion limits. For the hot standby and shutdown cases, the initial boron concentration is the mode specific concentration that will result in the Technical Specification required shutdown margin with all RCCAs fully inserted except for the most reactive single RCCA, which is assumed to be fully withdrawn.

(b) Define the concentration of boron in the reactor coolant that will result in criticality (C2). For the at-power and start-up cases, the critical boron concentration is based on the critical condition at HZP. For the shutdown cases, the critical boron concentration is based on the assumption that the reactor is critical with all RCCAs fully inserted except for the most reactive single RCCA, which is assumed to be fully withdrawn.

(c) Determine conservative values for (1) the volume of water in the reactor coolant system (RCS) (dilution volume), (2) the rate at which low-boron-concentration water is added to the RCS (dilution flow); and (3) the boron concentration in the low-boron-concentration water.

(d) Calculate the time required for the RCS boron concentration to be reduced from C1 to C2. For Mode 1 with automatic rod control, hot standby, hot shutdown, and cold shutdown (Modes 3, 4, & 5) this time is the duration between the annunciator alarm and criticality or return to criticality. For the Mode 1 with manual rod control and start-up cases, this is the period of time between the start of the boron dilution and a return to criticality.

In these calculations, reactor coolant and dilution water is assumed to be mixed completely.

15.4.6.3.2 Input Parameters and Initial Conditions

For cases where the reactor is at power or start-up (Modes 1 & 2), the immediate response to the dilution depends on whether the rod control system is in automatic or manual operation. The continuous reactivity insertion by the boron dilution could lead to reactor trip and a return to criticality at HZP.

For the cases where the reactor is in hot standby or shut down (Modes 3, 4, & 5), the analysis assumes the transient starts with all RCCAs fully inserted (except the highest-reactivity-worth RCCA fully withdrawn) and the initial RCS boron concentration is provided to ensure at least 15 minutes from the high source range neutron flux alarm to the criticality. The core operating limits report (COLR) provides the differential boron concentration (the provided initial concentration minus the critical concentration) as the Technical Specification required shutdown margin which is a function of a critical boron concentration. The reactivity insertion caused by the boron dilution could cause a reduction in shutdown margin, leading to criticality.

- Table 15.4.6-1 shows the initial boron concentration, critical boron concentration, dilution volume, and dilution flow for each of the evaluated cases.

- All fuel assemblies are in the core for all analysis cases.
- The flow rate to the reactor coolant system is assumed to be 265 gpm. This flow rate is the conservative setpoint value for the flow rate high deviation alarm and the automatic closing of the isolation valve. The boron concentration of this added water is assumed to be zero ppm.
- The dilution volume is the same for all cases. The value of the dilution volume is 12,370 ft^3. This volume is the effective RCS volume minus pressurizer volume and accounts for 10% steam generator tube plugging.
- All four RCPs are assumed to be running for the Modes 1 and 2 cases shown in Table 15.4.6-1. For the Modes 3, 4, and 5 cases shown in Table 15.4.6-1 at least one RCP is assumed to be running. Discussion of a boron dilution in Modes 4 & 5 with no RCPs running is provided in Section 15.4.6.3.3.1.
- The analytical limit for the high source range neutron flux alarm (shutdown cases) is assumed to be 0.8 times background.

15.4.6.3.3 Results

15.4.6.3.3.1 Inadvertent Dilution during Modes 4 and 5 (no RCPs running) and Mode 6

For Modes 4 and 5 with no RCPs running (including mid-loop operation), inadvertent boron dilution is also prevented by administrative control, except under the procedures controlling planned dilution and makeup.

For Mode 6, inadvertent dilution is prevented by the administrative controls in place to isolate the primary makeup water supply line in the CVCS (valves locked closed).

For these cases, a boron dilution event is not postulated to occur and is therefore not analyzed.

15.4.6.3.3.2 Inadvertent Dilution during Power, Startup, Hot Standby, and Shutdown

For events initiating at power with automatic rod control, the available time for the operator to diagnose the event and take corrective action following the rod insertion limit alarm is 73.0 minutes prior to the return to criticality. Since the period of time available for operator actions is in excess of the minimum required time of 15 minutes it is acceptable.

For the at-power in manual rod control case or the start-up case, the time from the initiation of the transient to the return to criticality is 61.2 minutes. The time from the beginning of the event to the reactor trip is a small fraction of the time from the initiation of the transient to the return to criticality. This assures that the period of time available for operator actions (defined as the duration between the reactor trip and criticality) is in excess of the minimum required time of 15 minutes.

For boron dilution events occurring from power conditions, core parameters such as reactor power, coolant average temperature, and minimum DNBR ratio are bounded by the response of the uncontrolled RCCA bank withdrawal at power event described in Section 15.4.2.

For shutdown and hot standby cases, the available time for the operator to diagnose the inadvertent decrease in boron concentration and take corrective action following a high source range neutron flux at shutdown alarm is at least 16.0 minutes prior to criticality. Since the period of time available for operator actions is in excess of the minimum required time of 15 minutes it is acceptable.

15.4.6.4 Barrier Performance

For cases where reactor power does not increase during the transient, the barrier performance during a boron dilution is bounded by the results of the inadvertent CVCS operation event documented in Section 15.5.2.

For cases where the transient is initiated at power and reactor power increases, the barrier performance during the transient is bounded by the results for the uncontrolled control rod assembly bank withdrawal at power event documented in Section 15.4.2.

15.4.6.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the rod ejection accidents evaluated in Section 15.4.8.5.

15.4.6.6 Conclusions

For transients initiated at power, the minimum DNBR remains above the 95/95 limit so that fuel failure is not predicted.

The reactor coolant system pressure remains well below 110% of its system design pressure for all cases, so the integrity of the reactor coolant pressure boundary is maintained.

For all cases, when the boron dilution is in progress when the reactor is shut down (or tripped), sufficient indications are available to alert the operator to the uncontrolled reactivity addition and sufficient time is available for the operators to diagnose the situation and take corrective action before criticality or post-trip return to criticality occurs.

This event has does not lead to a more serious fault condition.

Table 15.4.6-1
Summary of Analysis Input Parameters and Results Boron Dilution Analysis

Mode	Title	Initial Boron Concentration, C1 [ppm]	Critical Boron Concentration, C2 [ppm]	Dilution Flow [gpm]	Dilution Volume [ft^3]	Alarm	Available Time [min]
1 (Automatic)	Power Operation	2150	1600	265	12,370	Rod Insertion Limit	73.0
1 (Manual) **2**	Power Operation Start-up	2050	1600	265	12,370	High power range neutron flux (high/low setpoint) Over temperature ΔT	61.2
3	Hot Standby	2250	1600	265	12,370	High source range neutron flux	16.1
4	Hot Shutdown	2600	1850	265	12,370	High source range neutron flux	19.3
5	Cold Shutdown	2550	1950	265	12,370	High source range neutron flux	16.0
6	Refueling	See Text					

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Frequency Classification

This section addresses an event in which a fuel assembly or incore control component (ICCC; i.e., burnable absorbers, source assemblies, etc.) is loaded into an incorrect position in the core. This event is highly unlikely to occur because fuel assembly loading is strictly managed by administrative procedures as described in Chapter 14. The fuel assemblies and ICCCs are marked with identification numbers, and are loaded into the core in accordance with a core loading diagram, which is checked by multiple designated operators. Neutron detectors provide continuous indication of the neutron flux as each fuel assembly is placed into the core. After the fuel loading is completed, the identification numbers are checked again to assure that all the fuel assemblies are loaded into the correct position. An in-core power distribution measurement is then performed during Low Power Testing and/or Power Ascension Testing. Therefore, if an incorrect fuel assembly loading were to occur, it would likely be detected before the core reaches high power, thereby avoiding the possibility of exceeding fuel failure limits.

This event is classified as a postulated accident (PA), as defined in Section 15.0.0.1.

15.4.7.2 Sequence of Events and Systems Operation

Loading errors described in Section 15.4.7.1 can lead to high local heat flux conditions which can result in higher power peaking larger than predicted in the analyses. High power peaking could, in turn, result in a reduced DNBR.

A loading error that leads to a larger increase in power peaking can be detected by the core instrumentation that provides core mapping and temperature measurement prior to full power operation. During fuel loading, neutron detectors provide continuous indication of the neutron flux. After fuel loading, but before the core is returned to high power, a procedurally required in-core power distribution measurement is performed. Loading errors that result in larger-than-expected power peaking can be discovered and corrected at this point.

A loading error that causes a relatively small increase in power peaking has no safety impact. Core analyses include an 8% allowance for uncertainties in local power peaking. A loading error that results in power peaking being within this allowance is covered by the existing analyses.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

Mechanical constraints are in place to prevent a situation where a fuel assembly is in the correct location but has an incorrect azimuthal alignment (i.e., rotated 90° or 180° from the correct position). According to Section 4.2.2.2.2, an indexing hole located in one corner of the top nozzle top plate of each fuel assembly is used to prevent this incorrect rotation. The fuel assembly identification number is engraved on the opposite corner clamp, and is used to visually confirm appropriate assembly loading.

15.4.7.3 Core and System Performance

15.4.7.3.1 Evaluation Model

No transient occurs for this event, thus the typical transient analysis codes are not used. The ANC code is used to calculate both a normal expected radial power distribution and the radial power distributions resulting from the assumed fuel loading errors. ANC is an NRC approved, three-dimensional, two-group diffusion core calculation code based on the nodal expansion method, as described in Section 4.3.3.1.

15.4.7.3.2 Input Parameters and Initial Conditions

Analyses were performed using ANC where radial steady state power distributions were calculated at thirty percent of rated thermal power for four possible types of fuel loading errors. Thirty percent rated thermal power was assumed as a bounding power level for the Low Power Testing during which the power distribution measurements will be obtained. These results are compared to the result of an ANC analysis where a radial power distribution was calculated at thirty percent power for the normal fuel loading pattern shown in Figure 4.3-2 of Section 4.3.

Figures 15.4.7-1 through 15.4.7-4 are core cross-section maps that show the percent deviation in assembly power at each in-core detector location between a correctly loaded core and the incorrectly loaded core for each of the following four cases:

- Case A is an interchange of two fuel assemblies that have large reactivity differences (e.g., an interchange of a Region 1 assembly and a Region 3 assembly).
- Case B is an interchange of two fuel assemblies that have small reactivity differences (e.g., an interchange of a Region 1 assembly and a Region 2 assembly).
- Case C is an interchange between fuel assemblies with and without discrete burnable absorber, where burnable absorber location in the core is correct.
- Case D is where discrete burnable absorber is loaded in an improper position, with all fuel assemblies loaded correctly.

15.4.7.3.3 Results

As shown in Figures 15.4.7-1 through 15.4.7-4, the differences between measured and predicted power distribution are abnormally large and the core symmetries are broken because of the strong power distortion. The power distribution measurement is performed during Low Power Testing and/or Power Ascension Testing; therefore, the vast majority of fuel loading errors can be detected before the core reaches a high power level. Figure 15.4.7-2 shows relatively small power distribution distortion compared to other cases due to the relatively small reactivity changes. Even if it is assumed that the fuel loading error is not identified for scenarios like Case B, the effect on the power distribution is sufficiently small that the fuel integrity is maintained and no fuel damage is predicted.

15.4.7.4 Barrier Performance

This event involves only changes in the distribution of power and heat flux within the core. Overall core power, RCS flow, and RCS pressure are not changed. Therefore, the maximum reactor coolant pressure remains well below 110% of the design pressure. The integrity of the reactor coolant pressure boundary is maintained.

15.4.7.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the rod ejection accidents evaluated in Section 15.4.8.5.

15.4.7.6 Conclusions

Procedures will be followed to ensure that a fuel loading error that could potentially cause fuel damage would be detected before the reactor reaches high power, thus no fuel damage is postulated to occur.

Additionally, if a core loading error is undetected by the power distribution measurement, it has been determined that fuel rod design limits are not exceeded.

The reactor coolant pressure stays below 110% of the design pressure so that the integrity of the reactor coolant pressure boundary is maintained.

This event does not lead to a more serious fault condition.

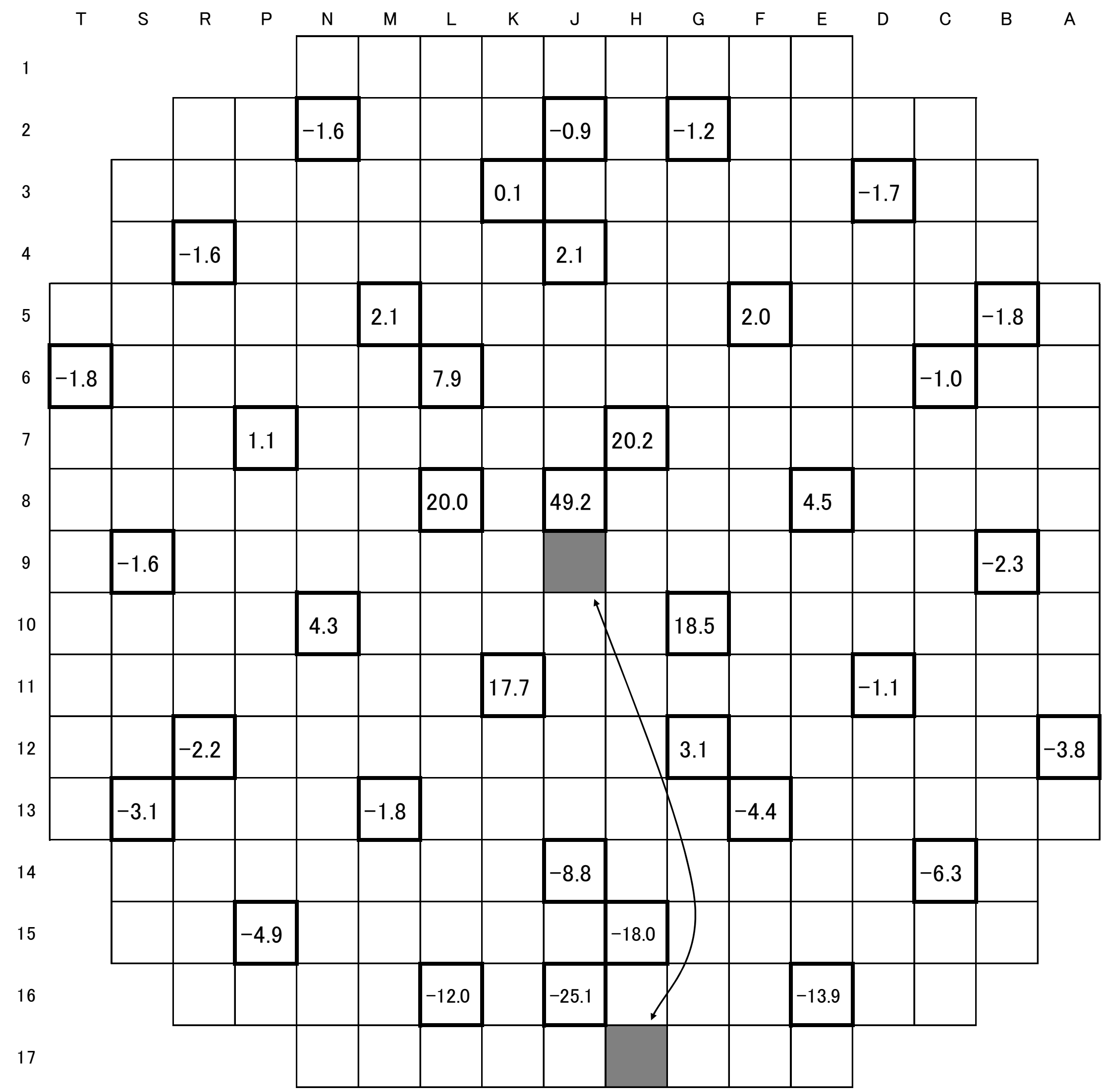


Figure 15.4.7-1 Percent Deviation in Assembly Power at each In-core Detector Location between the Correctly Loaded Core and the Incorrectly Loaded Core
Case A: Assembly Interchange with a Large Reactivity Difference

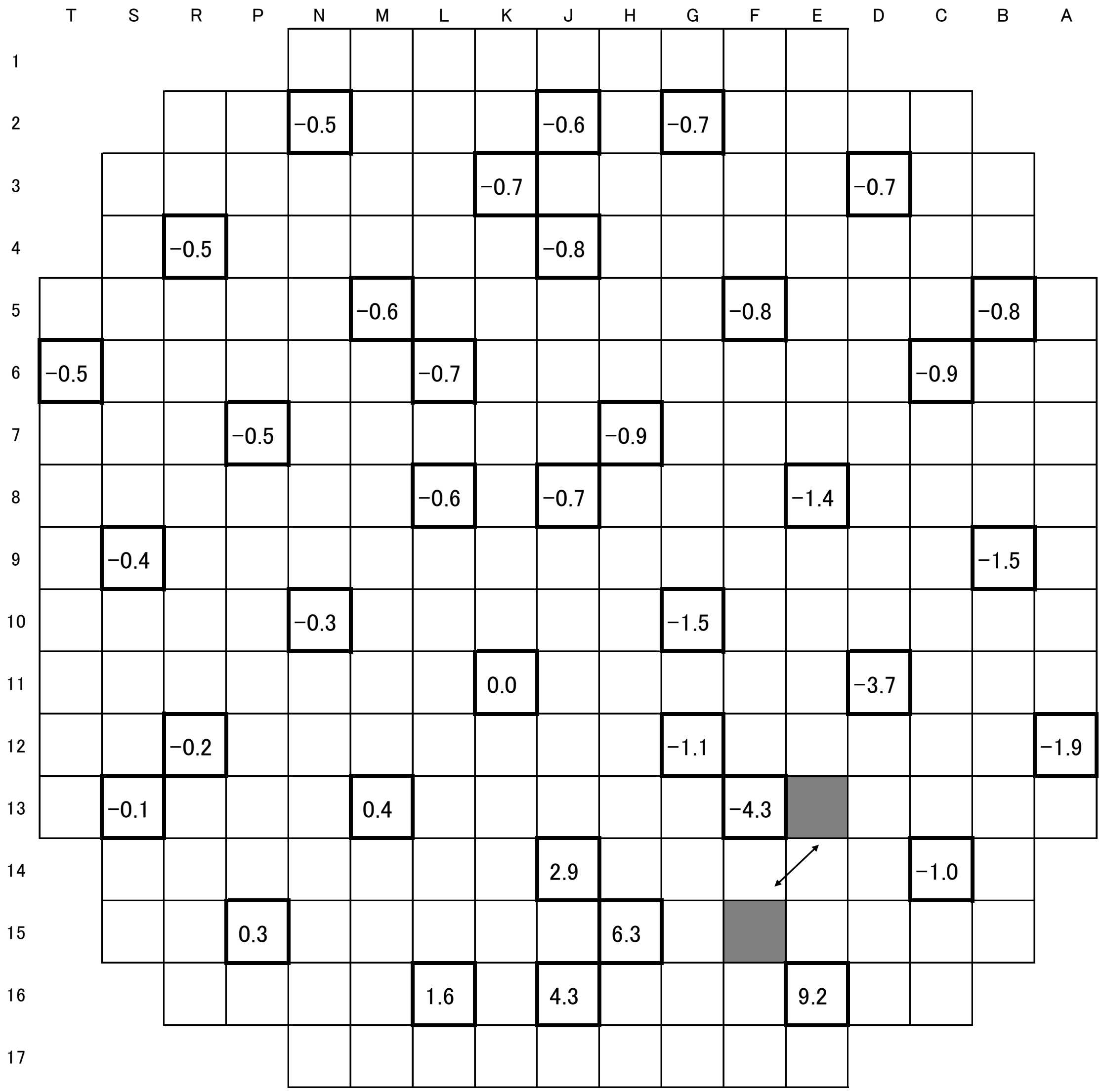


Figure 15.4.7-2 Percent Deviation in Assembly Power at each In-core Detector Location between the Correctly Loaded Core and the Incorrectly Loaded Core
Case B: Assembly Interchange with a Small Reactivity Difference

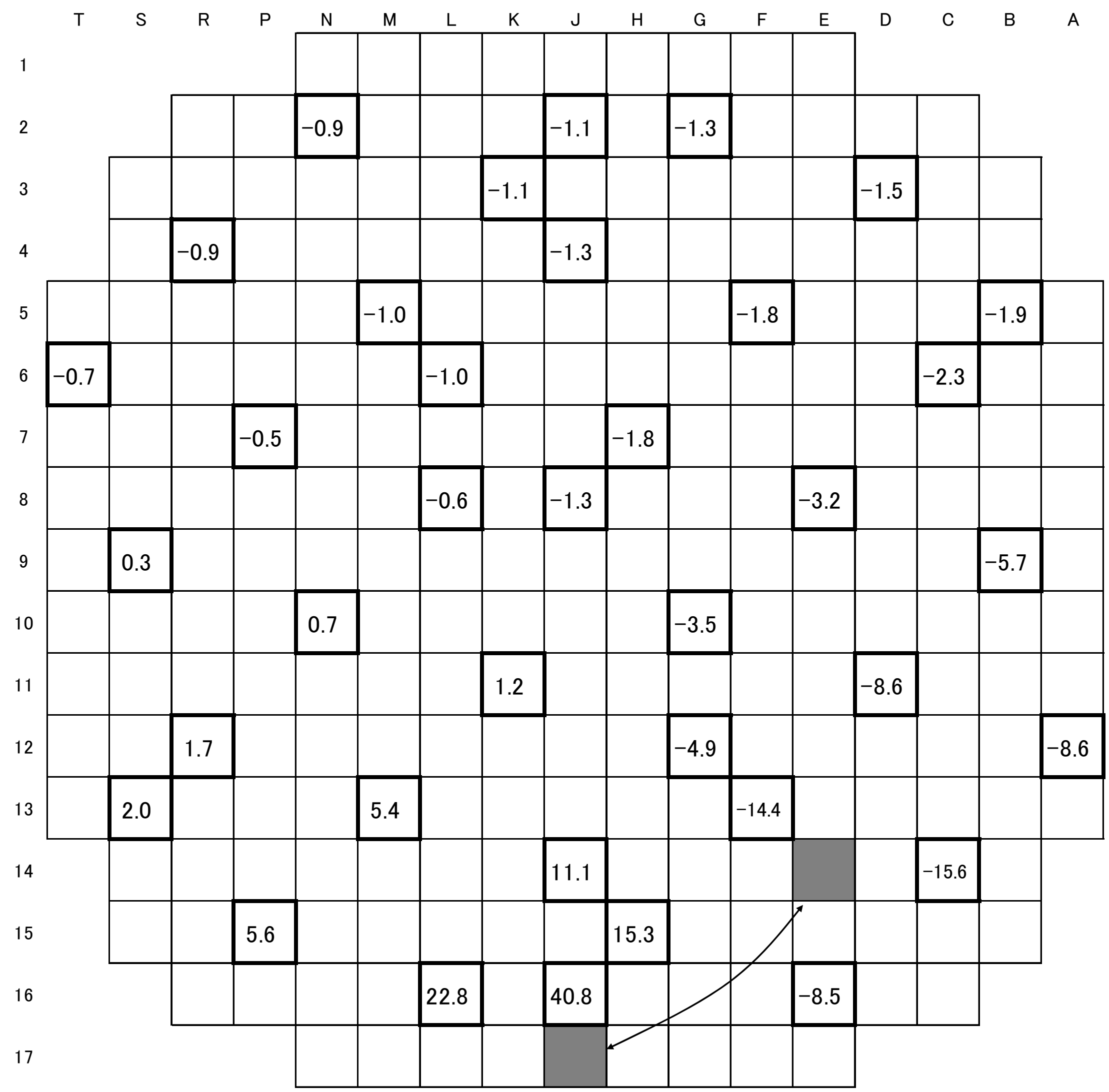


Figure 15.4.7-3 **Percent Deviation in Assembly Power at each In-core Detector Location between the Correctly Loaded Core and the Incorrectly Loaded Core**
Case C: Assembly Interchange with and without Burnable Absorber

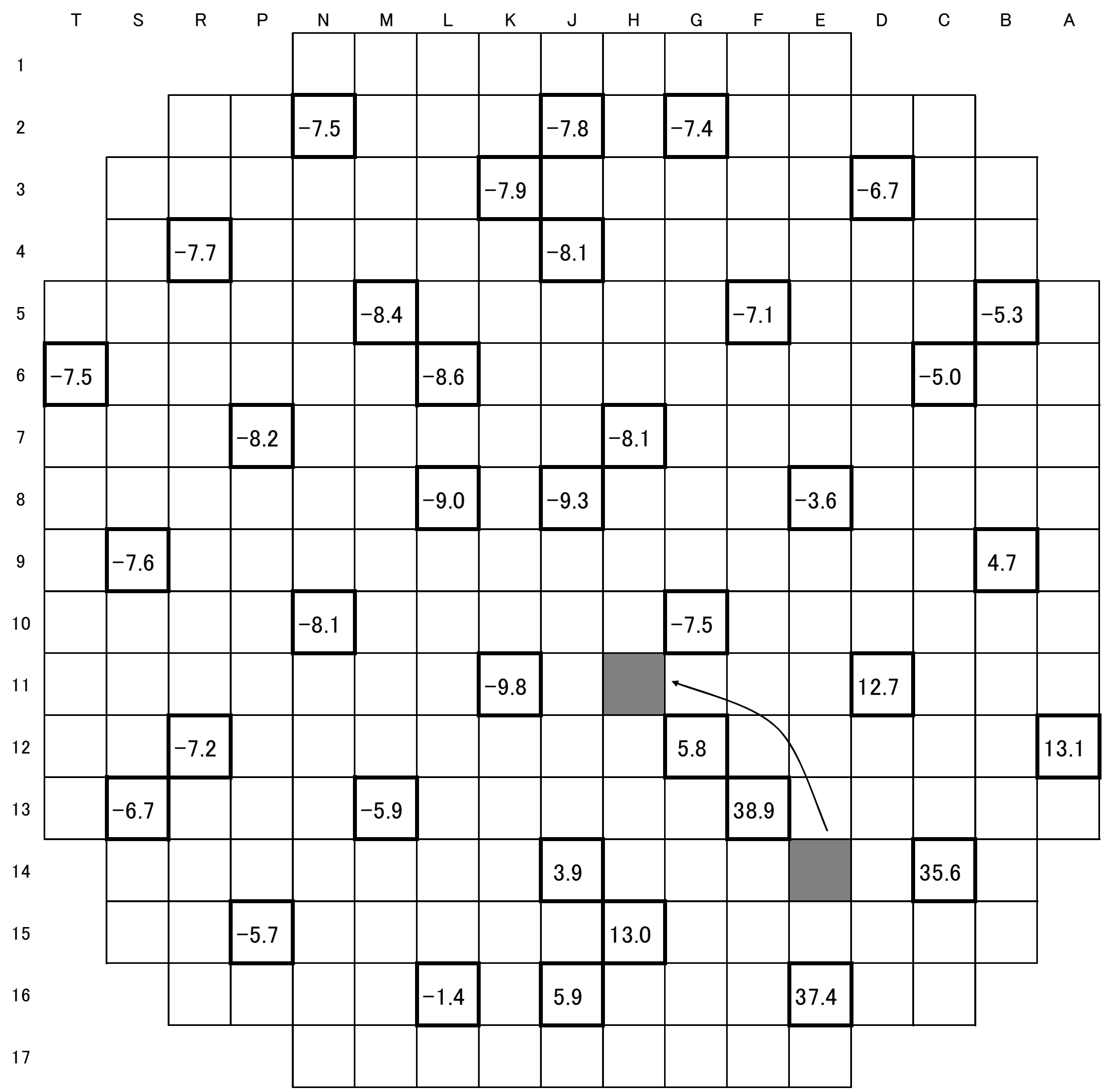


Figure 15.4.7-4 **Percent Deviation in Assembly Power at each In-core Detector Location between the Correctly Loaded Core and the Incorrectly Loaded Core**
Case D: Burnable Absorber Loaded in Incorrect Location

15.4.8 Spectrum of Rod Ejection Accidents

15.4.8.1 Identification of Causes and Frequency Classification

This accident is defined as the mechanical failure of a control rod drive mechanism (CRDM) housing, which results in the ejection of an rod cluster control assembly (RCCA) and its drive shaft. The consequence of this RCCA ejection is a rapid positive reactivity insertion with an increase of core power peaking, possibly leading to localized fuel rod failure. The nuclear excursion is terminated by Doppler reactivity feedback from increased fuel temperature, and the core is shut down by the high power range neutron flux trip (high and low setpoint for hot full power (HFP) and hot zero power (HZP), respectively).

This event is classified as a postulated accident (PA) as defined in Section 15.0.0.1. Historically, these events have been classified as Condition IV events as defined in ANSI N18.2 (Ref. 15.4-1). Additional event-specific acceptance criteria are described in Section 15.4.8.2.5.

15.4.8.2 Sequence of Events and Systems Operation

This postulated accident is initiated by the failure of the CRDM housing. Sudden ejection of an RCCA adds positive reactivity to a localized region of the core in a very short period of time. This RCCA ejection results in a power excursion in the region near the affected fuel assembly. With the reactivity feedback, the core power eventually reaches an equilibrium state, which is characterized by highly asymmetric power distribution in the radial dimension. This adverse power distribution subsequently leads to overheating of the affected fuel assemblies and possible fuel damage.

The sequence and timing of major events for the spectrum of rod ejection accidents is described in the results section.

The following automatic trip signals are assumed to be available to provide protection from this transient:

- High power range neutron flux (high setpoint)
- High power range neutron flux (low setpoint)
- High power range neutron flux rate

In the safety analysis, the high power range neutron flux rate trip is conservatively ignored.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause a reactor coolant pump (RCP) coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of peak radial average fuel enthalpy, peak fuel temperature and peak reactor coolant pressure so that these maximum values for the entire transient are the same whether offsite power is available or unavailable. Since the two cases have equally limiting peak radial average fuel enthalpy, peak fuel temperature and peak reactor coolant pressure, the case where offsite power is unavailable is not presented.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

15.4.8.2.1 Nuclear Design

The US-APWR reactivity control functions are provided by two independent mechanisms: adjusting the boron concentration in the reactor coolant system (RCS) (chemical shim) and maneuvering the RCCAs.

Chemical shim is used to compensate slow reactivity changes such us fuel depletion. It also provides sufficient negative reactivity to bring the reactor to cold shutdown.

The RCCAs are typically used for rapid reactivity changes, such as changes in power demand or temperature transients. During normal operation, the RCCAs can be inserted up to their insertion limits, as specified in the Technical Specifications. Therefore, the control banks are assumed to be at their respective insertion limits prior to the rod ejection accident. The most limiting ejected rod location is identified for each core condition.

15.4.8.2.2 Mechanical Design

Since rod ejection is potentially a PA, mechanical design and certain quality control programs are implemented to prevent its occurrence

- The structural reliability of the CDRM housing for the US-APWR is increased by the elimination of the canopy seals.

- All CRDM pressure housings are performed hydrostatic test in accordance with ASME code Section Ⅲ.

- All CRDM pressure housings are individually hydrotested after they are attached to the reactor vessel head.

- The latch mechanism housing and the rod travel housing are single piece forged stainless steel. This material has demonstrated excellent notch toughness at temperatures anticipated to be encountered during the reactor operating life time.

- Anticipated system transients have little effect on the stress levels in CRDM housings. Moments induced by the design basis earthquake are within the allowable range specified by the ASME code, Section III.

15.4.8.2.3 Reactor Protection

The automatic features of the RTS in an RCCA ejection incident include the high power range neutron flux trip (high and low setpoints) and the high power range neutron flux rate. The reactor trip functions are described in Section 7.2.

Under the conditions created by the rod ejection accident, the reactor is shut down by the high power range neutron flux trip. The high power range neutron flux rate trip is conservatively ignored in the safety analyses.

15.4.8.2.4 Effects on Neighboring Control Rod Housings

It is assumed that the break of the CRDM housing occurs at a weld. . The broken CRDM housing is ejected vertically upward because it is guided by the drive rod, and the driving force from the reactor coolant is vertical. However, the travel of the ejected CRDM housing is limited by the missile shield, which dissipates its kinetic energy. The broken part of the CRDM housing rebounds after impact with the missile shield. However, the broken CRDM contains the drive rod inside, and the top end plates of the rod position indicator coil assemblies prevent it from hitting a second CRDM housing. Even if the rebounding CRDM directly hits an adjacent CRDM housing , its kinetic energy would be too low to cause the mechanical failure of a second CRDM housing. Therefore, the adjacent control rod housing failure does not further increase the severity of the accident.

15.4.8.2.5 Acceptance Criteria

For the rod ejection accident, the objective is to eliminate or minimize the potential for fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. In an effort to accomplish this goal, this analysis applies the following additional acceptance criteria (beyond those for a typical PA):

- Peak reactor coolant pressure is less than that could cause stresses, which exceed the “Service Limit C” as stipulated by the ASME code (SRP 15.4.8).

- The total number of failed fuel rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing each of the criteria below. The fuel rods that are predicted to fail more than one of the criteria are not double counted (SRP 4.2 Appendix B).

  (a) The high cladding temperature failure criterion for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure, or 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure. For full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g. DNBR).

(b) The pellet/cladding mechanical interaction (PCMI) failure criterion is an increase in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure 15.4.8-1.

In addition to the fuel failure and boundary criteria above, the following criteria from SRP 4.2 Appendix B apply to core coolability:

- Peak radial average fuel enthalpy must remain below 230 cal/g.
- Peak fuel temperature must remain below incipient melting conditions.

15.4.8.3 Core and System Performance

15.4.8.3.1 Evaluation Model

The TWINKLE-M code (Ref. 15.4-2) is used to determine the core transient including core average and local power behavior following a RCCA ejection. An increase of local power and the Doppler feedback due to an increase of fuel effective temperature are calculated in each spatial mesh.

The three-dimensional method is applied to the HZP condition in order to conform to the PCMI fuel failure criteria. The applied core mesh division is 2 x 2 meshes per assembly in the radial direction. For the HFP case, a one-dimensional method is applied and an external reactivity insertion is simulated by changing the eigenvalue of the neutron kinetics. A small Doppler weighting factor is used to compensate for collapsing the 3-D problem into a 1-D axial model. The suitability and conservatism of this approach is confirmed in Appendix C of Reference 15.4-2. Changes made by MHI to increase the number of meshes and the use of TWINKLE-M for transient calculations are further described in Reference 15.4-2.

The VIPRE-01M code (Ref. 15.4-3) calculates fuel temperature, fuel enthalpy, and DNBR at the hot spot during the transient using two interface files created by the TWINKLE-M code. One of the interface files is a time-dependant history of the core average power and the other is a time-dependant history of the hot channel factor. The hot channel factor time history is used for the three-dimensional calculation only. The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation. Additional details regarding the VIPRE-01M methodology are available in Reference 15.4-2.

Additional details on the overall evaluation methodology for the rod ejection accident analysis can be found in the MHI Non-LOCA Methodology topical report (Ref. 15.4-2). Analyses of the spectrum of rod ejection accidents are performed for the following cases:

- Hot full power initial condition at beginning-of-cycle (HFP BOC)
- Hot full power initial conditions at end-of-cycle (HFP EOC)
- Hot zero power initial condition at beginning-of-cycle (HZP BOC)
- Hot zero power initial condition at end-of-cycle (HZP EOC)

15.4.8.3.2 Input Parameters and Initial Conditions

Plant initial conditions are given in Section 15.0.0.2. The following assumptions are utilized in order to calculate conservative transient results for the four previously described rod ejection accident cases. Analysis assumptions and calculation conditions for the core kinetics follow Regulatory Guide 1.77 Appendix A (Ref. 15.4-5). Table 15.4.8-2 tabulates the parameters used in the rod ejection analysis.

- Initial condition assumptions are based on a 24 month equilibrium core at the beginning-of-cycle (BOC) and end-of-cycle (EOC)
    - HFP with initial uncertainty for fuel temperature evaluation (102% of the licensed core thermal power level with initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi below the nominal value), and without initial uncertainty for rods-in-DNB evaluation (consistent with the use of the RTDP). The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.
    - HZP for fuel enthalpy evaluation (the initial values of reactor coolant average temperature and RCS pressure used in VIPRE-01M are assumed to be 4°F above and 30 psi below the values corresponding to hot standby conditions).
- A conservative large reactivity, chosen at the design limit, is inserted within 0.1 seconds.
    - In the three-dimensional methodology case, the most reactive RCCA ejection is selected. The inserted reactivity is directly simulated by the change of the absorption cross section caused by the ejection of the most reactive RCCA. The deficit of the inserted reactivity compared with the design limit is made up for by changing the eigenvalue of the neutron kinetics.
    - In the one-dimensional methodology case, the reactivity design limit is externally added to the core within 0.1 seconds.
- The Doppler feedback is conservatively estimated by multiplying the fast absorption cross section for the given change in the calculated fuel effective temperature by a conservative multiplier. In the MHI one-dimensional methodology, a small Doppler weighting factor is used to compensate for collapsing the 3-D problem into a 1-D axial model. The suitability and conservatism of this approach

is confirmed by a comparison between the three-dimensional and one-dimensional kinetic results presented in Appendix C of Reference 15.4-2. Additional details regarding the Doppler feedback are discussed in Table 15.0-1.

- Moderator reactivity feedback has a relatively minor contribution during the initial phase of the transient. The reason is that the heat transfer between the fuel and moderator takes much longer than the neutron response time. However, after the initial neutron flux peak occurs, the moderator reactivity feedback slows the decrease of neutron power. The moderator reactivity is conservatively estimated by multiplying the moderator slowing down cross section by a conservative multiplier. Additional details regarding moderator reactivity feedback are discussed in Table 15.0-1.

- For the hot spot fuel calculation using the VIPRE-01M code, the film heat transfer coefficient is calculated using the Dittus-Boelter correlation for single phase heat transfer, the Thom correlation for nucleate boiling heat transfer, and the Bishop-Sandberg-Tong correlation for film boiling heat transfer after DNB. Hot spot DNB is conservatively assumed to start at the beginning of the accident. Additional details regarding film heat transfer are available in Reference 15.4-3.

- Conservative assumptions for the trip simulation (trip reactivity, rod drop time, RTS signal processing delays) are used in the analysis. The reactor trip is simulated by dropping partially and fully withdrawn rod banks into the core. Maximum time delay from reactor trip signal to rod motion and a conservative RCCA insertion curve are simulated as described in Table 15.0-4 and Section 15.0.0.2.5, respectively. The trip reactivity used is the design limit, which is -4%Δk/k for the hot full power case and -2%Δk/k for the HZP case, respectively.

- The reactor is assumed to be automatically tripped on the high power range neutron flux signal. The reactor trips on the high setpoint for the full power cases and the low setpoint in the zero power cases. Table 15.0-4 summarizes the reactor trip analytical limits assumed in the analysis.

- Minimum delayed neutron fraction and minimum neutron lifetime are used.

- In the case of three-dimensional methodology, a history of hot channel factor is calculated by the TWINKLE-M code. For conservatism, the maximum value of the hot channel factor used in the VIPRE-01M code is adjusted to the design limit.

- In the case of one-dimensional methodology, the hot channel factor used in the VIPRE-01M code is assumed to instantaneously increase to the design limit and is conservatively assumed to remain constant, ignoring feedback effects during the transient.

- Initial conditions of hot spot fuel temperature are consistent with the results of the fuel design code FINE (Ref. 15.4-6). According to the evaluation purpose, the following assumptions are applied conservatively to pellet and cladding gap conductance in the transient analysis using the VIPRE-01M code.

- Remains constant for fuel temperature and enthalpy analysis
- Instantaneously decreases to zero for the adiabatic fuel enthalpy analysis
- Rapidly increases to the maximum value for the cladding temperature analysis
- Realistically increases for the DNB rods and RCS pressure analysis

15.4.8.3.3 Results

Analyses are performed for RCCA ejection at the BOC and EOC with HFP and HZP. For all cases, the RCCAs are inserted to their insertion limits before the rod ejection occurs. The reactor power, fuel and cladding temperature, and radial average fuel enthalpy transients for the HFP BOC case are presented in Figures 15.4.8-2 through 15.4.8-4. The same transient parameter information for the HFP EOC case is in Figures 15.4.8-5 and 15.4.8-7. The reactor power and fuel enthalpy transients for HZP cases are presented in Figures 15.4.8-8 through 15.4.8-10 for the BOC case and Figures 15.4.8-11 through 15.4.8-13 for the EOC case, respectively. The calculated sequence of events corresponding to these limiting events is provided in Table 15.4.8-1. These analytical results are discussed in the following paragraphs:

- Beginning-of-cycle, full power

  For the HFP BOC case, control bank-D is assumed to be inserted to its insertion limit when the rod ejection occurs. A bounding maximum ejected rod worth of 110 pcm and a design hot channel factor of 5.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power, and the power increase is terminated by Doppler feedback. The reactor trip is initiated by high power range neutron flux (high setpoint) and the reactor returns to subcritical following the trip. The peak fuel centerline temperature is 4220°F, which remains below the fuel melting temperature limit.

  The rods in DNB analysis confirmed that the number of rods predicted to be in DNB is less than 10% of the core, which is the value used in the radiological consequence analysis.

- Beginning-of-cycle, zero power

  For the HZP BOC case, control bank-D is assumed to be fully inserted and the others inserted to their insertion limit when the rod ejection occurs. A bounding maximum ejected rod worth of 600 pcm and a hot channel factor of 14.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power, and the power excursion is terminated by Doppler feedback. The reactor trip is initiated by high power range neutron flux (low setpoint) and the reactor returns to subcritical following the trip. The peak fuel enthalpy is 97.5 cal/g (the increase of the peak fuel enthalpy from its initial condition is 49.0 cal/g). The number of PCMI failed fuel rods is zero.

- End-of-cycle, full power

  For the HFP EOC case, control bank-D is assumed to be inserted to its insertion limit when the rod ejection occurs. A bounding maximum ejected rod worth of 120 pcm and a design hot channel factor of 6.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power, and the power increase is terminated by Doppler feedback. The reactor trip is initiated by high power range neutron flux (high setpoint) and the reactor returns to subcritical following the trip. The peak fuel centerline temperature is 4325°F, which remains below the fuel melting temperature limit.

  The rods in DNB analysis confirmed that the number of rods predicted to be in DNB is less than 10% of the core, which is the value used in the radiological consequence analysis.

- End-of-cycle, zero power

  For the HZP EOC case, Control Bank-D is assumed to be fully inserted, and the others inserted to their insertion limits when the rod ejection occurs. A bounding maximum ejected rod worth of 800 pcm and a hot channel factor of 35.0 are assumed to provide margin for future cores. The reactivity insertion causes a rapid increase in power, and the power excursion is terminated by Doppler feedback. The reactor trip is initiated by high power range neutron flux (low setpoint) and the reactor returns subcritical following the trip. The hot spot peak fuel enthalpy is 72.7 cal/g and the prompt fuel enthalpy rise is 50.8 cal/g. The number of PCMI failed fuel rods is zero.

For all four cases analyzed, the average fuel pellet enthalpy at the hot spot remains significantly below 230 cal/g.

In the BOC HZP and EOC HZP cases, the hot spot peak fuel enthalpy is well below the high cladding temperature failure criterion. Therefore, high cladding temperature failure does not occur.

In the BOC HZP and EOC HZP cases, the prompt fuel enthalpy rise is less than 60 cal/g, which is the lowest criterion of the PCMI failure depicted in Figure 15.4.8-1. Additionally, the oxide/wall thickness rate is less than 0.2 (described in Section 4.2.3.3.6). Therefore, the PCMI failure does not occur in either case.

If a water-logged fuel rod is assumed to exist near the hot spot, this fuel rod may fail at a lower enthalpy rise than the intact fuel rods. However, the probability that a water-logged fuel rod exists, and the probability that such a fuel rod is near the hot spot are both extremely low; thus, the probability of fuel failure in a water-logged fuel rod is negligible.

The rod ejection accident creates an opening in the reactor coolant system. Following the RCCA ejection, the plant response is the same as a small-break loss-of-coolant accident (LOCA). The effects and consequences of a small break LOCA are discussed in Section 15.6.5.

15.4.8.4 Barrier Performance

15.4.8.4.1 Evaluation Model

The evaluation for the peak RCS pressure analysis is similar to the model used for hot spot analysis described in Section 15.4.8.3.1. The TWINKLE-M code is used to analyze the core average power histories following a rod ejection accident. The VIPRE-01M code generates a time-dependent core total void fraction and core heat flux interface file which is used by the MARVEL-M code to calculate the RCS pressure transient. Additional details regarding this methodology are provided in Reference 15.4-2.

15.4.8.4.2 Input Parameters and Initial Conditions

The barrier performance case for peak RCS pressure is similar to the hot spot analysis described in Section 15.4.8.3.2 with the following differences:

- The initial power level is 102% of the licensed core thermal power level with initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi above the nominal value. This combination of initial condition uncertainties maximizes RCS pressure calculated by MARVEL-M. The nominal value of core power, reactor coolant temperature , and RCS pressure conditions are described in Table 15.0-3.

- Pressurizer spray is not assumed.

- The void fraction used in the peak RCS pressure analysis is conservatively assumed by multiplying the void fraction result of the thermal-hydraulic calculation by a conservative multiplier.

15.4.8.4.3 Results

The RCP outlet pressure is the highest pressure in the RCS and is presented in place of RCS pressure for the purpose of confirming the reactor coolant pressure boundary limits are not exceeded. Figure 15.4.8-14 indicates that the RCS pressure remains well below 110% of the system design pressure, so the integrity of the reactor coolant pressure boundary is maintained. By meeting this criterion, the peak RCS pressure also remains less than the “Service Limit C” as stipulated by the ASME code.

Primary safety valve flow is not presented because the peak RCS pressure barely exceeds the pressurizer safety valve set pressure. Containment response is bounded by the containment response to the LOCA event, as described in Section 6.2.

The main steam system pressure is not challenged by this event.

15.4.8.5 Radiological Consequences

The radiological consequences evaluation for this transient is based on the alternative source term (AST) guidance documented in Reference 15.4-4.

The evaluation of the radiological consequences of the postulated rod ejection accident assumes that the reactor is operating at the maximum allowable limit for reactor coolant activity and that leaking steam generator tubes results in a buildup of activity in the secondary system.

It is assumed for the rod ejection accident that 10% of the fuel rods are assumed to be damaged. The activity in the fuel-cladding gap for these fuel rods is released to the reactor coolant. In addition, a small fraction of the fuel is assumed to melt. The activity inventory from the melted fuel is added to the reactor coolant.

The analysis assumes a primary-to-secondary leakage rate of 600 gpd at the steam generators. Activity carried over to the secondary system is assumed to be released to the environment through the main steam safety valves or main steam relief valves.

Activity spills from the reactor vessel head into the containment. A portion of this activity is assumed to be available for release to the environment through the containment leakage.

15.4.8.5.1 Evaluation Model

Mathematical models used in the analysis are described in the following sections:

- The offsite and onsite doses are calculated with the RADTRAD code.
- The atmospheric dispersion factors (χ/Q values) used in the analysis are described in Section 15.0.3.3.
- The total effective dose equivalent (TEDE) doses to a receptor at the exclusion area boundary (EAB) and outer boundary of the low-population zone (LPZ) are analyzed using the models described in Section 15.0.3.1 and Appendix 15A.

The potential release paths to the environment are from:

- The initial activity in the secondary side.
- The primary-to-secondary leakage after the rod ejection accident.

- The primary leakage to containment which then leaks via the assumed design basis containment leakage.

- Engineered safety features (ESF) system leakage from manual actuation

Figure 15A-5 depicts the leakage sources to the environment modeled in the dose computation.

Additionally, radionuclide decay of the nuclides is credited prior to release to the environment. After release, no decay is credited.

15.4.8.5.2 Input Parameters and Initial Conditions

The major assumptions and parameters used in the analysis are itemized in Table 15.4.8-3, Tables 15.0-8 through 15.0-10 and Tables 15.0-12 through 15.0-14.

The rod ejection accident releases iodines, alkali metals, and noble gases. In the reactor coolant, equilibrium iodine levels are assumed to exist along with the noble gas and alkali metal concentrations from the allowable design fuel defect. When compared to the releases from the fuel after the ejection event, the pre-existing activities are minor.

The total fission-product gap fraction available for release following any reactivity-initiated accident includes the steady-state gap inventory (present prior to the event) plus any fission gas released during the event.

The steady-state releases from the gap between the cladding and fuel is based on RG 1.183 (Ref. 15.4-4). The fission product gap inventory is increased to 10% of the iodines and noble gases, and 12% of the alkali metals. It is conservatively assumed that the failed fuel rods are operating at levels above the core average, and the releases are increased by the appropriate radial peaking factor.

In addition, transient fission gas release from the fuel rod is considered to be based on SRP 4.2, Appendix B (Ref. 15.4-7). The transient release is assumed to be 11% of the iodines and noble gases.

For this analysis, it is conservatively assumed that the fraction of melted fuel is 0.25%. For the melted fuel, 100% of the noble gases and 50% of the iodines and alkali metals are assumed to be released to the reactor coolant.

It is further assumed that the secondary concentration due to pre-existing primary-to-secondary leakage is 10% of the maximum primary equilibrium concentration for the iodines and alkali metals. The primary-to-secondary leakage of 600 gpd is high compared to the actual leakage rate.

It is assumed that leakage from ESF system leakage occurs when operator starts manual actuation of containment spray systems. With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the refueling water storage pit (RWSP) water. With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

Manual actuations of the containment spray systems and the annulus emergency exhaust systems are assumed for a rod ejection accident (See Section 7.5). The containment spray systems are assumed to start at 35 minutes after the initiation of an accident, and the penetration areas are assumed to become negative pressure at 34 minutes by the annulus emergency exhaust systems. The containment leakage is assumed to be the same as that assumed for LOCA analyses (See Section 15.6.5).

The filtration system considered in the analysis which limits the consequences of the rod ejection accident is the annulus emergency exhaust system and the main control room (MCR) heating, ventilation, and air conditioning (HVAC) system.

The χ/Q values and breathing rates are listed in Table 15.0-13. The breathing rates are obtained from NRC Regulatory Guide 1.183 (Ref. 15.4-4).

Other assumptions relating to the transport, reduction, and release of radioactive material to the environment are those covered in Appendix H of RG 1.183 (Ref. 15.4-4).

15.4.8.5.3 Results

The calculated TEDE doses have been analyzed for the limiting 2-hour dose at the EAB and for the duration of the transient at the LPZ outer boundary. The results are listed in Table 15.4.8-4.

As shown in Table 15.4.8-4, the TEDE doses are calculated to be 5.1 rem at the EAB and 4.5 rem at the LPZ outer boundary. These doses are less than 25% of the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34.

The doses for the MCR for the rod ejection accident are bounded by the doses calculated for the LOCA event described in Section 15.6.5.5. Consequently, no doses are provided for the rod ejection accident.

15.4.8.6 Conclusions

For an rod ejection accident, the analysis for HFP conditions demonstrates that the number of rods predicted to be in DNB is less than the number assumed for the radiological dose analysis. Additionally, the peak fuel centerline temperature remains below the fuel melting temperature limit. The analysis for HZP conditions demonstrates that fuel enthalpy and the fraction of local cladding oxidation due to the zirconium-water reaction rate remain below the limits for PCMI failure.

The peak RCS pressure remains below 110% of the RCS design pressure. Therefore, the integrity of the reactor coolant pressure boundary is maintained. Similarly, the main steam design pressure does not exceed 110% of the main steam system design pressure, so the integrity of the main steam system is also maintained.

The resultant doses are well within the guideline limit of 25 rem identified in 10 CFR 50.34.

In addition, this event does not lead to a more serious fault condition.

Table 15.4.8-1
Time Sequence of Events for Rod Ejection

Accident	Event	Time (seconds)
Case 1: HFP Beginning-of-cycle	Rod ejection occurs	0.0
	High power range neutron flux (high setpoint) analytical limit reached	0.07
	Peak reactor power occurs	0.11
	Reactor trip initiated (rod motion begins)	0.67
	Maximum fuel temperature occurs	2.5
	Maximum fuel enthalpy occurs	2.5
Case 2: HFP End-of-Cycle	Rod ejection occurs	0.0
	High power range neutron flux (high setpoint) analytical limit reached	0.06
	Peak reactor power occurs	0.11
	Reactor trip initiated (rod motion begins)	0.66
	Maximum fuel temperature occurs	2.6
	Maximum fuel enthalpy occurs	2.5
Case 3: HZP Beginning-of-cycle	Rod ejection occurs	0.0
	High power range neutron flux (low setpoint) analytical limit reached	0.24
	Peak reactor power occurs	0.28
	Reactor trip initiated (rod motion begins)	0.84
	Maximum fuel enthalpy occurs	1.8
Case 4: HZP End-of-Cycle	Rod ejection occurs	0.0
	High power range neutron flux (low setpoint) analytical limit reached	0.15
	Peak reactor power occurs	0.16
	Reactor trip initiated (rod motion begins)	0.75
	Maximum fuel enthalpy occurs	1.2

Table 15.4.8-2
Parameters Used In Rod Ejection Analysis

Parameter	HFP		HZP	
	BOC	EOC	BOC	EOC
RCCA Ejection Time	0.1 sec			
Initial Hot Channel Factor	2.60	2.60	(N/A)	(N/A)
Peak Hot Channel Factor	5	6	14	35
Ejected RCCA Worth	110 pcm	120 pcm	600 pcm	800 pcm
Doppler Weighting Factor	1.31	1.28	(N/A)	(N/A)
Minimum scram reactivity	-4 %Δk/k		-2 %Δk/k	
Delayed Neutron Fraction (β_{eff})	0.49 %	0.44 %	0.49 %	0.44 %
Neutron Lifetime	8 μsec	8 μsec	8 μsec	8 μsec

Table 15.4.8-3
Parameters Used in Evaluating the Radiological Consequences of the Rod Ejection Accident (Sheet 1 of 2)

Parameter	Value
Core thermal power level (%)	4540 (2% above the design core thermal power)
Initial reactor (primary) coolant activity (from rods leaking prior to transient)	
Iodine	Initial concentration equal to the 1.0 μCi/g DE I-131 in the reactor coolant. (See Table 15.0-10.)
Alkali metals	Based on 1% fuel defect (See Table 11.1-2.)
Noble gases	300 μCi/g DE Xe-133 (See Table 15.0-12.)
Initial secondary coolant activity (from rods leaking prior to transient)	
Secondary system initial iodine and alkali metal concentration	Based on 10% of reactor coolant concentration
Source Term	
Core Activity	See Table 15.0-14.
Fraction of core inventory released from failed rods	
Fraction of fuel rods assumed to fail during transient (%)	10
Radial power peaking factor (to calculate fraction of total inventory in failed rods)	1.78
Iodine fission product gap fraction Alkali metal fission product gap fraction Noble gas fission product gap fraction	See Table 15.0-8.
Transient release fraction from the fuel rods (%)	11
Fraction of melted fuel (%)	0.25
Fraction of activity released from melted fuel: Iodine and alkali metals (containment release case) (Secondary side release case) Noble gases	 0.25 0.5 1.0
Iodine chemical form released from the SGs Iodine chemical form released from the containment: Iodine chemical form released from the ESF:	elemental: 97%, organic: 3% elemental: 4.85%, organic: 0.15%, particulate: 95% elemental: 97%, organic: 3%

Table 15.4.8-3
Parameters Used in Evaluating the Radiological Consequences of the Rod Ejection Accident (Sheet 2 of 2)

Parameter	Value
RCS and Steam System Parameters	
Total steam generator tube leakage (gal/day)	600
Reactor coolant mass (lb) Secondary coolant mass, 4 steam generators (lb)	646,000 456,000
Primary-to-secondary leakage duration (h)	14
Steam released (lb)	
0 to 8 h	1,540,000
8 to 14 h	1,540,000
Iodine partition coefficient	100
Particulate partition coefficient for moisture carryover in the steam generators	1000
Offsite power	Lost after trip
Containment leakage release data	
Elemental iodine deposition removal rate Decontamination factor for elemental iodine Removal rate for particulates	See Appendix 15A.1.2. See Appendix 15A.1.2. See Appendix 15A.1.2.
Containment volume (ft^3)	2,800,000
Containment leak rate (%/d), 0-24 hr Containment leak rate (%/d), > 24 hr	0.15 0.075
Leakage fraction to the penetration areas (%) Leakage fraction to the environment (%)	50 50
Initiating time of containment spray system (min) Negative pressure arrival time of annulus emergency exhaust system (min)	35 34
HEPA filter efficiency for particulates of annulus emergency exhaust system (%)	99
ESF system leakage release data	
Recirculation water mass (lb) Recirculation water leakage rate (lb/h) Start time of recirculation water leakage (min) Flashing fraction (%)	2,360,000 17.6 30 10
Accident period (d)	30
Radiological dose parameters	
χ/Q	See Table 15.0-13.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.4.8-4
Radiological Consequences of Rod Ejection Accident

Dose Location	TEDE Dose (rem)
EZB (0 to 2 hours)	5.1
LPZ outer boundary	4.5

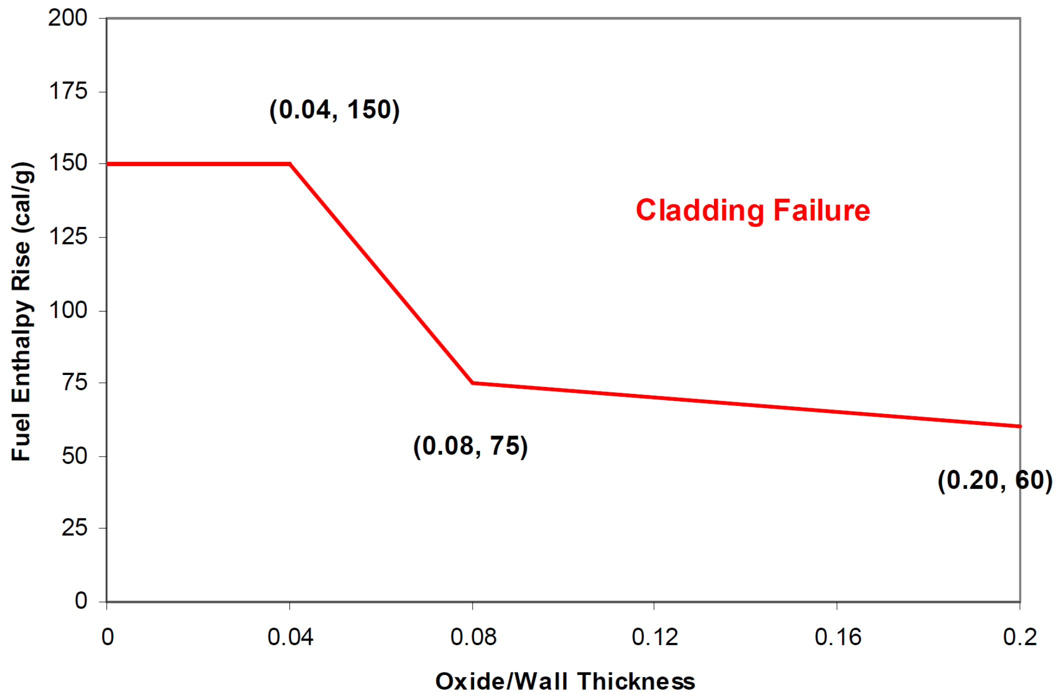


Figure 15.4.8-1 PCMI Fuel Cladding Failure Criteria

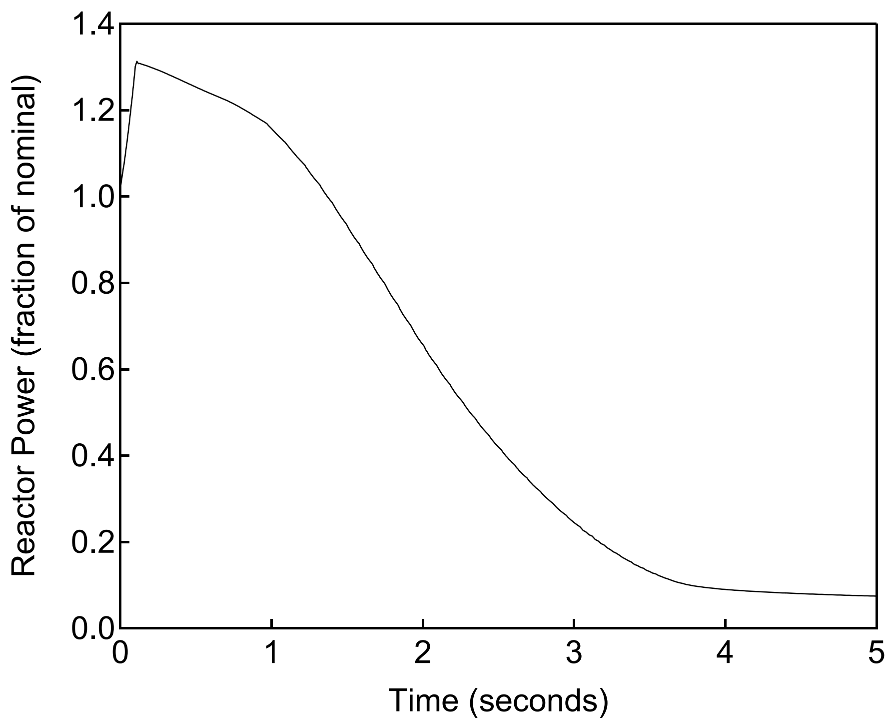


Figure 15.4.8-2 Reactor Power versus Time

Rod Ejection (HFP, BOC)

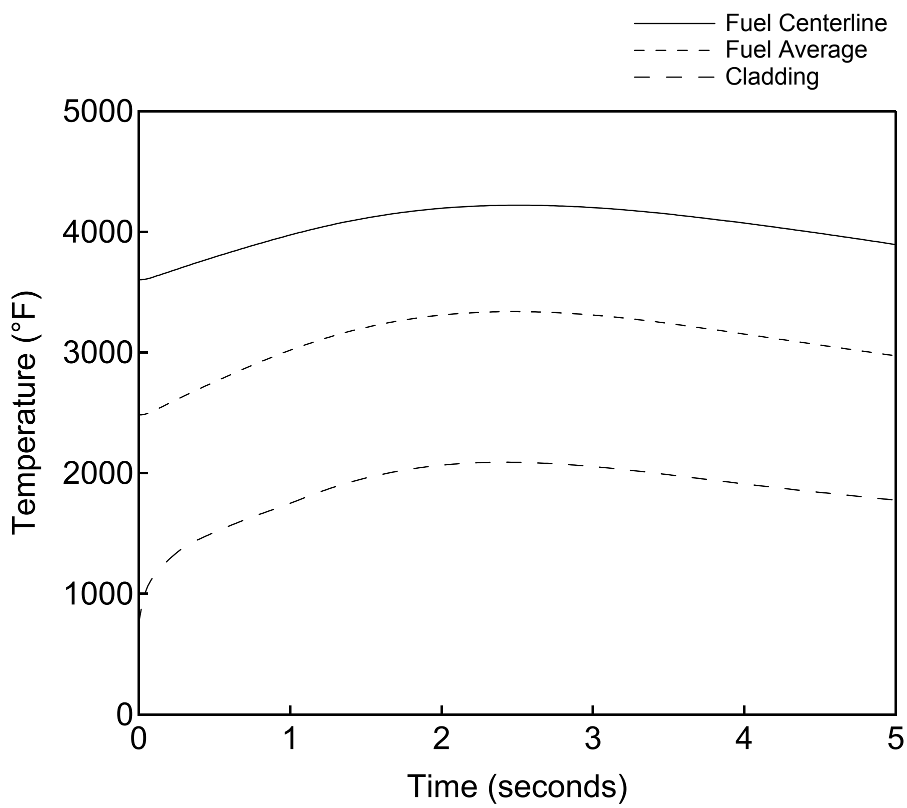


Figure 15.4.8-3 Fuel and Cladding Temperature versus Time

Rod Ejection (HFP, BOC)

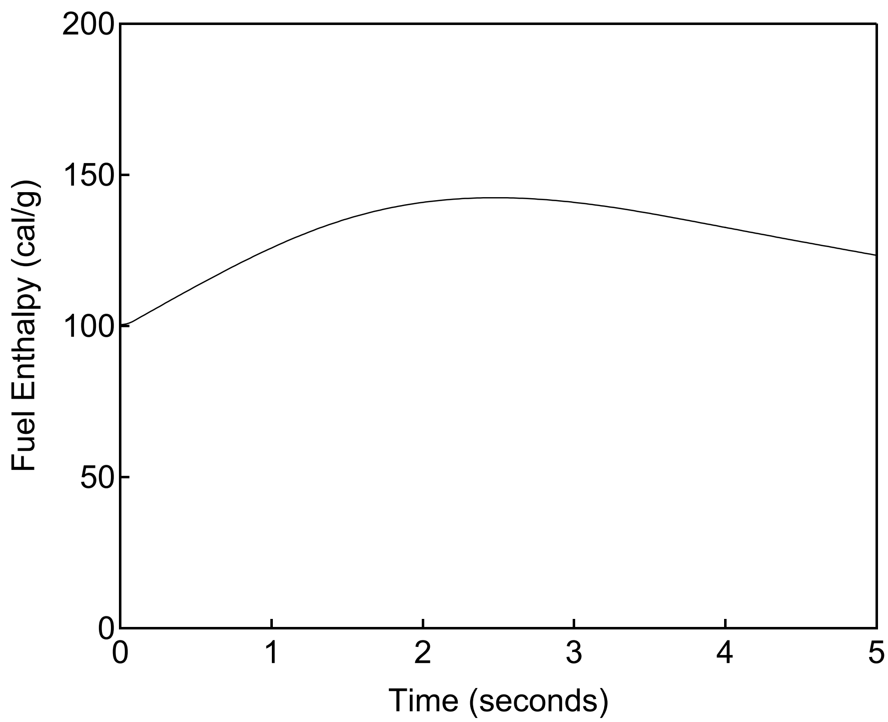


Figure 15.4.8-4 Radial Average Fuel Enthalpy versus Time

Rod Ejection (HFP, BOC)

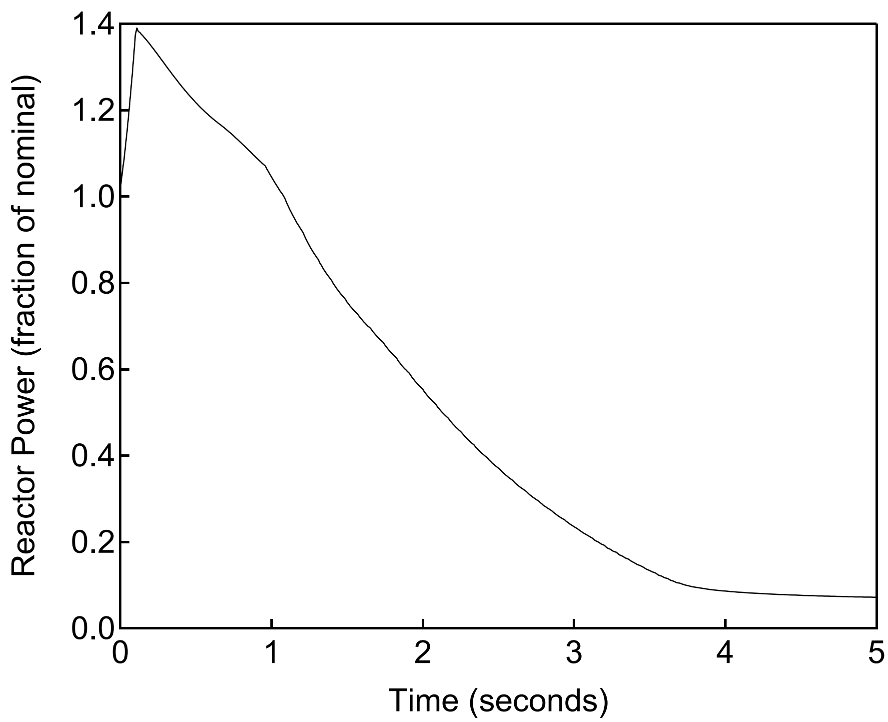


Figure 15.4.8-5 Reactor Power versus Time

Rod Ejection (HFP, EOC)

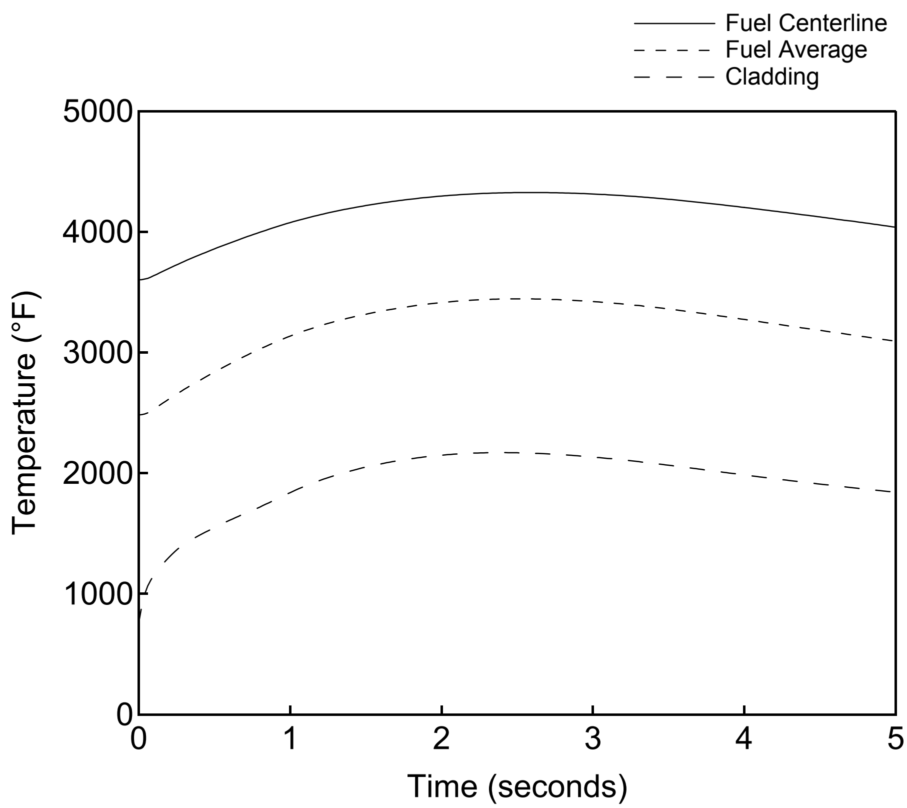


Figure 15.4.8-6 Fuel and Cladding Temperature versus Time

Rod Ejection (HFP, EOC)

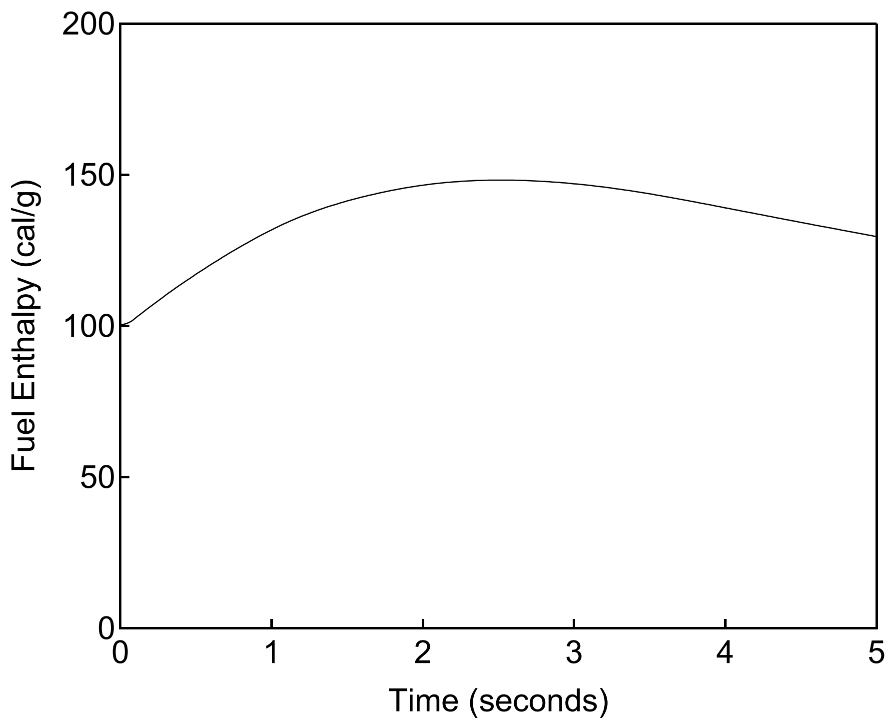


Figure 15.4.8-7 Radial Average Enthalpy versus Time

Rod Ejection (HFP, EOC)

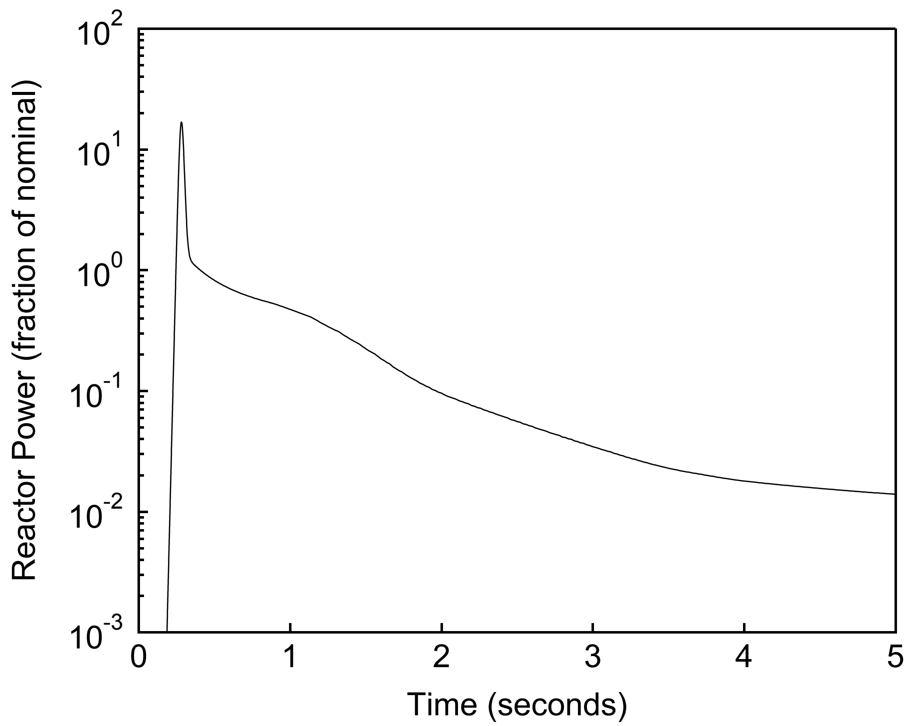


Figure 15.4.8-8 Reactor Power versus Time

Rod Ejection (HZP, BOC)

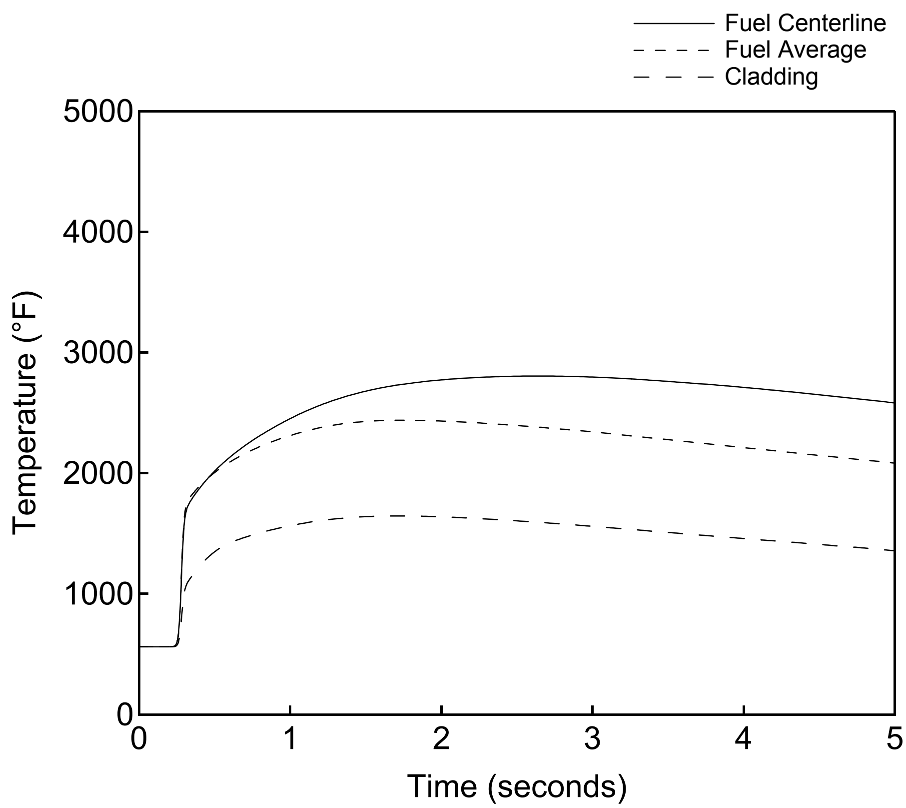


Figure 15.4.8-9 Fuel and Cladding Temperature versus Time Rod Ejection (HZP, BOC)

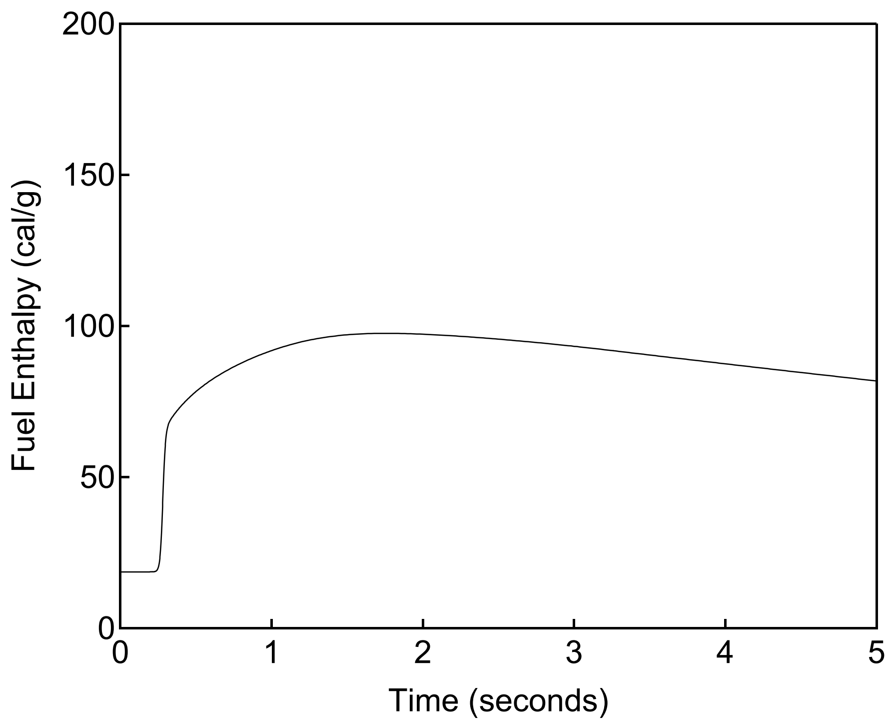


Figure 15.4.8-10 Radial Average Fuel Enthalpy versus Time

Rod Ejection (HZP, BOC)

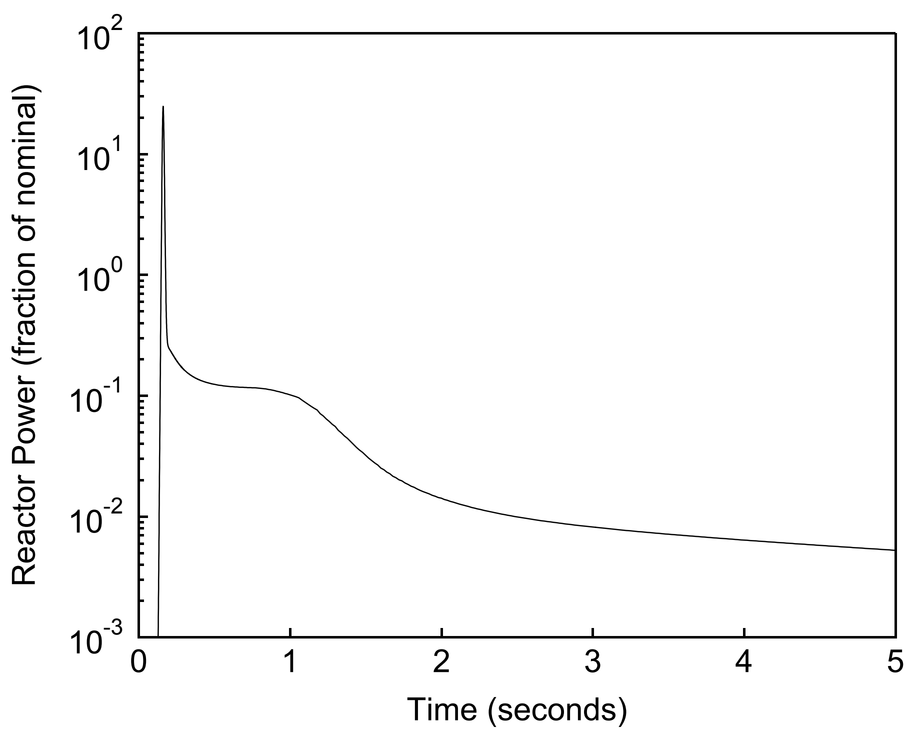


Figure 15.4.8-11 Reactor Power versus Time

Rod Ejection (HZP, EOC)

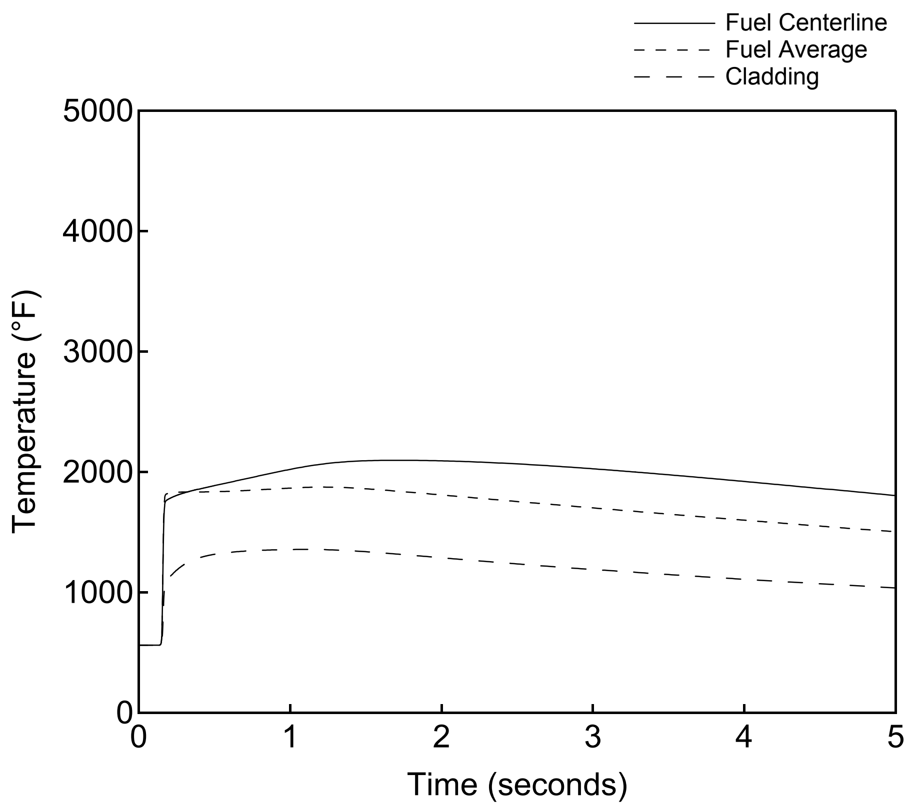


Figure 15.4.8-12 Fuel and Cladding Temperature versus Time

Rod Ejection (HZP, EOC)

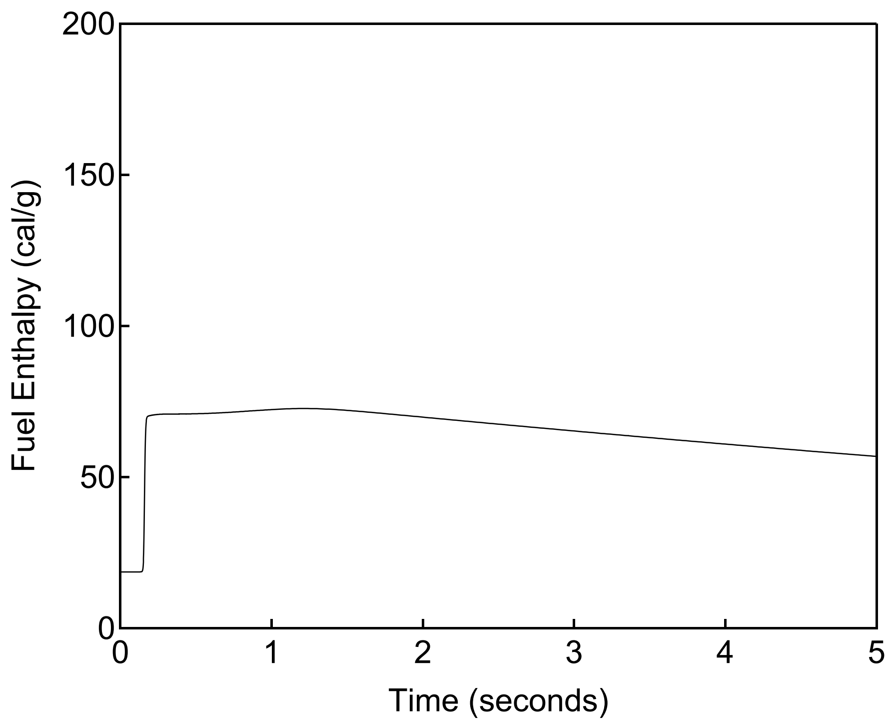


Figure 15.4.8-13 Radial Average Fuel Enthalpy versus Time

Rod Ejection (HZP, EOC)

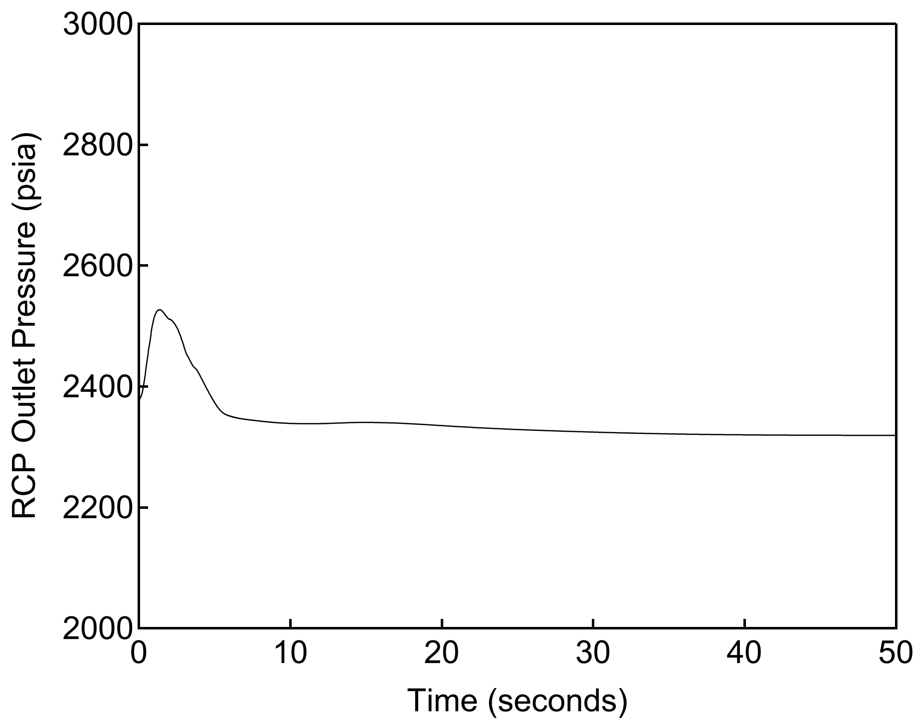


Figure 15.4.8-14 RCP Outlet Pressure versus Time

Rod Ejection (HFP, BOC)
- Peak RCS Pressure Analysis

15.4.9 Spectrum of Rod Drop Accidents in a BWR

Not applicable to US-APWR.

15.4.10 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.4.11 References

15.4-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.4-2 Non-LOCA Methodology, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.4-3 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.4-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.4-5 Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors, NRC Regulatory Guide 1.77, May 1974.

15.4-6 Mitsubishi Fuel Design Criteria and Methodology, MUAP-07008-P (Proprietary) and, MUAP-07008-NP (Non-Proprietary), May 2007.

15.4-7 U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Section 4.2 Revision 3, Appendix B, March 2007.

15.5 Increase in Reactor Coolant Inventory

This section describes analyses that have been performed for events that could result in an increase in the reactor coolant inventory, which, in turn, can lead to reduced reactor coolant system (RCS) pressure, which can decrease DNBR and to the pressurizer filling with liquid.

Analyses of the following events are described in this section:

- Inadvertent Operation of Emergency Core Cooling System (ECCS) that Increases Reactor Coolant Inventory (not applicable to the US-APWR)

- Chemical and Volume Control System (CVCS) Malfunction that Increases Reactor Coolant Inventory

These events are considered anticipated operational occurrences (AOOs) as defined in Section 15.0.0.1.

The results of these analyses determined that the CVCS malfunction that increases RCS inventory event is the more severe of the events listed above. However, even this more severe event has no radiological releases to the environment.

15.5.1 Inadvertent Operation of Emergency Core Cooling System that Increases Reactor Coolant Inventory

This section is not applicable to the US-APWR. It is not applicable because none of the components of the emergency core cooling system (ECCS) (safety injection pumps or accumulators) are capable of injecting water into the reactor coolant system (RCS) at normal, at-power operating pressures. If the ECCS was inadvertently actuated and the safety injection pumps attempted to start, they would not be able to inject water into the RCS because the safety injection pump shut-off head is below the RCS normal operating pressure. Instead, the water from the safety injection pump would be diverted to the refueling water storage pit (RWSP) through a minimum flow path designed to protect the pump against deadheading. The accumulators are available as well, but they also cannot inject water into the RCS when it is at normal operating pressure.

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

This section describes the analysis of the increase in reactor coolant inventory due to the addition of borated water to the RCS by the chemical and volume control system (CVCS). Section 15.4.6 analyzes the reactivity aspects of a boron dilution due to the addition of unborated water to the RCS by the CVCS.

15.5.2.1 Identification of Causes and Frequency Classification

A CVCS malfunction that increases RCS inventory can be caused by an operator error, a test sequence error, or an electrical malfunction. The CVCS normally operates with one charging pump running and a constant letdown flow through the letdown path. The increase of RCS inventory may be caused by an increase in charging flow with letdown operating or by isolation of the letdown path (letdown line and excess letdown line). If the CVCS boron concentration is larger than the RCS boron concentration, the reactor may experience a negative reactivity insertion resulting in a decrease in reactor power and subsequent coolant shrinkage.

This event is classified as an anticipated operational occurrence (AOO). Historically, this event has been classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.5-1). Event frequency conditions are described in Section 15.0.0.1.

15.5.2.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the CVCS malfunction that increases RCS inventory event is described in the results section.

Three cases are considered for this event. The CVCS normally operates with one charging pump running and a constant letdown flow through the letdown path. The increase of RCS inventory may be caused by the full-open failure of the charging flow control valve with one pump running, the spurious startup of a non-operating charging pump, or by the closure of the letdown path (letdown line and excess letdown line). Of these cases, the continuation of a the full-open failure of the charging flow control valve with one pump running has been shown to result in a slightly larger net CVCS flow addition, so only this case is described and analyzed in this section.

The full-open failure of the charging flow control valve causes a net increase in CVCS borated water flow from the volume control tank (VCT) into an RCS cold leg. This results both in a net increase in coolant mass to the RCS and, if the VCT boron concentration is larger than that in the RCS, an increase in RCS boron concentration. During the initial phase of the transient, the boration can cause an insertion of negative reactivity, which in turn can result in a power and RCS pressure decrease. Because the power decrease and pressure decrease have opposite effects on the DNBR, there is very little sensitivity to the difference in boron concentration between the VCT and the RCS. As a result, the CVCS boron is assumed to be injected at the RCS boron concentration, and the event is analyzed for pressurizer overfill only.

The net addition of mass to the RCS by the CVCS will result in an increase in pressurizer level. The pressurizer high level alarm is set 15% above the normal programmed level and will alarm in the control room to alert the operator that a level increase is in progress. If left unmitigated, the reactor will trip on a high pressurizer water level signal. After the reactor trip, the CVCS charging pumps are assumed to continue to inject water, causing the potential for filling the pressurizer. The Barrier Performance evaluation addresses the maximum pressurizer level encountered during this transient and the time available for operator action to isolate the CVCS flow; however, no specific operator actions are assumed in the analysis.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause an RCP coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

The following automatic reactor trip signals are assumed to be available to provide protection from this transient:

- High pressurizer pressure
- High pressurizer water level

The availability and adequacy of instrumentation and control is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event as discussed in Section 15.0.0.5.

15.5.2.3 Core and System Performance

This event is not limiting with respect to fuel damage limits. As a result, DNBR and related fuel parameters (e.g., heat fluxes, and RCS temperatures) are not presented. A single case is analyzed to evaluate peak pressurizer water volume crediting operator actions to isolate CVCS as described in Section 15.5.2.4.

15.5.2.4 Barrier Performance

15.5.2.4.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following a CVCS malfunction that increases RCS inventory. The evaluation model also includes pressurizer spray and RCS safety valves. This evaluation model is described in Section 15.0.2.2.1. Additional details on the MARVEL-M code are provided in Reference 15.5-2.

15.5.2.4.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative results for a CVCS malfunction that increases RCS inventory:

- The initial power level is taken as 102 percent of the licensed core thermal power level. The nominal value of core power condition is described in Table 15.0-3.

- The initial reactor coolant temperature is 4°F below the nominal value and the initial pressurizer pressure is 30 psi above the nominal value. This combination of initial uncertainties minimizes the coolant shrinkage after reactor trip. The nominal values of reactor coolant temperature and pressure conditions are described in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4 corresponding to beginning of fuel cycle conditions. The Doppler power coefficient is assumed to be the minimum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- The reactor is assumed to be automatically tripped by the high pressurizer water level signal. Table 15.0-4 summarizes the trip setpoint and signal delay time used in the analysis.

- The analysis setpoint for the pressurizer high level alarm is conservatively assumed 20% above the normal programmed level.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. RCCA insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- The plant is assumed to be operating in manual rod control.

- The pressurizer heaters and pressurizer spray are assumed to operate as designed. This will minimize the time available for operator action (i.e., period of time between the high pressurizer level alarm and when the pressurizer fills).

- The pressurizer safety valves are modeled for this event. They are assumed to open at 2525 psia and be fully open at 2575 psia.

- Borated water from the volume control tank is assumed to be at the same concentration as the RCS.

- CVCS flow is conservatively assumed to be injected into the RCS cold legs by one charging pump from full power conditions at a constant 310 gpm. Letdown is assumed to be isolated.

15.5.2.4.3 Results

A single limiting case is analyzed to evaluate pressurizer overfill and the associated time available for manual actions to isolate the CVCS flow. The sequence and timing of major events for the CVCS malfunction that increases RCS inventory event is shown in Table 15.5.2-1.

Figures 15.5.2-1 through 15.5.2-5 are plots of the transient response of system parameters for the Barrier Performance Evaluation case.

In the evaluated case, the full-open failure of the charging flow control valve with one pump running leads to an addition of mass to the RCS resulting in an increase in the pressurizer water volume. Table 15.5.2-1 shows that the high pressurizer water level alarm occurs 404 seconds after the CVCS malfunction occurs. Table 15.5.2-1 shows that the reactor trips at 1062 seconds, as indicated by the distinctive drop in reactor power and RCS temperature at this time in Figures 15.5.2-1 and 15.5.2-4. The CVCS charging pump continues to inject water until the pressurizer fills, which occurs at 1176 seconds per Figure 15.5.2-3. Thus, there are 12.8 minutes available after the high pressurizer level alarm for the operator to perform actions to end the transient before the pressurizer fills.

The CVCS malfunction that increases RCS inventory event does not result in exceeding any reactor coolant pressure boundary or containment volume fission product barrier design limits. The results of the pressurizer water volume case demonstrate that the RCS pressure and main steam system pressure remain well below 110% of their respective system design pressures. Therefore, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained.

15.5.2.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the steam system piping failure evaluated in Section 15.1.5.

15.5.2.6 Conclusions

The chemical and volume control system malfunction that increases reactor coolant system inventory event does not challenge the DNBR 95/95 limit, and no fuel failures are predicted.

Sufficient time exists to enable operator action to prevent the pressurizer from filling and the RCS pressure and main steam system pressure remain well below 110% of their respective system design pressures, so the integrity of the reactor coolant pressure boundary and main steam system are maintained.

This event does not lead to a more serious fault condition.

Table 15.5.2-1
Time Sequence of Events for CVCS Malfunction that Increases Reactor Coolant Inventory

Event	Time (sec)
CVCS malfunction that increases reactor coolant inventory	0.0
High pressurizer level alarm	404
High pressurizer water level reactor trip analytical limit reached	1062
Reactor trip initiated (rod motion begins)	1064
Peak pressurizer water volume occurs	1176

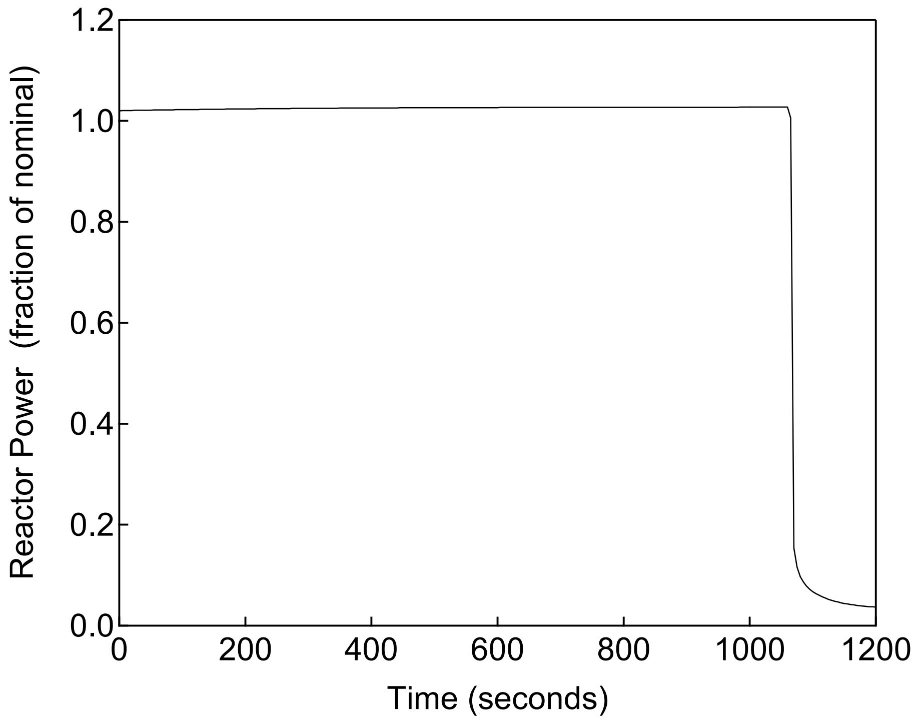


Figure 15.5.2-1 Reactor Power versus Time

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

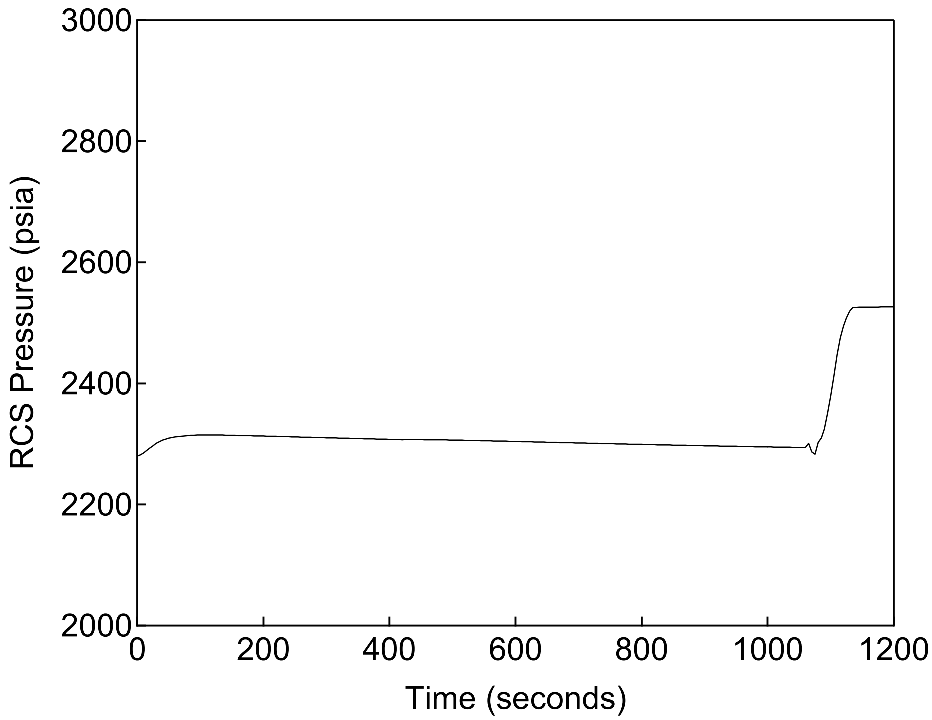


Figure 15.5.2-2 RCS Pressure versus Time

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

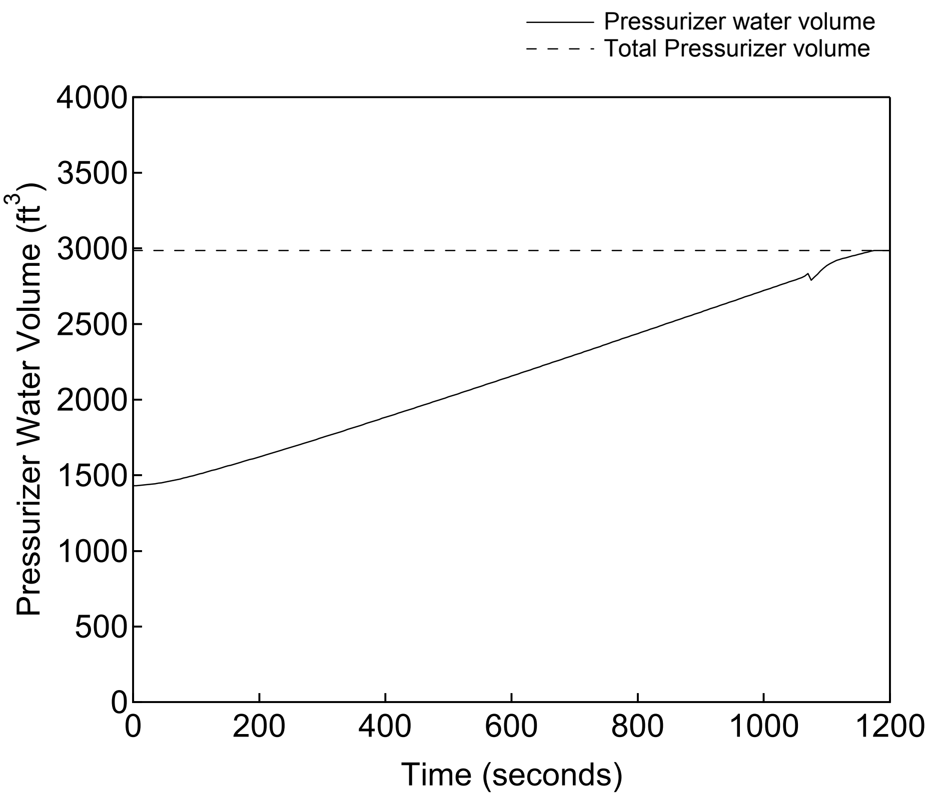


Figure 15.5.2-3 Pressurizer Water Volume versus Time

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

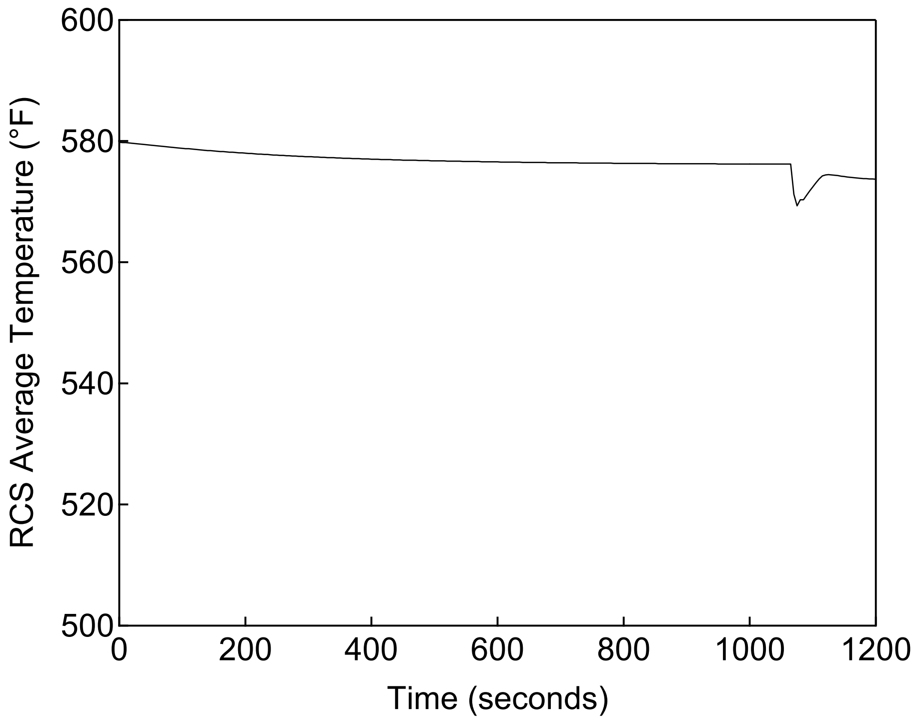


Figure 15.5.2-4 RCS Average Temperature versus Time

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

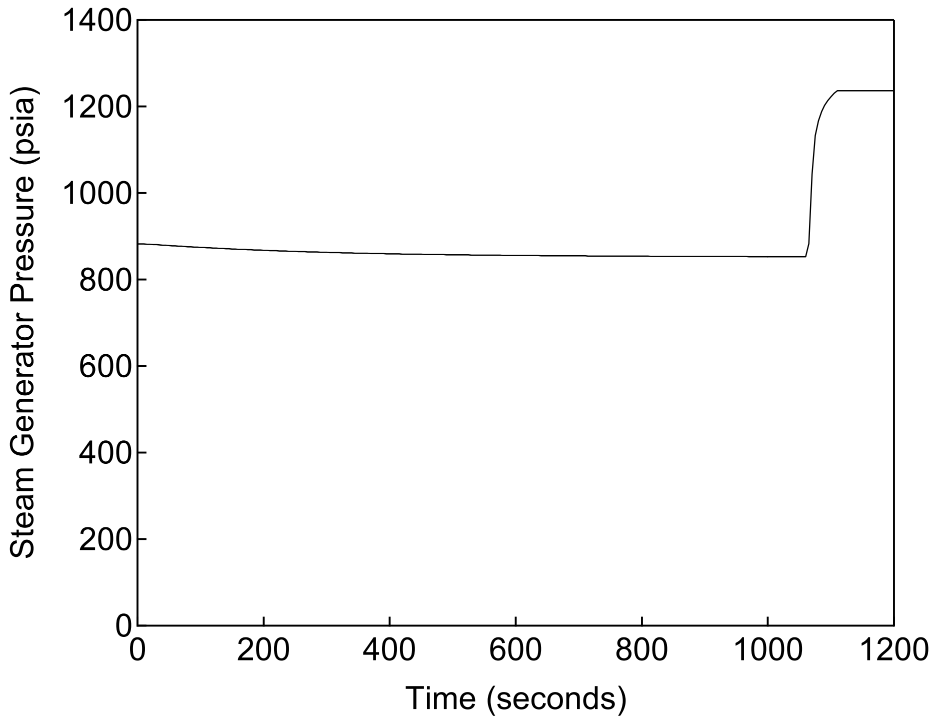


Figure 15.5.2-5 **Steam Generator Pressure versus Time**

Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

15.5.3 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.5.4 References

15.5-1 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.5-2 Non-LOCA Methodology, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.6 Decrease in Reactor Coolant Inventory

This section describes analyses that have been performed for events that could result in a decrease in reactor coolant inventory, which, in turn, can lead to a temperature increase in the reactor coolant system (RCS).

Analyses of the following events are described in this section:

- Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve
- Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment
- Radiological Consequences of Steam Generator Tube Failure
- Radiological Consequences of Main Steam Line Failure Outside Containment (BWR) (not applicable to the US-APWR)
- Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

These events are considered postulated accidents (PAs) as defined in Section 15.0.0.1, except for the inadvertent opening of a PWR pressurizer relief valve event which is classified as an anticipated operational occurrence (AOO).

The results of the applicable analyses determined that the most severe radiological consequences result from the major loss-of-coolant accident (LOCA) described in Section 15.6.5. The LOCA, chemical and volume control system (CVCS) letdown line break outside containment and the steam generator tube rupture (SGTR) events are also analyzed for radiological consequences. Radiological consequences for all other events analyzed in this section are bounded by these analyses.

15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve

15.6.1.1 Identification of Causes and Frequency Classification

An accidental depressurization of the reactor coolant system (RCS) could occur by the inadvertent opening of a pressurizer pressure relief valve. The causes could be a spurious electrical signal or an operator error. In the US-APWR, there are spring loaded safety relief valves (SRVs), motor operated safety depressurization valves (SDVs) and a motor operated depressurization valve (DV) used for the mitigation of severe accidents. A DV has more relief capacity than a SRV or a SDV, and will result in a more rapid depressurization upon opening. Therefore, the most severe core conditions for this event result from an inadvertent opening of a DV.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.6-3). Event frequency conditions are described in Section 15.0.0.1.

15.6.1.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the inadvertent opening of a DV event is described in the results section.

The inadvertent opening of a DV initially results in a decrease of the RCS inventory and pressure. Assuming that the rod control system is in automatic mode, it responds by maintaining power and average coolant temperature until the reactor trips.

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

This event results in a turbine trip when initiated from at-power conditions. A turbine trip could cause a disturbance to the utility grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause a reactor coolant pump (RCP) coastdown. As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs, the case where offsite power is unavailable is not presented.

The following automatic reactor trip signals are assumed to be available to provide protection from this transient:

- Over temperature ΔT
- Low pressurizer pressure

The availability and adequacy of instrumentation and control is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event (as discussed in Section 15.0.0.5).

15.6.1.3 Core and System Performance

15.6.1.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of reactor power, RCS pressure, and reactor coolant temperature following an inadvertent opening of a pressurizer pressure relief valve. This evaluation model is described in

Section 15.0.2.2.1. Additional details on the MARVEL-M code are provided in Reference 15.6-1.

The DNBR calculations use the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

15.6.1.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for an inadvertent DV opening event:

- Consistent with the use of RTDP, the assumed initial values of reactor power, reactor coolant average temperature, and RCS pressure are assumed to be the nominal values as defined in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to have the maximum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- The primary coolant blowdown rate is assumed to be 120% of the rated capacity of one DV. Full opening of a DV produces more severe depressurization results than any other type of pressurizer pressure relief valve.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. rod cluster control assembly (RCCA) insertion characteristics assumed in analysis is described in Section 15.0.0.2.5.

- The reactor is assumed to be automatically tripped by the low pressurizer pressure signal. Table 15.0-4 summarizes the trip setpoint and signal delay time assumed in the analysis.

- The rod control system is assumed to be in the automatic mode to maintain the core at full power until the reactor is tripped by the RTS. This assumption results in a more severe transient than if the rod control system was not in automatic mode. Therefore, normal automatic rod control systems are required to function.

15.6.1.3.3 Results

Table 15.6.1-1 lists the key events and the times at which they occur, relative to the inadvertent opening of a DV with offsite power available.

Figures 15.6.1-1 through 15.6.1-7 are plots of system parameters versus time for the inadvertent opening of a DV. Figure 15.6.1-1 shows that reactor power remains at full power until the trip occurs on low pressurizer pressure. The RCS pressure history and reactor coolant average temperature history are shown in Figures 15.6.1-3 and 15.6.1-6, respectively. The pressure drops more rapidly when core heat generation is reduced via the trip. Figure 15.6.1-7 shows that the DNBR decreases initially but increases rapidly following the trip. The DNBR remains above the 95/95 limit throughout the transient; therefore, fuel integrity is not degraded.

The hot spot heat flux is presented as the core heat flux since it is the parameter most closely related to DNBR. There is no sort-term impact to containment from the depressurization valve flow shown in Figure 15.6.1-5, consistent with the average temperature response. Inlet coolant temperature and core average temperature are not provided since the average temperature response is more appropriately represented by the reactor coolant average temperature. The main steam system design pressure will not be challenged during a depressurization vent, and therefore, steam line pressure is not presented.

Because there is significant subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided; these are not key parameters for this event.

This analysis demonstrates that no fuel failures occur from initiation through post-trip stable conditions. The long term portion of the transient is bounded by the small break LOCA analysis described in Section 15.6.5 as part of the LOCA accidents.

RCP seal reliability and integrity during loss of alternating-current power and loss of coolant to the seals (e.g., a result of containment isolation) must comply with 10 CFR 50.34(f)(1)(iii). For this event, the containment is isolated by the emergency core cooling system actuation signal. Therefore, seal water to the reactor coolant pumps may be lost. RCP seal integrity following containment isolation is discussed in Section 15.0.0.9.

15.6.1.4 Barrier Performance

The breach in the reactor coolant pressure boundary is the initiating condition of this event. Since the RCS pressure decreases once the DV is opened, the maximum RCS pressure for this transient is the initial RCS pressure, which is assumed to be the maximum nominal RCS pressure as defined in Table 15.0-3.

This transient does not challenge the main steam system pressure design limits.

15.6.1.5 Radiological Consequences

An inadvertent opening of a DV releases primary coolant, which is contaminated, to the pressurizer relief tank (PRT). However, even assuming a direct release to the containment atmosphere, the radiological consequences of this event would be substantially less than that of a LOCA (Section 15.6.5) because less primary coolant is released and fuel damage is not predicted as a result of this event.

15.6.1.6 Conclusions

The resulting transient following an inadvertent opening of a DV does not cause the minimum DNBR to decrease below the 95/95 limit and, therefore, no fuel failures are predicted. Additionally, since the breach in the reactor coolant pressure boundary is the initiating condition of this event there is no need to perform a barrier evaluation.

The radiological consequences of this event are substantially less than that of the LOCA analyzed in Section 15.6.5.

This event does not lead to a more serious fault condition.

Table 15.6.1-1
Time Sequence of Events for Inadvertent Opening of a Depressurization Valve - DNBR Analysis

Event	Time (sec)
Depressurization Valve fully opens	0.0
Low pressurizer pressure analytical limit reached	28.3
Reactor trip initiated (rod motion begins)	30.1
Minimum DNBR occurs	30.5

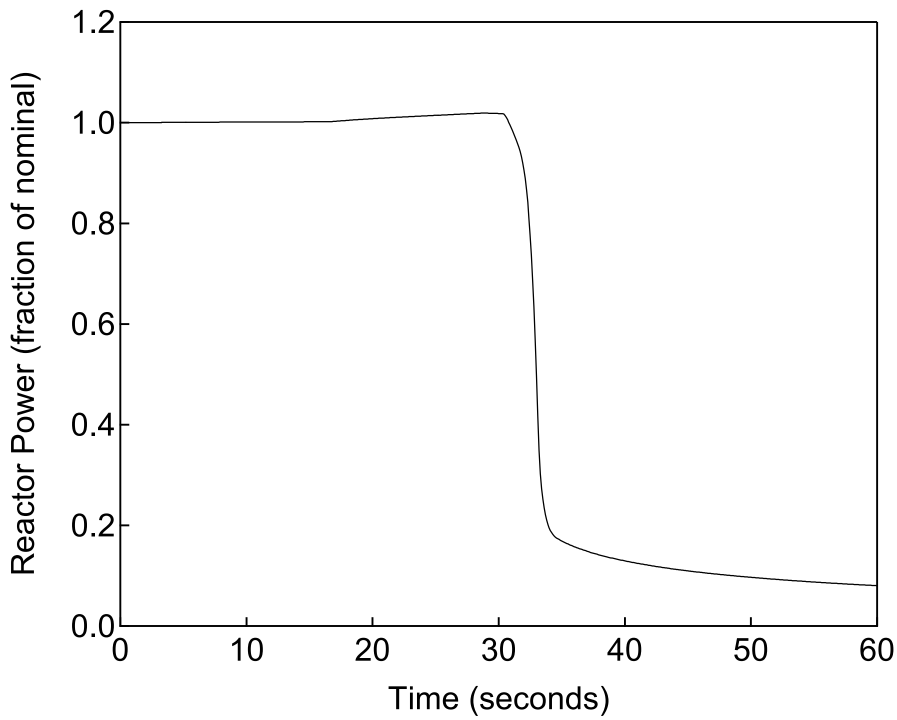


Figure 15.6.1-1 Reactor Power versus Time

Inadvertent Opening of a Depressurization Valve

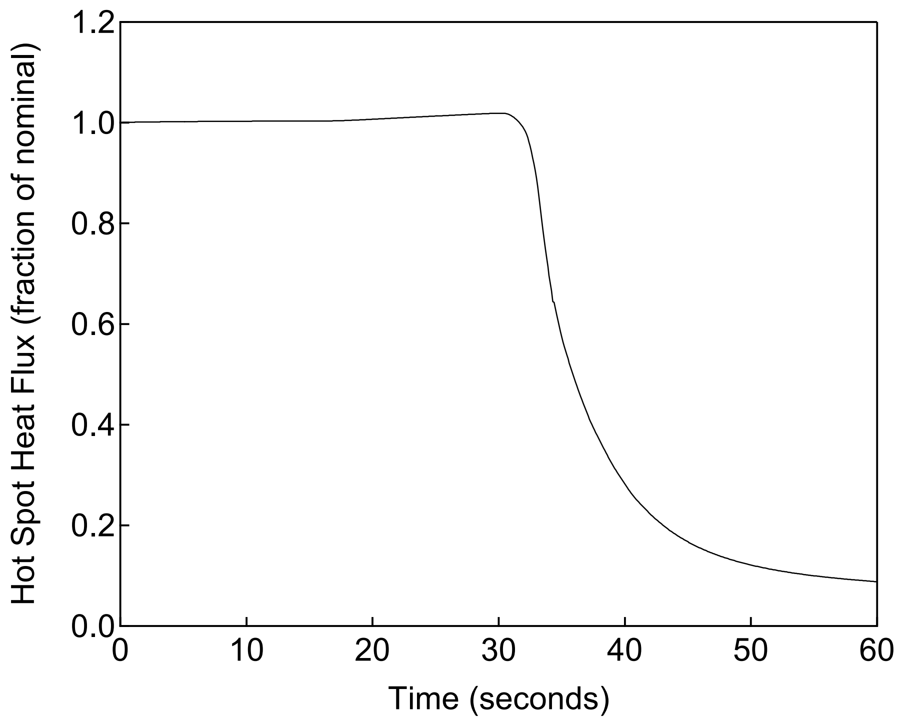


Figure 15.6.1-2 Hot Spot Heat Flux versus Time

Inadvertent Opening of a Depressurization Valve

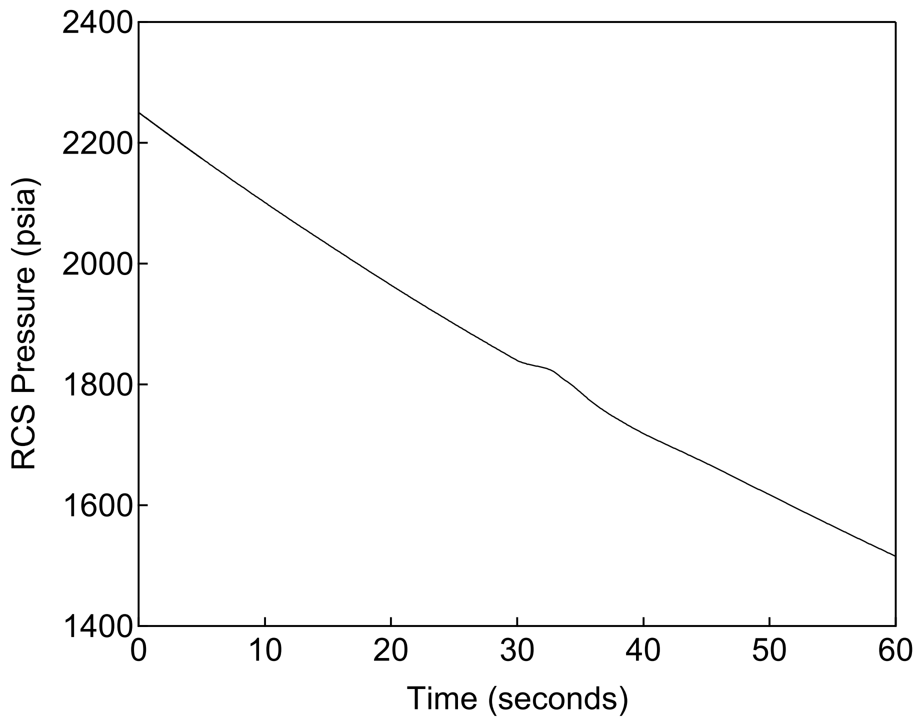


Figure 15.6.1-3 RCS Pressure versus Time

Inadvertent Opening of a Depressurization Valve

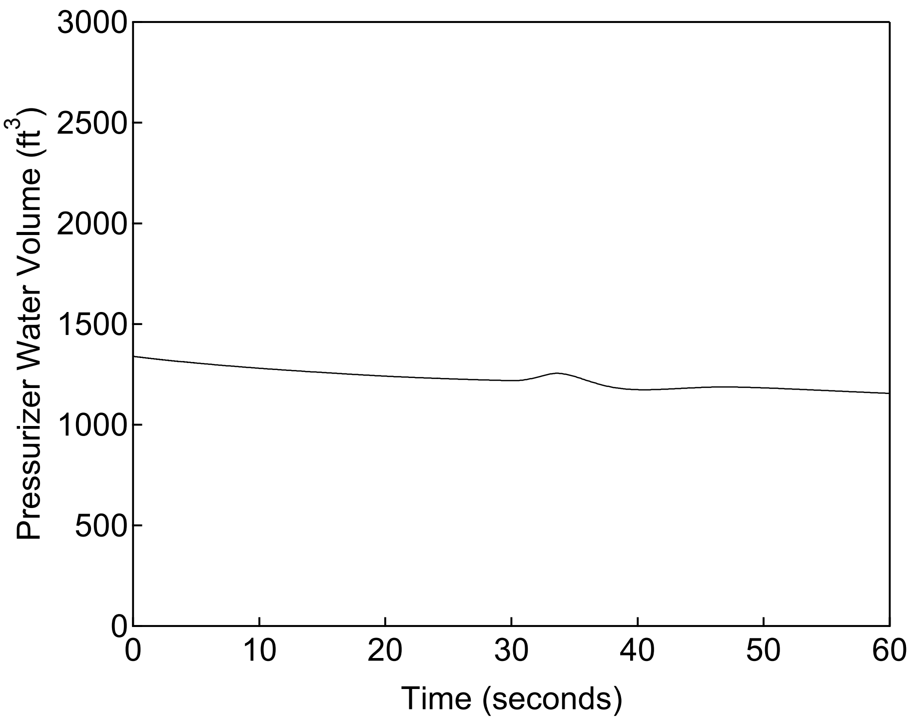


Figure 15.6.1-4 Pressurizer Water Volume versus Time

Inadvertent Opening of a Depressurization Valve

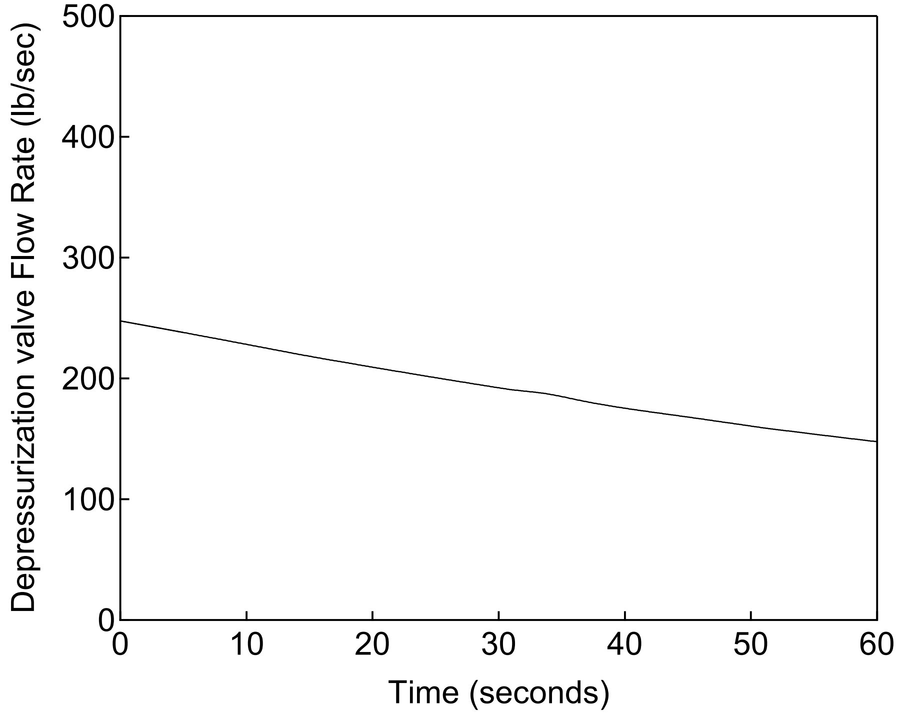


Figure 15.6.1-5 Depressurization Valve Flow Rate versus Time

Inadvertent Opening of a Depressurization Valve

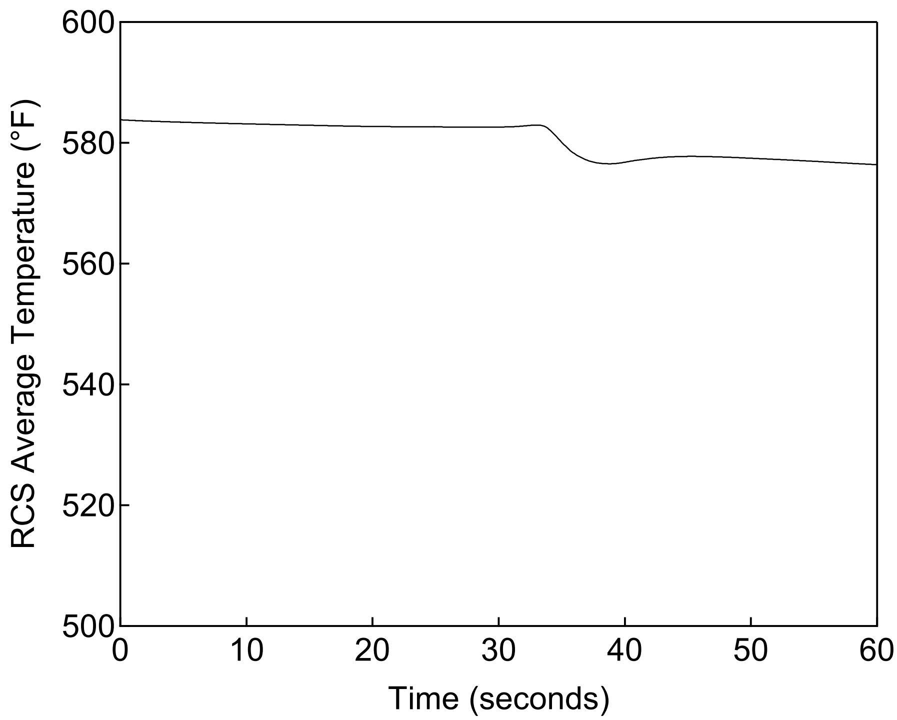


Figure 15.6.1-6 RCS Average Temperature versus Time

Inadvertent Opening of a Depressurization Valve

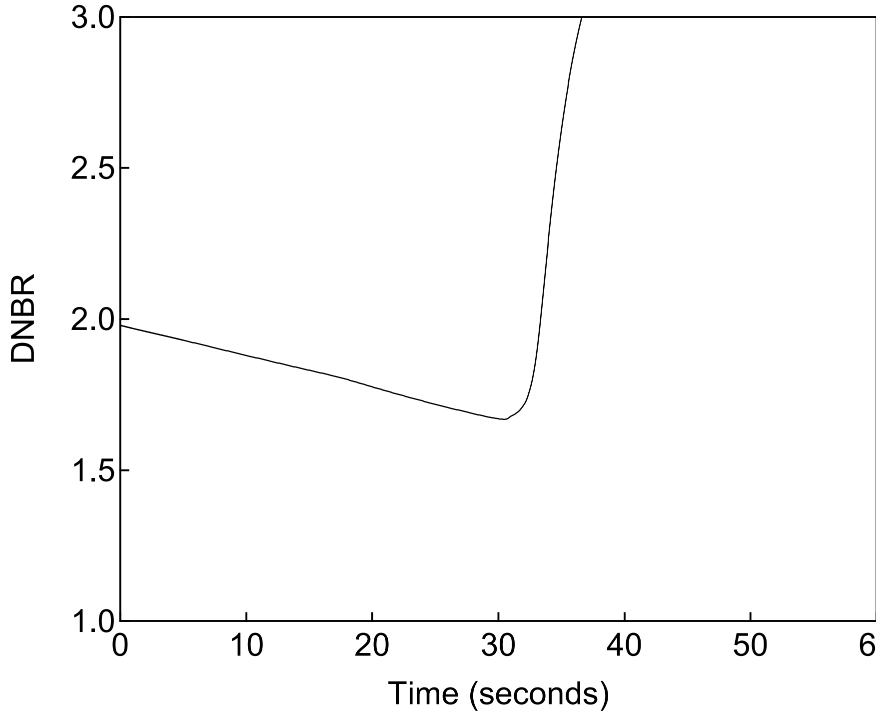


Figure 15.6.1-7 DNBR versus Time

Inadvertent Opening of a Depressurization Valve

15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes and Frequency Classification

A failure of small lines carrying primary coolant outside containment results in radiological consequences, resulting from a release containing the radionuclide concentration of the reactor coolant. The cause may be a leak in the instrument, sample, or chemical and volume control system (CVCS) letdown lines due to manufacturing defect, corrosion, or maintenance activities.

This event is classified as an anticipated operational occurrence (AOO). Event frequency conditions are described in Section 15.0.0.1.

15.6.2.2 Sequence of Events and Systems Operation

The failure of small lines carrying primary coolant outside containment is the reactor coolant system (RCS) sample lines and the CVCS letdown line to the demineralizers. No instrument lines carry primary coolant outside the containment. The sample lines and the CVCS letdown line to the demineralizers are provided with isolation valves on both sides of the containment wall and are designed in accordance with the requirements of GDC 55.

There are sample lines from the hot legs of reactor coolant loops, the steam and liquid space of the pressurizer, and the CVCS letdown line penetrating the containment. The sample line isolation valves and CVCS letdown line isolation valves are normally open. For small lines that meet GDC 55, the failure is assumed to occur downstream of the outboard containment isolation valve in conjunction with a single failure of one of the two containment isolation valves.

The amount of reactor coolant released outside the containment is determined by the time required to detect such a failure and the time required to isolate the failure (i.e., time to close the operable isolation valve). The amount of reactor coolant released is conservatively estimated by assuming critical flow at the small line break location with the reactor coolant fluid enthalpy corresponding to normal reactor operating conditions.

For the sample line break outside the containment, the loss of sample flow provides indication of a break to plant personnel since the loss of coolant reduces the volume control tank level and creates a demand for automatic makeup from CVCS. Frequent operation of the automatic makeup system will provide indication of the loss of reactor coolant. Upon indication of a sample line break, the operator takes action to isolate the break by closing the operable isolation valve for the damaged line. The operator is assumed to detect and isolate the break within 45 minutes.

For the CVCS letdown break outside the containment, the flow leaving the containment is at a low temperature, because the flow stream passes through the CVCS heat exchangers. This event is not analyzed, because the postulated sample line break is more limiting.

15.6.2.3 Core and System Performance

The size of a sample line is smaller than the break size corresponding to the makeup flow rate. Thus, the pressurizer water level can be maintained for a break in that line. As the makeup water is sufficient to maintain pressurizer water level within its normal operational range, no fuel damage results from this transient. The transient is terminated when the operator isolates the break and performs an orderly shutdown.

15.6.2.4 Barrier Performance

The RCS pressure remains well below 110% of the design pressure. However, this event postulates that both the primary system and containment systems fail due to a failure of small lines carrying primary coolant outside containment.

15.6.2.5 Radiological Consequences

The radiological consequences evaluation for this event uses the alternative source term (AST) guidance documented in Reference 15.6-4.

The radiological consequences evaluation assumes the reactor has been operating with at the maximum allowable limit for reactor coolant concentration. The equilibrium concentrations assumed in the analysis are based on Technical Specification coolant concentration limits.

In addition, it is assumed that iodine spiking (concurrent iodine spikes) occurs at the time of the accident. The concurrent iodine spike case postulates the iodine release rate from the detective rods increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (1 μCi/g dose equivalent (DE) I-131) specified in the Technical Specifications. This is consistent with the methodology given in Reference 15.6-4.

The activity released from the defect fuels is assumed to be released instantaneously and homogeneously through the reactor coolant. The reactor coolant which is spilled in the auxiliary building collects in the floor drain sumps before being pumped to the radwaste treatment system. Therefore, the only release paths that present a radiological hazard involve the volatile fraction of spilled coolant, which are the noble gases and iodines.

In the failure of the sample line, the reactor coolant that is spilled from the break is assumed to be at maximum normal RCS pressure. A large portion of the flow flashes to steam, and the iodine in the flashed liquid is assumed to become airborne.

The iodine and noble gases are assumed to be released directly to the environment with no credit for depletion or filtration. In reality, a large fraction of the airborne iodine is expected to be deposited on building surfaces or be removed by the building filtration system.

15.6.2.5.1 Evaluation Model

Mathematical models used in the analysis are described in the following sections:

- The offsite and onsite doses are calculated with the RADTRAD code.
- The atmospheric dispersion factors (χ/Q values) used in the analysis are described in Section 15.0.3.3.
- The total effective dose equivalent (TEDE) doses to a receptor at the exclusion area boundary (EAB) and outer boundary of the low-population zone (LPZ) are analyzed using the models described in Section 15.0.3.1 and Appendix 15A.

Figure 15A-7 depicts the leakage sources to the environment modeled in the dose computation.

All noble gases and iodines are released to the environment with no consideration given to radioactive decay or cloud depletion by ground deposition during transport to the EAB and LPZ. Hence, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the event.

15.6.2.5.2 Input Parameters and Initial Conditions

The major assumptions and parameters used in the analysis are provided in Table 15.6.2-1, and Tables 15.0-10 through 15.0-14

- The reactor coolant iodine concentration is based on a concurrent iodine spike corresponding to 500 times the release rate of iodine at the equilibrium value (1 μCi/g DE I-131).
- The noble gas concentration in the reactor coolant is based on the maximum equilibrium value, which is 300 μCi/g DE Xe-133.
- Break flow rate is assumed to be 97 gpm at density of 62.4 lb/ft^3.
- Fraction of reactor coolant flashing is 47% based on initial reactor coolant enthalpy at maximum normal RCS pressure and final reactor coolant enthalpy at atmospheric pressure.
- Upon indication of a sample line break, the operator takes action to isolate the break by closing the operable isolation valve for the damaged line (See Table 7.5-5). The operator is assumed to detect and isolate the break within 45 minutes.

- The only filtration system considered in the analysis which limits the consequences of the failure of small lines carrying primary coolant outside containment is the main control room (MCR) heating, ventilation, and air conditioning(HVAC) system.

- The χ/Q values and breathing rates are listed in Table 15.0-13. The breathing rates are obtained from NRC Regulatory Guide 1.183 (Ref. 15.6-4).

15.6.2.5.3 Results

As shown in Table 15.6.2-2, the calculated TEDE doses are determined to be 1.5 rem at the EAB and 0.60 rem at the LPZ outer boundary.

These doses are less than 10% of the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34. The dose guideline is based on the acceptance criterion given in SRP 15.6.2.

The doses for the MCR for the failure of small lines carrying primary coolant outside containment are bounded by the doses calculated for the loss-of-coolant accident (LOCA) event described in subsection 15.6.5.5. Consequently, no doses are provided for the failure of small lines carrying primary coolant outside containment.

15.6.2.6 Conclusions

A single charging pump provides enough makeup water to the bounding small line break size so that the pressurizer level can be maintained in its normal operational range throughout the transient. Therefore, no fuel damage results from this event.

The resultant doses are well within the guideline values of 10 CFR 50.34.

Table 15.6.2-1
Parameters Used in Evaluating the Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

Parameter	Value
Core thermal power level (MWt)	4540 (2% above the design core thermal power)
Reactor coolant iodine concentration	The reactor coolant iodine concentration is based on a concurrent iodine spike corresponding to 500 times the release rate of iodine at the equilibrium value (1 μCi/g DE I-131).
Reactor coolant noble gas concentration	300 μCi/g DE Xe-133
Break flow rate (gpm)	97 (at density of 62.4 lb/ft3)
Fraction of reactor coolant flashing	47% based on initial reactor coolant enthalpy at maximum normal RCS pressure and final reactor coolant enthalpy at atmospheric pressure
Duration of accident (min)	45
χ/Q	See Table 15.0-13.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.6.2-2
Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

Dose Location	TEDE Dose (rem)
EAB dose (0 to 2 hours)	1.5
LPZ boundary dose	0.60

15.6.3 Radiological Consequences of Steam Generator Tube Failure

15.6.3.1 Identification of Causes and Frequency Classification

In the steam generator tube rupture (SGTR) event, the complete severance of a single steam generator tube is assumed. The event is assumed to take place at full power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defect fuels. The event leads to leakage of radioactive coolant from the RCS to the secondary system. In the event of a coincident loss of offsite power or the failure of the steam dump system, atmospheric discharge of radioactivity can take place via the main steam relief valves (MSRVs), or the main steam safety valves (MSSVs), if the setpoint is reached. The assumption of a complete tube severance is considered conservative because the tube material Alloy 690 is a corrosion resistant and ductile material. Since the radioactivity in the secondary system is under continual surveillance, an accumulation of activity as a result of such minor leaks that exceeds the limits established in Technical Specifications is not permitted during operation.

The operator is expected to recognize the occurrence of a SGTR event, to identify and isolate the ruptured steam generator, and to take appropriate actions to stabilize the plant. These operator actions should be performed in a timely manner to minimize contamination of the secondary system and the release of radioactivity to the atmosphere. In addition, recovery procedures should be carried out on a time scale that ensures that the break flow to the secondary system is terminated before the water level in the ruptured steam generator reaches the steam generator outlet nozzle.

This event is classified as a postulated accident (PA). Historically, the SGTR event was classified as a Condition IV event as defined by ANSI N18.2 (Ref. 15.6-3). Event frequency conditions are described in Section 15.0.0.1.

In addition to the SRP acceptance criteria for PAs, MHI conservatively adopts two additional acceptance criteria: (1) to not allow steam generator overfill and (2) to maintain the reactor coolant system (RCS) and main steam pressures below 110% of their respective design pressure to assure that rupture of the primary or steam system piping does not occur.

15.6.3.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the SGTR event is described in the results section. This sequence can be summarized as follows:

- Low pressurizer pressure and low pressurizer level alarms are actuated. The flow from the charging pumps increases and the pressurizer heaters are actuated in an attempt to maintain pressurizer level and RCS pressure. On the secondary side, steam flow/feedwater flow mismatch occurs because the primary-to-secondary leak flow increases the water level of the steam generator.

- The main steam line radiation monitors and the condenser air ejector radiation monitor indicate an increase in radioactivity in the secondary system. The steam generator blowdown liquid monitor is also available.

- The continuous loss of reactor coolant inventory leads to a reactor trip generated by low pressurizer pressure or over temperature ΔT. Plant cooldown after the reactor trip leads to a rapid decrease of both RCS pressure and pressurizer level, which is counteracted by the chemical and volume control system flow and operation of the pressurizer heaters. If the pressurizer pressure decreases below the low pressurizer pressure setpoint, emergency core cooling system (ECCS) is actuated. The ECCS actuation signal starts the safety injection pumps and also trips the reactor coolant pumps which then coast down to natural circulation conditions. In addition, an ECCS actuation signal provides feedwater isolation by automatically tripping the main feedwater pumps and fully closing all regulation valves and feedwater isolation valves in the feedwater system. For the purposes of this analysis, if the reactor trip system (RTS) has not automatically tripped the reactor, operators are assumed to manually trip the reactor 15 minutes after SGTR initiation.

- Operators can identify the ruptured steam generator from the available information, and can start isolating the ruptured steam generator from the remaining intact steam generators.

- With the turbine tripped (caused by the reactor trip), the steam can be diverted to the condenser via the turbine bypass valves, if offsite power is still available. In the event of a loss of offsite power, the steam dump valves remain closed and the steam generator pressure increases rapidly, resulting in the opening of the MSRVs (or safety valves if their setpoint is reached) and steam is discharged to the atmosphere.

- Operators start to reduce the RCS temperature by opening the main steam depressurization valves (MSDVs) on the intact steam generators.

- The makeup water from the safety injection flow increases the RCS water inventory, and stabilizes the RCS pressure and pressurizer water level.

- Operators reduce the RCS pressure by opening a safety depressurization valve(SDV) until the primary-to-secondary pressure balance is attained.

- After the safety injection is terminated, the break flow eventually stops when the RCS pressure equalizes with the ruptured steam generator pressure. At this point, the plant is stabilized. Residual heat removal (RHR) is initiated to provide long term cooling after RCS temperature is sufficiently reduced via heat removal by the intact steam generators.

In the event of a SGTR, the plant operators must be able to diagnose the event and perform the required recovery actions to bring the plant to a stable configuration. The operator actions for SGTR recovery are proceduralized in the Emergency Operating

Procedures. The major operation actions required for the recovery from a SGTR are discussed in Section 15.6.3.4.2.

A loss of one emergency feedwater system (EFWS) train is assumed as a limiting single failure for the cases described in Section 15.6.3.4.2.

The following signals are assumed to be available to automatically trip the reactor and therefore provide protection from this transient:

- Low pressurizer pressure
- Over temperature ΔT
- High-high steam generator water level
- ECCS actuation

The following engineered safety features (ESF) are assumed to be available to mitigate the accident:

- EFWS automatic actuation
- Emergency feedwater (EFW) isolation
- ECCS

The availability and adequacy of instrumentation and control is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event (as discussed in Section 15.0.0.5).

15.6.3.3 Core and System Performance

A SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. An analysis must be performed to assure that the offsite radiological consequences resulting from a SGTR are within the guidelines.

Fuel failure due to DNB occurrence is only an issue prior to reactor trip. The primary parameters of concern for DNB – reactor power and average core temperature – remain constant between the initiation of the SGTR and the reactor trip. The RCS pressure decreases due to the rupture of a steam generator tube. However, the reduction in RCS pressure is small enough that it does not result in low pressurizer pressure signal occurrence after the initiation of the SGTR event. Therefore, the effect of the RCS pressure reduction does not result in DNB occurrence, and DNB is not separately evaluated.

15.6.3.4 Barrier Performance

15.6.3.4.1 Evaluation Model

The SGTR event uses the MARVEL-M code capability to calculate primary-to-secondary flow. The initial break flow is conservatively determined assuming critical flow calculated using the primary pressure at the break location, accounting for the pressure drop between the tube inlet or outlet and the break location. From that point on, the break flow is calculated by MARVEL-M as a function of the square root of the primary-to-secondary differential pressure, scaled to match the initial flow. Conservatism of this flow model was evaluated in the Non-LOCA Methodology topical report (Ref. 15.6-1).

The analysis covers the time period from the initiation of the event up to the termination of the primary-to-secondary flow. The MARVEL-M analysis simulates the RTS, the automatic actuation of the ESF systems, and major operator actions to establish steam generator cooling using the intact steam generators, manual opening of the MSDVs, and manual opening of the SDV.

Two cases are analyzed for barrier performance. One case evaluates primary-to-secondary break flow and the mass releases via the secondary system, and the other case verifies that the ruptured steam generator does not overfill. The result of the primary-to-secondary break flow and the mass releases case is used to evaluate radiological dose release to the environment. The primary differences between the two cases involve the assumptions regarding the operation of feedwater control and the actuation of the MSRV, as discussed below.

15.6.3.4.2 Input Parameters and Initial Conditions

The accident is modeled as a double-end severance of a steam generator tube located at the top of the tube sheet on the outlet (or the “cold side”) of the steam generator. The accident is assumed to take place at power.

The following bullets summarize the major input parameters and assumptions used in the case for radiological dose evaluation:

- The initial power level is assumed at 102% of the licensed core thermal power level with initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi above the nominal value. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

- Initial pressurizer water level is assumed at its expected nominal programmed level without uncertainty applied.

- Initial steam generator water level is assumed at its expected nominal programmed level with negative uncertainty applied.

- The assumed single failure is the loss of one EFWS train, which results in one of the remaining steam generators not receiving EFW flow. EFW flow supplied to each intact steam generator is assumed to be at the minimum flow rate from the time the system is initiated until EFW isolation. The ruptured steam generator is also supplied with EFW flow at the minimum flow rate.

- The analysis assumes that a coincident loss of offsite power occurs at the time of reactor trip.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to have the maximum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. Rod cluster control assembly insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- The pressurizer heater, CVCS, MSRVs, and feedwater control systems are assumed to be available for the radiological dose evaluation case.

- An additional, conservative failure of the MSRV on the ruptured steam generator is assumed for the radiological consequence analysis case. Failure of this MSRV in the open position causes an uncontrolled depressurization of the ruptured steam generator. In the analysis, the MSRV on the ruptured steam generator is assumed to fail open when the MSDVs of the intact steam generators are opened. Because the ECCS is initiated as a result of the RCS cooling operation, this assumption increases the primary-to-secondary pressure, thus, increases primary-to-secondary leakage and mass release to the atmosphere. The MSRV on the ruptured steam generator is eventually automatically isolated when the associated block valve is closed by the low main steam line pressure signal.

- The major operator actions required for the recovery from a SGTR are summarized below. These operator actions are included in the MARVEL-M simulation.

(a) Detection of the accident

A SGTR event initiates several indications in the control room, including the reduction in pressurizer water level, the reduction in pressurizer pressure, and the increase in water level in the ruptured steam generator. The event can also be detected by the steam generator blowdown radiation monitors, the steam condenser ejector radiation monitors, and the main steam line N-16 high-sensitivity radiation monitors installed on each steam line (the high radiation level alarms occur within 2 minutes of the SGTR initiation).

(b) Identification of the ruptured steam generator and reactor trip

Operators can identify the ruptured steam generator from the N-16 radiation monitors and from the increase in the water level in the ruptured steam generator.

A time margin of 10 minutes is assumed for operators to identify the ruptured steam generator after the audible alarms indicate the event has occurred. If the RTS has not automatically tripped the reactor, operators are assumed to manually trip the reactor 15 minutes after SGTR initiation.

(c) Isolating the ruptured steam generator

Once a tube rupture has been identified, recovery actions begin by isolating the ruptured steam generator from the remaining intact steam generators, and by isolating feedwater flow to the ruptured steam generator. The actions to isolate the ruptured steam generator are assumed to be completed within 5 minutes after the reactor trip.

(d) Reducing the RCS temperature

Operators are assumed to start to reduce the RCS temperature by opening the MSDVs on the intact steam generators 5 minutes after the isolation of the ruptured steam generator.

(e) Depressurizing the RCS

After the RCS hot leg temperatures of the intact loops are reduced enough to assure subcooling, and even though the RCS pressure has been reduced to the pressure of the ruptured steam generator, operators further reduce the RCS pressure by opening a SDV until the primary-to-secondary pressure balance is attained.

(f) Terminating the ECCS

ECCS must be terminated to stop primary-to-secondary leakage. The ECCS is terminated manually according to the ECCS termination criteria specified in the Emergency Operating Procedures. After ECCS is terminated, leakage flow will continue until the RCS and steam generator pressures equalize.

- ECCS is assumed to be provided by all four safety injection (SI) pumps at the maximum flow rate.
- The initial primary-to-secondary leakage is assumed to be 55 lb/sec and the transient leakage is calculated in proportion to the square root of primary-to-secondary pressure.
- The following conditions must be satisfied in order for SI to be terminated:
    - RCS subcooling is greater than 50°F (allowance for uncertainty).

- Minimum EFW is available, or the water level in at least one steam generator is in the narrow range.

- Pressurizer level is greater than 5% (allowance for uncertainty).

- The RCS pressure is increasing (the analysis assumes that the reactor operator notices this increase once the RCS pressure increases 142 psi from its minimum pressure).

The same input parameters as in the case for radiological dose evaluation analysis are used for the steam generator overfill analysis with the exception of the following items:

- The initial power level is assumed at 102% of the licensed core thermal power level with initial reactor coolant temperature 4°F below the nominal value and the pressurizer pressure 30 psi above the nominal value. The nominal values for core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

- EFW flow supplied to the ruptured steam generator is assumed to be at the maximum flow rate.

- No credit is taken for the feedwater control system.

- The automatic EFW isolation is conservatively ignored to maximize the EFW supply to the ruptured steam generator.

- No credit is taken for the MSRV in the steam generator overfilled analysis case.

- Initial water level in the ruptured steam generator is assumed at its expected nominal programmed level with positive uncertainty applied.

15.6.3.4.3 Results

(1) Radiological Dose Evaluation Case

After the steam generator tube rupture occurs, reactor coolant flow leaks from the primary side into the secondary side of the ruptured steam generator. As a result of the loss of reactor coolant, the pressurizer level decreases as shown in Figure 15.6.3-2. As the vapor space in the pressurizer expands, the RCS pressure also decreases as shown in Figure 15.6.3-1.

The reactor trip automatically leads to the turbine trip. The main feedwater is terminated and the EFW is automatically actuated. With the turbine stop valves close, the steam dump system is designed to actuate to divert the steam to the condenser. Due to the loss of the offsite power, the condenser vacuum is also lost and the steam dump valves remain closed. After the reactor trip, the secondary side pressure continues to increase until the MSSVs open, as shown in Figure 15.6.3-5.

When the pressurizer pressure reaches the ECCS setpoint, safety injection will start and deliver flow to the RCS at pressures below the pump shutoff pressure.

Immediately following the reactor trip, the temperature differential across the reactor decreases as core power reduces to decay heat level (see Figures 15.6.3-3 and 15.6.3-4). As the accident progresses, the temperature differentials gradually increase. For the ruptured steam generator, the cold leg temperature also continues to decrease until the failed MSRV is closed.

The sequence of events for this transient is summarized in Table 15.6.3-1.

Key Operator Actions and their timing as determined by analysis are as follows:

(a) Detection of the accident

Operators detect the accident by the N-16 high-sensitivity radiation level alarm within 120 seconds from the SGTR initiation.

(b) Identification of the ruptured steam generator and reactor trip

The ruptured steam generator is identified 600 seconds after the audible alarms which indicate that the event has occurred and operators trip the reactor manually at 900 seconds.

(c) Isolating the ruptured steam generator

The main steam isolation valve (MSIV) is closed 1200 seconds after SGTR initiation. (Therefore, EFW flow is not provided for the ruptured steam generator since the MSIV closed before the EFW initiated).

(d) Reducing the RCS temperature

To reduce the RCS temperature, the MSDVs of the intact steam generators are opened by operators 1500 seconds after SGTR initiation to reduce the RCS temperature.

At the same time, the MSRV on the ruptured steam generator is assumed to fail open. This failure causes the ruptured steam generator to rapidly depressurize, resulting in an increase in primary-to-secondary leakage and energy transfer. The MSRV on the ruptured steam generator is automatically isolated at 1826 seconds when the associated block valve is closed by the low main steam line pressure signal.

(e) Depressurizing the RCS

Operators reduce the RCS pressure by opening a SDV at 2717 seconds until the primary-to-secondary pressure balance is attained.

(f) Terminating the ECCS

After the successful establishment of a secondary heat sink, adequate subcooling margin, and RCS depressurization, SI is no longer needed and should be stopped to prevent repressurization of the primary system.

All the requirements for the termination of SI described in Section 15.6.3.4.2 are met and SI termination occurs 2880 seconds into the transient. After SI termination, the RCS pressure decreases as shown in Figure 15.6.3-1. Figure 15.6.3-7 shows that the primary-to-secondary leak flow continues after the SI is stopped until the pressures of the RCS and the ruptured steam generator equalize, which occurs at 4183 seconds.

The water volume in the ruptured steam generator as a function of time is shown in Figure 15.6.3-6. It can be seen that the water volume in the ruptured steam generator is 3030 ft^3 when the break flow stops, which is significantly less than the total steam generator volume of 7220 ft^3. Thus, the steam generator does not overfill. Radiological calculations were performed based on parameters shown in Figures 15.6.3-8 through 15.6.3-10.

(2) Steam Generator Overfill Case

In the steam generator overfill case, no credit is taken for the feedwater control system, which causes the increase of the water level in the ruptured steam generator. Therefore, a reactor trip is automatically initiated by the high-high steam generator water level signal. The main feedwater system is also isolated by the high-high steam generator water level signal. Figures 15.6.3-13 through 15.6.3-21 are plots of the system parameters versus time for the steam generator overfill case. Table 15.6.3-2 lists the key events and times at which they occur relative to the SGTR event for the steam generator overfill case.

The water volume in the ruptured steam generator as a function of time is shown in Figure 15.6.3-18. It can be seen that the water volume in the ruptured steam generator is 6980 ft^3 when the break flow stops, which is less than the total steam generator volume of 7220 ft^3. Thus, the steam generator does not overfill.

(3) Mass Releases for Radiological Calculation

The mass release of a SGTR event is used to evaluate the exclusion area boundary (EAB) and low-population zone (LPZ) radiation exposure. The steam releases from the ruptured and intact steam generators and the primary-to-secondary leakage into the ruptured steam generator are determined from the MARVEL-M results for the period from the initiation of the accident until the leakage flow is terminated.

Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to the reactor trip is via the condenser air ejector. After the reactor trip, the release to the atmosphere is assumed to be via the MSRVs, the MSDVs and the MSSVs in the intact steam generators, and via the MSRV and the MSSVs in the ruptured steam generator. Steam relief through the MSRVs and the MSDVs continues until the

RHR system is initiated to remove the decay heat. The integrated mass releases for the period between event initiation and leakage flow termination are shown in Table 15.6.3-3.

15.6.3.5 Radiological Consequences

The radiological consequences evaluation for this event is based on the alternative source term (AST) guidance documented in Reference 15.6-4.

The evaluation of the radiological consequences of the postulated SGTR event assumes that the reactor is operating at the maximum allowable limit for reactor coolant concentration and that leaking steam generator tubes results in a buildup of activity in the secondary system.

Following the rupture, any noble gases carried from the reactor coolant into the ruptured steam generator via the break flow are released directly to the environment. The iodine and alkali metal activity entering the secondary side is also available for release. The quantity of radioactivity released to the environment during an SGTR event depends on primary-to-secondary leakage flow, primary and secondary coolant activities, iodine spiking effect prior to accident, break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, and partitioning of iodine between the liquid and the gas phases.

15.6.3.5.1 Evaluation Model

Mathematical models used in the analysis are described in the following sections:

- The offsite and onsite doses are calculated with the RADTRAD code.
- The χ/Q values used in the analysis are described in Section 15.0.3.3.
- The total effective dose equivalent (TEDE) doses to a receptor at the EAB and outer boundary of the LPZ are analyzed using the models described in Section 15.0.3.1 and Appendix 15A.

Figure 15A-3 depicts the leakage sources to the environment modeled in the dose calculation.

For evaluating the radiological consequences due to a postulated steam generator tube rupture, the activity released from the affected steam generator (steam generator connected to the broken steam tube) is assumed to be released directly to the environment. The unaffected steam generators are assumed to continually discharge steam and entrained activity via the main steam safety and relief valves up to the time initiation of the residual heat removal (RHR) system can be accomplished.

All radioactivity is released to the environment with no consideration given to radioactive decay or cloud depletion by ground deposition during transport to the EAB and LPZ.

15.6.3.5.2 Input Parameters and Initial Conditions

The major parameters and assumptions used in the analysis are listed in Table 15.6.3-4, and Tables 15.0-10 through 15.0-14.

The concentrations of radionuclides in the primary and secondary system, prior to the transient, are determined as follows:

Reactor coolant activities are based on the Technical Specification limit of 1.0 μCi/g dose equivalent (DE) I-131 with extremely large iodine spike values that are described below.

The iodine concentrations in the reactor coolant are calculated using two separate assumptions that ensure the calculations account for conservatively large quantities of radioactive iodine: (1) assuming a pre-transient iodine spike and (2) assuming transient initiated iodine spikes. The use of these two separate cases is consistent with the guidance in Reference 15.6-4.

(1) Pre-transient Iodine Spike

A reactor transient has occurred prior to the SGTR transient and has raised the reactor coolant iodine concentration to 60 μCi/g DE I-131.

(2) Transient-Initiated Iodine Spike

The primary system transient associated with the SGTR transient causes an iodine spike in the primary system. The increase in reactor coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the reactor coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (1 μCi/g DE I-131) specified in Technical Specifications (i.e., concurrent iodine spike case). The assumed iodine spike duration is 8 hours.

The radioactivity released from the fuel is assumed to be released instantaneously and homogeneously through the reactor coolant.

The noble gas concentrations in the reactor coolant are based on the Technical Specification limit of 300 μCi/g DE Xe-133. Also, the pre-accident alkali metal concentrations in the reactor coolant are based on 1% fuel defect.

The secondary coolant iodine and alkali metal concentration is 10% of the reactor coolant concentration.

A 600 gallon per day steam generator primary-to-secondary leakage is assumed, which is the Technical Specification limit.

The chemical form of radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

These fractions apply to iodine released as a result of fuel damage and to iodine released during iodine spiking.

A part of break flow is assumed to be flashing. The flashing flow used in radiological consequence analysis is the break flow multiplied by flashing fraction. The flashing fraction is calculated based on the difference between reactor coolant and affected steam generator enthalpy. The flashing flows are based on Figure 15.6.3-8 and conservatively input as constant flows in each time step.

Similarly, the steam release flows from affected and intact steam generators are based on Figures 15.6.3-9 and 15.6.3-10. These steam release flows are inputted as constant flows each time steps, conservatively.

The only filtration system considered in the analysis which limits the consequences of the steam system piping rupture transient is the main control room (MCR) heating, ventilation, and air conditioning (HVAC) system.

The χ/Q values and breathing rates are listed in Table 15.0-13. The breathing rates are obtained from NRC Regulatory Guide 1.183 (Ref. 15.6-4).

Other assumptions relating to the transport, reduction, and release of radioactive material to the environment are those covered in Appendix F of RG 1.183 (Ref. 15.6-4).

15.6.3.5.3 Results

As shown in Table 15.6.3-5, for the case in which the iodine spike is initiated by the accident, the TEDE doses for the limiting 2 hours case are calculated to be 0.96 rem at the EAB and 0.43 rem at the LPZ outer boundary. These doses are less than 10% of the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34.

As shown in Table 15.6.3-5, for the case in which the SGTR occurs coincidently with a pre-existing iodine spike, the TEDE doses are calculated to be 3.6 rem at the EAB, and 1.5 rem at the LPZ outer boundary. These doses are less than the dose guideline of 25 rem TEDE stipulated by 10 CFR 50.34.

The doses for the MCR for the SGTR are bounded by the doses calculated for the loss-of-coolant accident (LOCA) event described in Section 15.6.5.5. Consequently, no doses are provided for the SGTR event.

15.6.3.6 Conclusions

Following the occurrence of a SGTR accident, the operators can identify and isolate the ruptured steam generator in a timely manner. It has also been shown that the RTS and the ESF, in conjunction with operator actions, can terminate the primary-to-secondary break flow and stabilize the reactor coolant system in a safe condition before steam generator overfill occurs.

The resultant doses are well within the guideline values of 10 CFR 50.34.

This event does not lead to a more serious fault condition.

Table 15.6.3-1
Time Sequence of Events for Steam Generator Tube Rupture - Radiological Dose Evaluation Input Analysis

Event	Time (sec)
SG tube rupture	0
Manual reactor trip (rod motion begins) and loss of offsite power	900
Reactor coolant pumps trip	900
Main feedwater isolation	900
Turbine trip	900
Ruptured steam generator isolated (MSIV closed)	1200
Intact SGs MSDV open (initiation of RCS cooling)	1500
Ruptured SG MSRV fails open	1500
ECCS initiated	1634
EFW pumps actuated	1774
Ruptured SG MSRV block valve closed	1826
SDV open	2717
SDV closed	2848
ECCS terminated (by operator)	2880
Primary leakage terminated	4183

Table 15.6.3-2
Time Sequence of Events for Steam Generator Tube Rupture
- Steam Generator Overfill Analysis

Event	Time (sec)
SG tube rupture	0
High-high SG water level reactor trip initiated(rod motion begins) and loss of offsite power	651
EFW initiated to be provided to ruptured SG	651
Turbine trip	651
Main feedwater isolation	657
EFW flow isolated to ruptured SG	951
Ruptured steam generator isolated (MSIV closed)	951
Intact SGs MSDV open (initiation of RCS cooling)	1251
ECCS Initiated	1693
EFW initiated to be provided to intact SGs	1833
SDV open	2391
SDV closed	2468
ECCS terminated (by operator)	2511
Primary leakage terminated	3391

Table 15.6.3-3
Steam Generator Tube Rupture - Mass Releases Results

	Total Mass Transfer (lb)	
	Start of Event to Break Flow Termination	Break Flow Termination to Initiation of RHR
Ruptured SG		
Condenser	1,280,000	0
Atmosphere	109,000	0
Total	1,390,000	0
Intact SGs		
Condenser	3,850,000	0
Atmosphere	454,000	3,090,000
Total	4,300,000	3,090,000
Break Flow	185,000	0

Table 15.6.3-4
Parameters Used in Evaluating Radiological Consequences of Steam Generator Tube Rupture (Sheet 1 of 2)

Parameter	Value
Source data	
Core thermal power level (MWt)	4540 (2% above the design core thermal power)
Accident initiated spike	Initial concentration equal to the 1.0 μCi/g DE I-131 with an assumed iodine spike that increases the rate of iodine release into the reactor coolant by a factor of 335. (See Table 15.0-11.) The duration is 8 hours.
Pre-accident spike	Reactor coolant concentration is 60 μCi/g DE I-131. (See Table 15.0-10.)
Reactor coolant noble gas and other radionuclides (both cases)	The noble gas concentrations in the reactor coolant are based on the Technical Specification limit of 300 μCi/g DE Xe-133. (See Table 15.0-12.) The alkali metal concentrations in the reactor coolant are based on 1% fuel defect. (See Table 11.1-2.)
Secondary system initial iodine and alkali concentration	10% of reactor coolant concentrations.
Reactor coolant mass (lb)	646,000
Initial steam generator mass (lb)	114,000 (each SG)
Offsite power	Lost after trip
Total steam generator tube leakage prior to accident (gpd)	600
Primary-to-secondary leakage duration (h)	14
Iodine chemical form	97% elemental, 3% organic

Table 15.6.3-4
Parameters Used in Evaluating Radiological Consequences of Steam Generator Tube Rupture (Sheet 2 of 2)

Parameter	Value
Ruptured steam generator	
Flashed break flow (lb/min)	0 - 0.0834 h : 686 0.0834 - 0.25 h : 655 0.25 – 0.334 h : 624 0.334 – 0.416 h : 343 0.416 – 0.49 h : 406 0.49 – 0.564 h : 312 0.564 – 0.637 h : 93.6 0.637 h – 14 h : 0
Steam released (lb)	0 - 0.25 h : 1,290,000 0.25 – 0.284 h : 67,500 0.284 – 0.447 h : 104,000 0.447 – 0.478 h : 16,200 0.478 – 0.508 h : 14,600 0.508 – 14 h : 0
Iodine partition coefficient	100
Particulate partition coefficient for moisture carryover in the steam generators	1000
Intact steam generators	
Total primary-to-secondary leakage rate (gpd)	600
Steam released (lb)	0 - 0.25 h : 3,860,000 0.25 – 0.417 h : 1,000,000 0.417 – 0.566 h : 279,000 0.566 – 0.715 h : 83,700 0.715 – 1.17 h : 170,000 1.17 – 8 h : 1,540,000 8 – 14 h : 1,540,000
Iodine partition coefficient	100
Particulate partition coefficient for moisture carryover in the steam generators	1000
Radiological dose parameters	
χ/Q	See Table 15.0-13.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.6.3-5
Radiological Consequences of Steam Generator Tube Rupture

Dose Location	TEDE Dose (rem)
Transient-initiated iodine spike	
EAB (0 to 2 hours)	0.96
LPZ outer boundary	0.43
Pre-transient iodine spike	
EAB (0 to 2 hours)	3.6
LPZ outer boundary	1.5

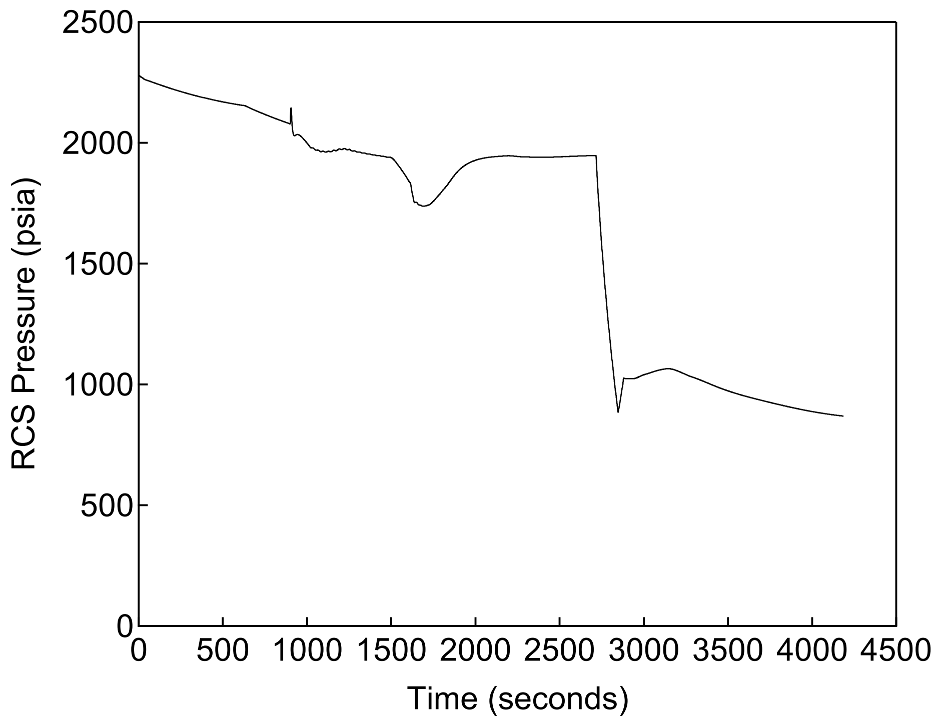


Figure 15.6.3-1 RCS Pressure versus Time

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

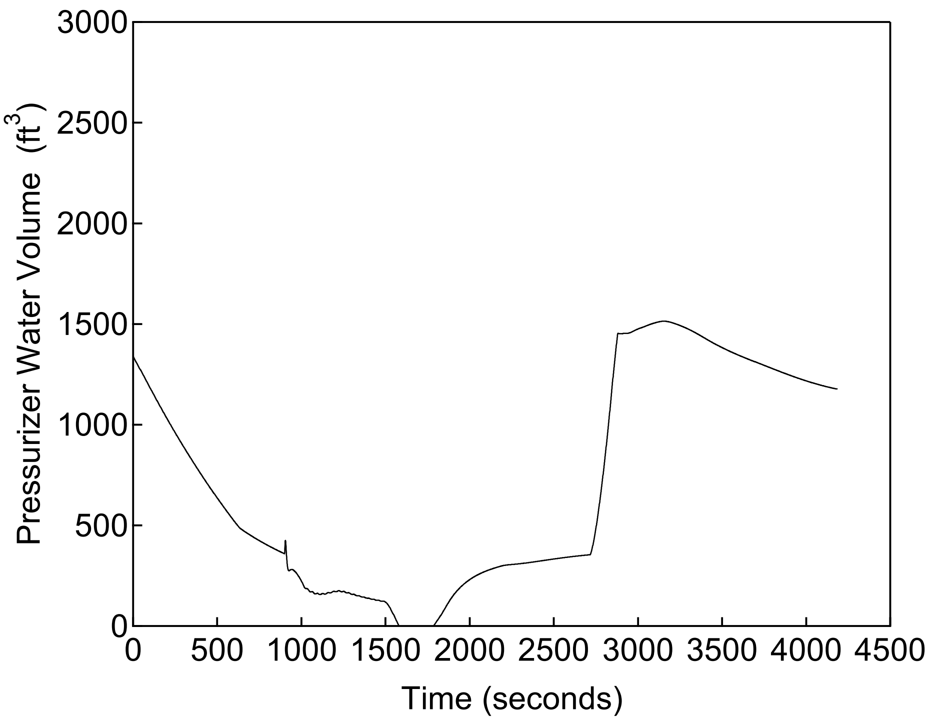


Figure 15.6.3-2 Pressurizer Water Volume versus Time

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

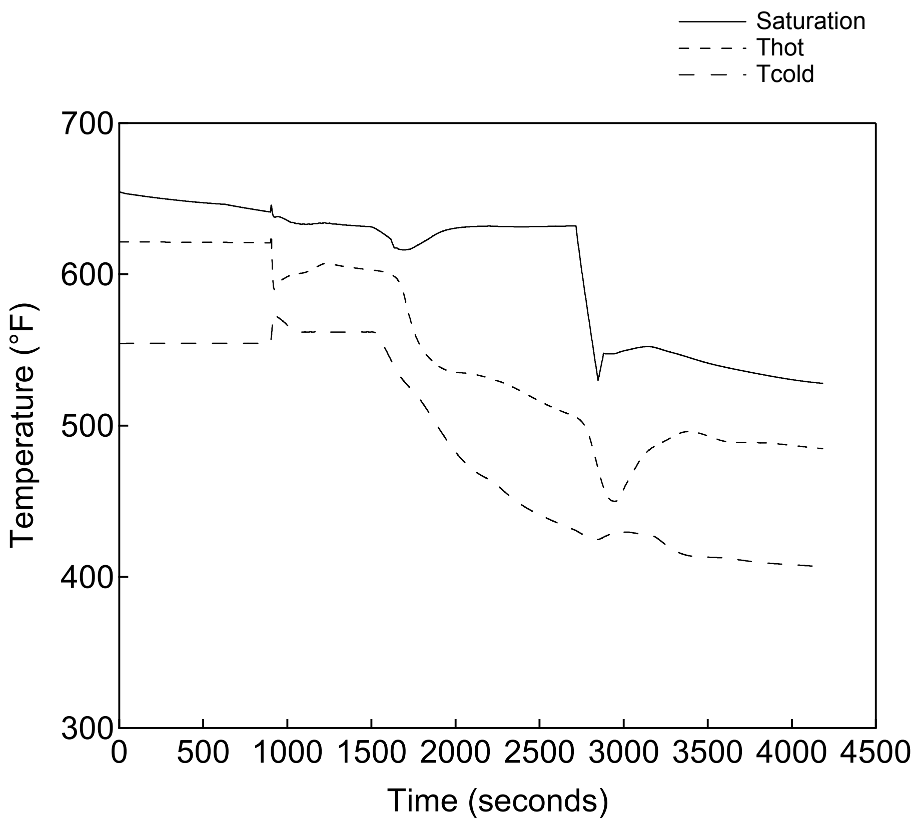


Figure 15.6.3-3 **Intact Loop Hot and Cold Leg Temperatures versus Time**

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

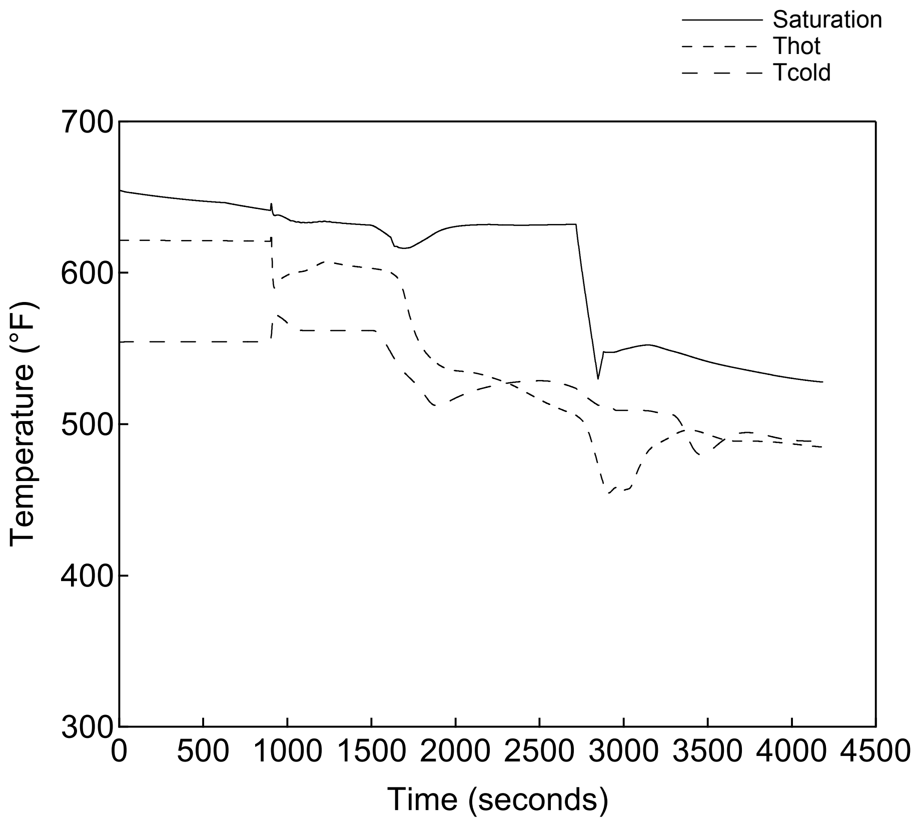


Figure 15.6.3-4 Ruptured Loop Hot and Cold Leg Temperatures versus Time

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

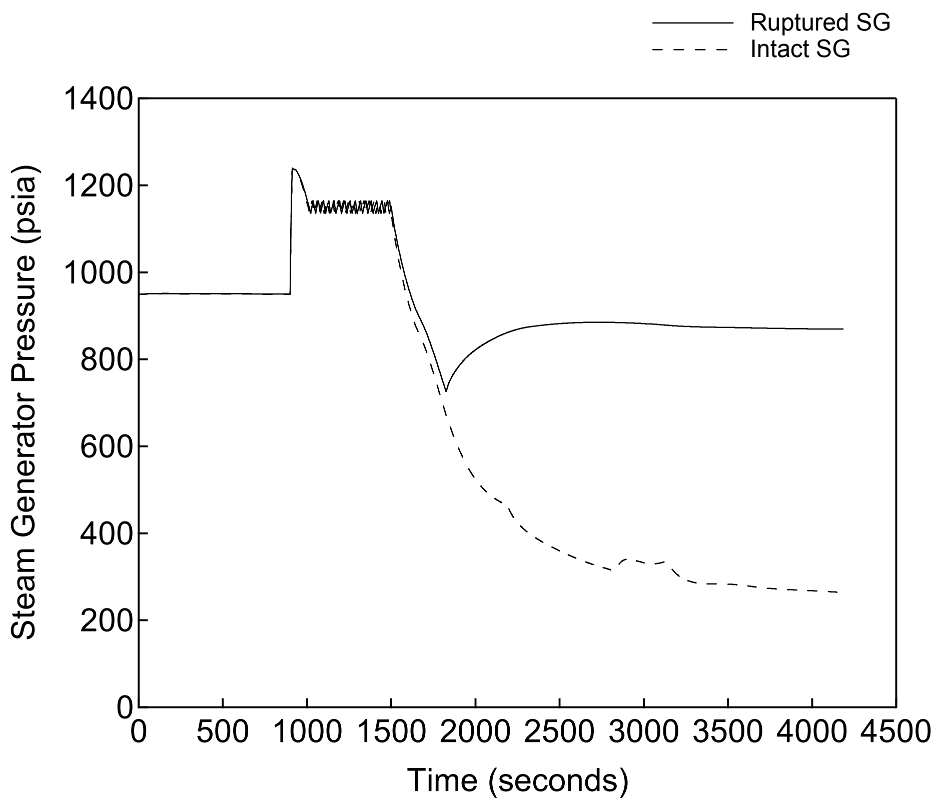


Figure 15.6.3-5 Steam Generator Pressure versus Time

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

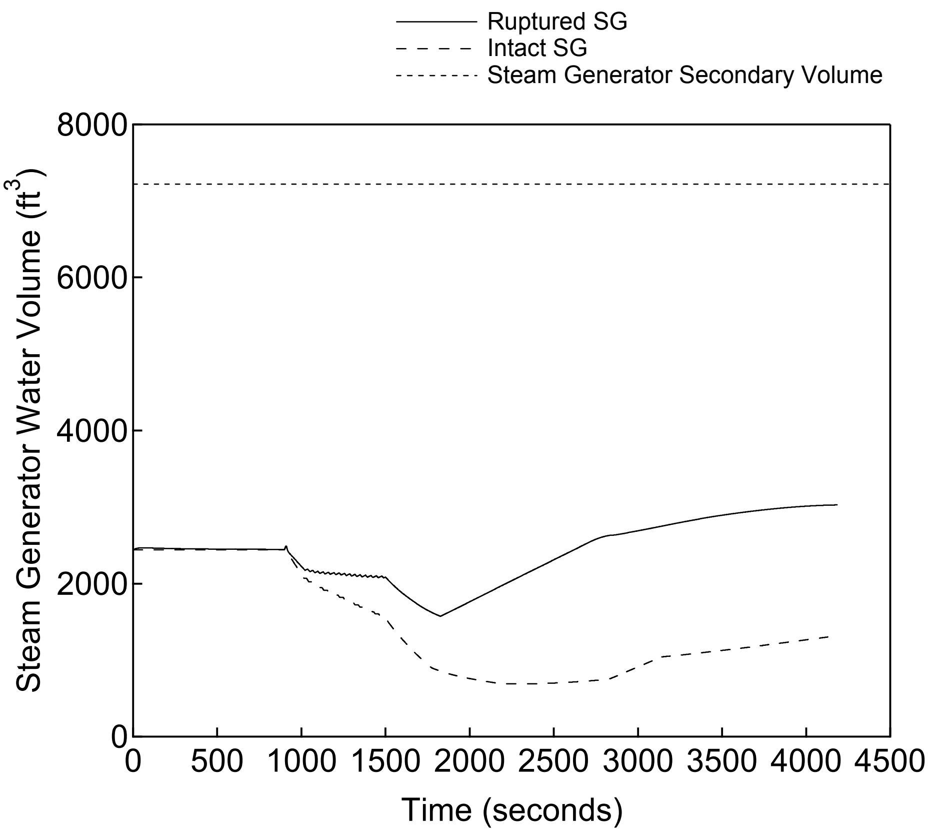


Figure 15.6.3-6 Steam Generator Water Volume versus Time

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

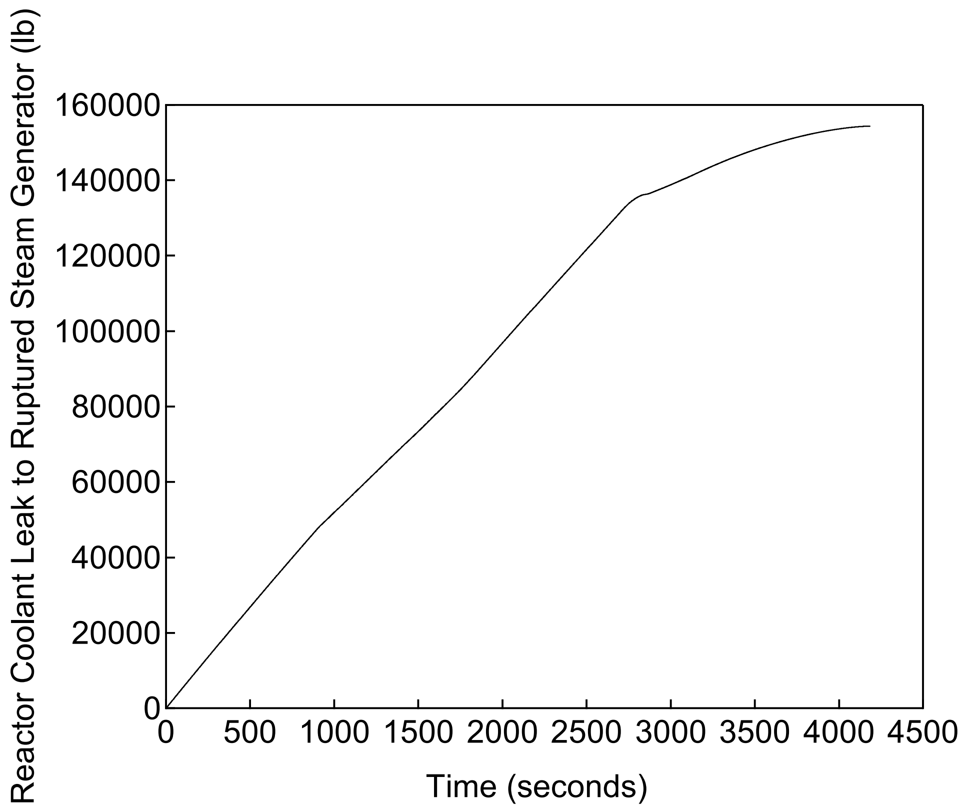


Figure 15.6.3-7 Integrated Primary-to-Secondary Break Flow versus Time

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

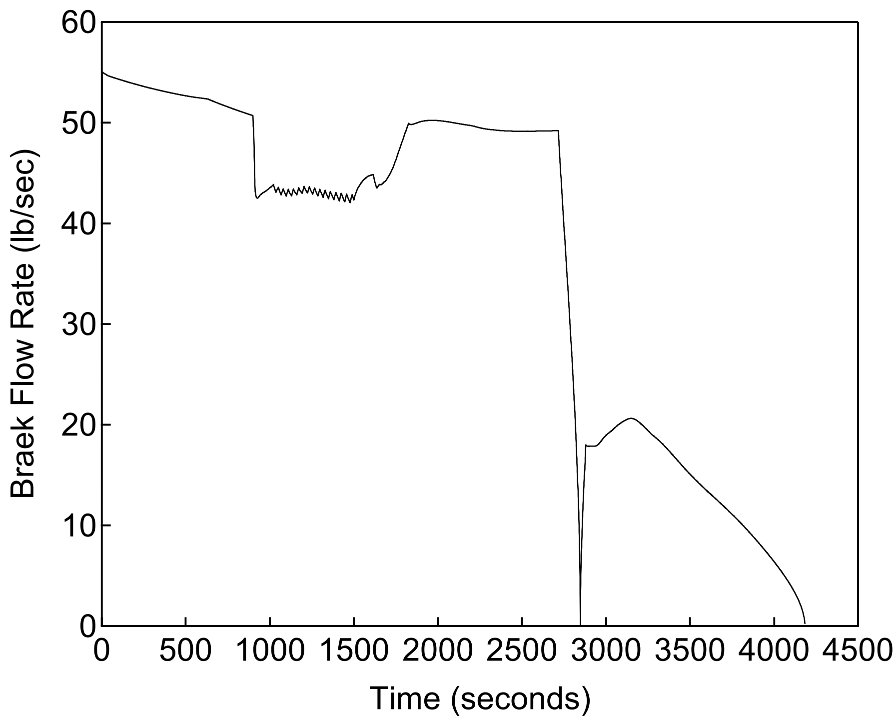


Figure 15.6.3-8 Primary-to-Secondary Break Flow Rate versus Time

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

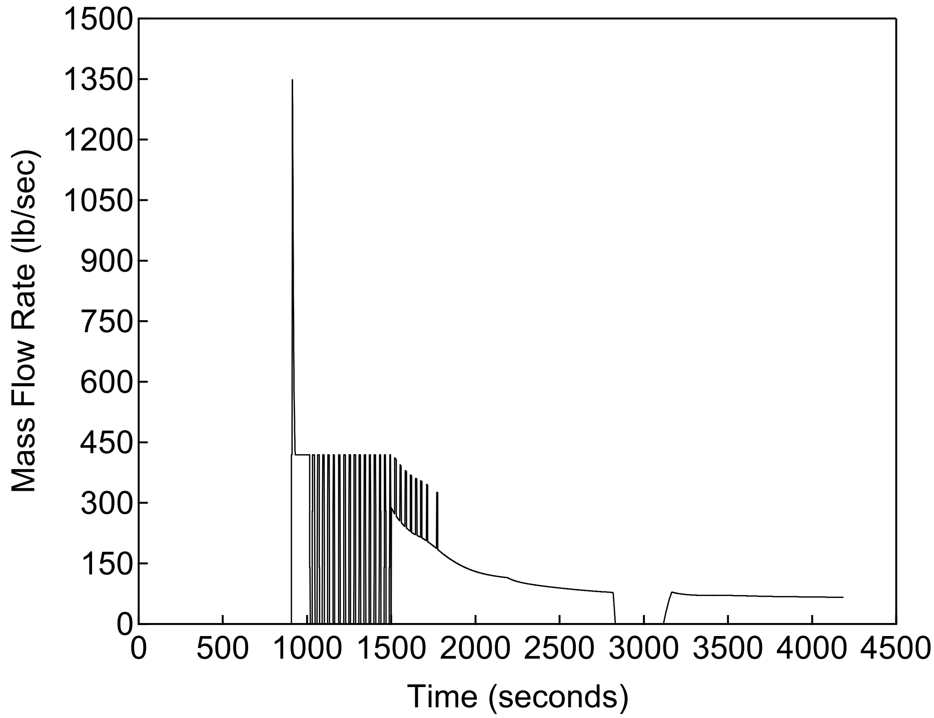


Figure 15.6.3-9 Intact Steam Generator Atmospheric Mass Release Rate versus Time

Steam Generator Tube Rupture
- Radiological Dose Evaluation Input Analysis

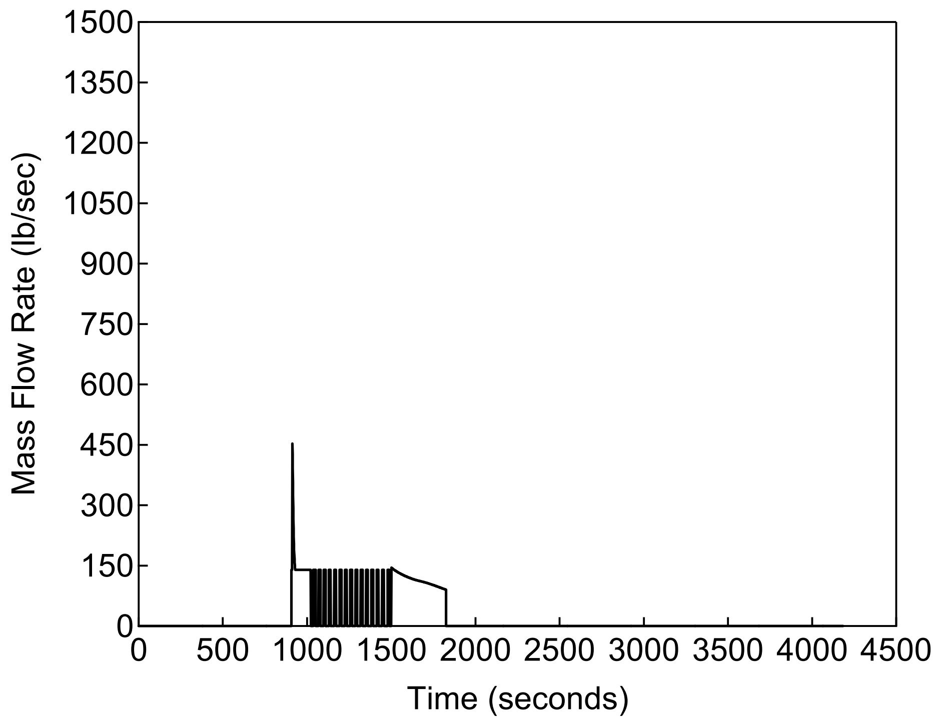


Figure 15.6.3-10 Ruptured Steam Generator Atmospheric Mass Release Rate versus Time

Steam Generator Tube Rupture
-Radiological Dose Evaluation Input Analysis

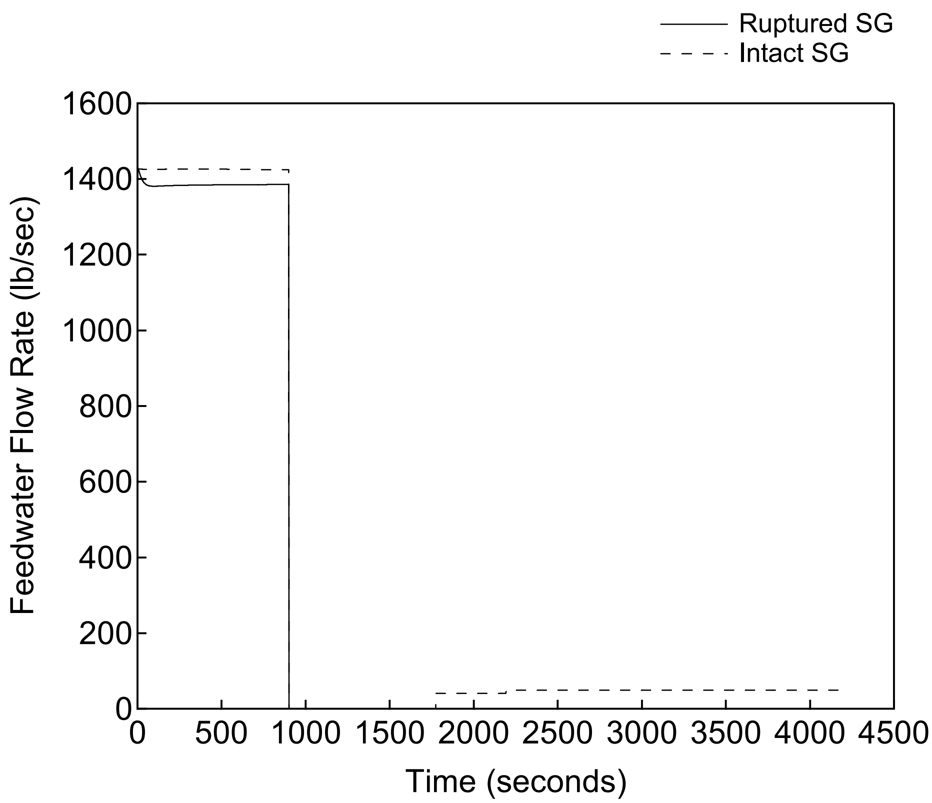


Figure 15.6.3-11 Feedwater Flow Rate versus Time

Steam Generator Tube Rupture
-Radiological Dose Evaluation Input Analysis

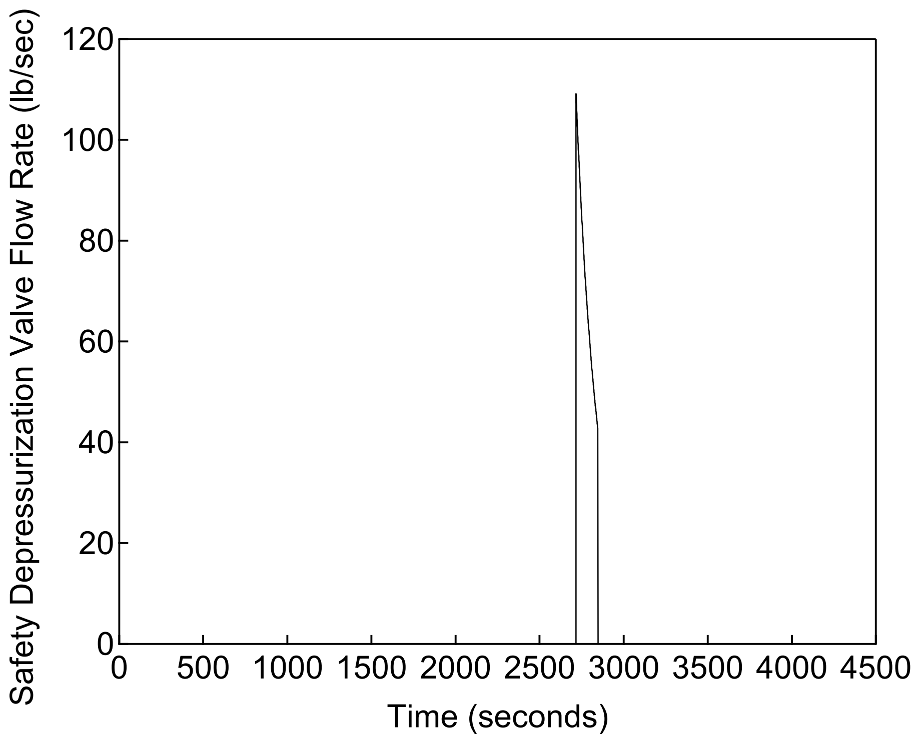


Figure 15.6.3-12 Safety Depressurization Valve Flow Rate versus Time

Steam Generator Tube Rupture
-Radiological Dose Evaluation Input Analysis

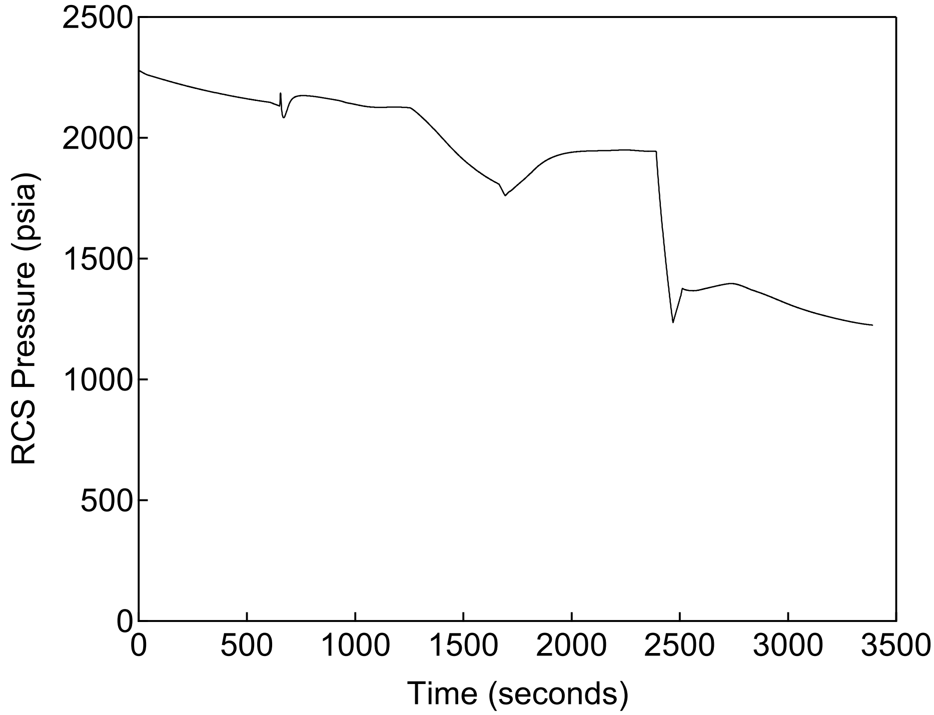


Figure 15.6.3-13 RCS Pressure versus Time

Steam Generator Tube Rupture - SG Overfill Analysis

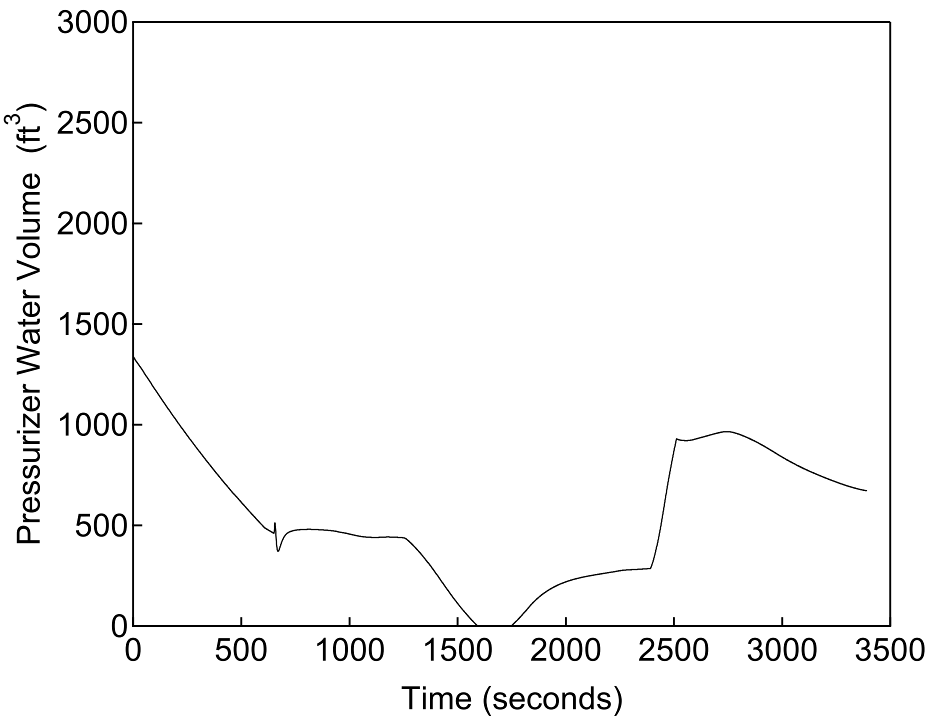


Figure 15.6.3-14 Pressurizer Water Volume versus Time

Steam Generator Tube Rupture
- SG Overfill Analysis

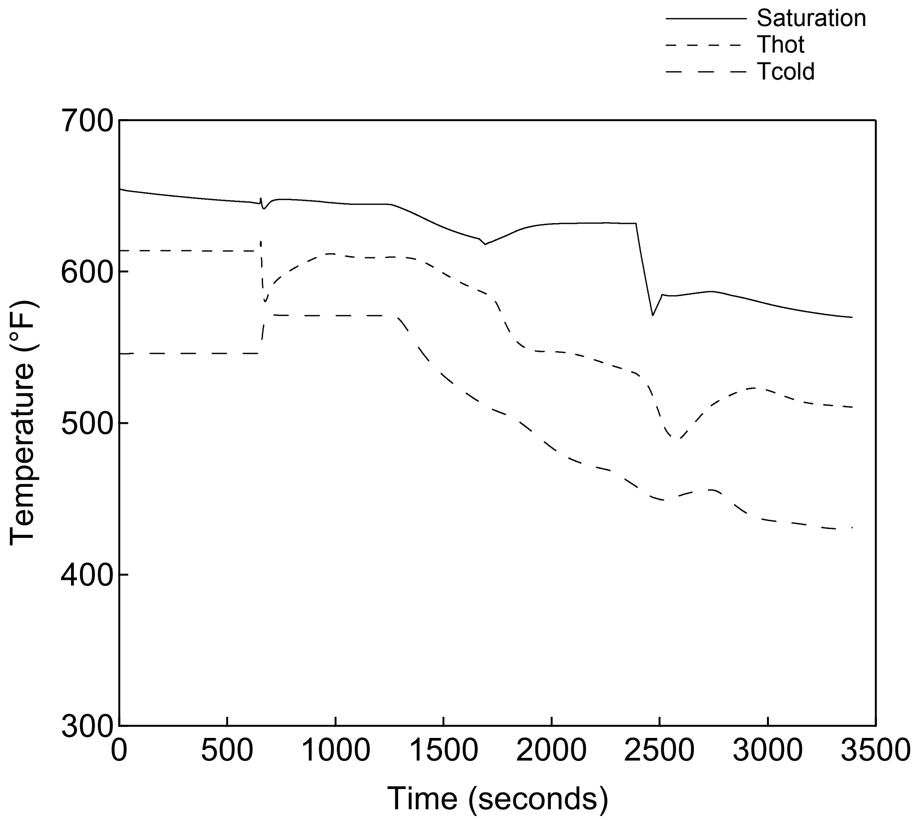


Figure 15.6.3-15 Intact Loop Hot and Cold Leg Temperatures versus Time

Steam Generator Tube Rupture
- SG Overfill Analysis

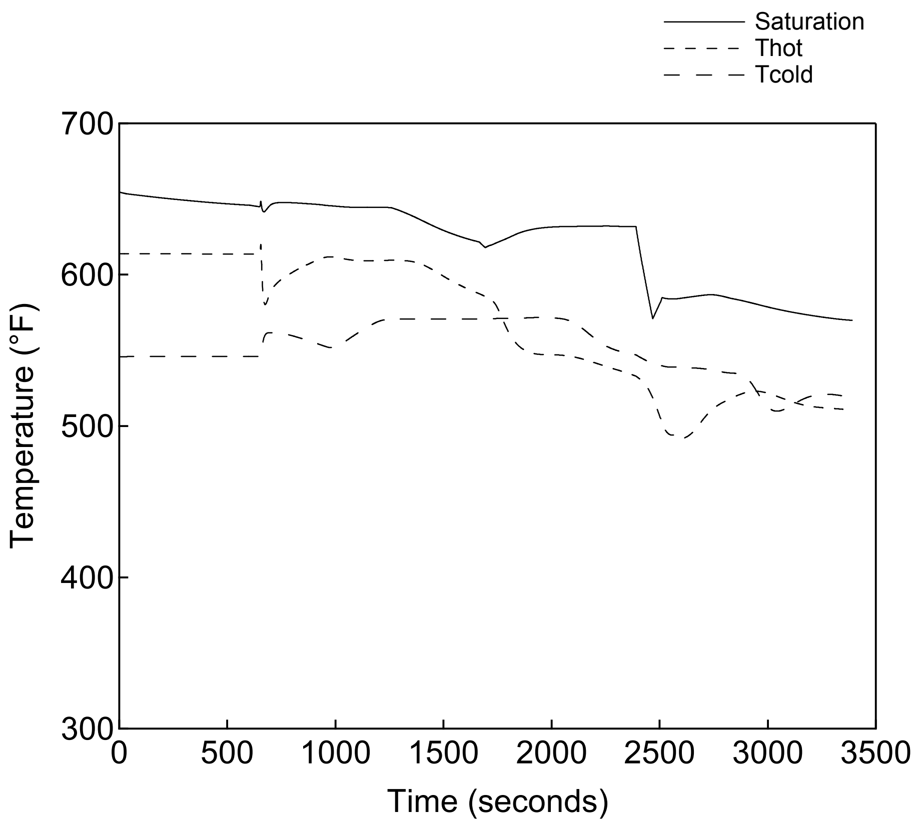


Figure 15.6.3-16 Ruptured Loop Hot and Cold Leg Temperatures versus Time

Steam Generator Tube Rupture
- SG Overfill Analysis

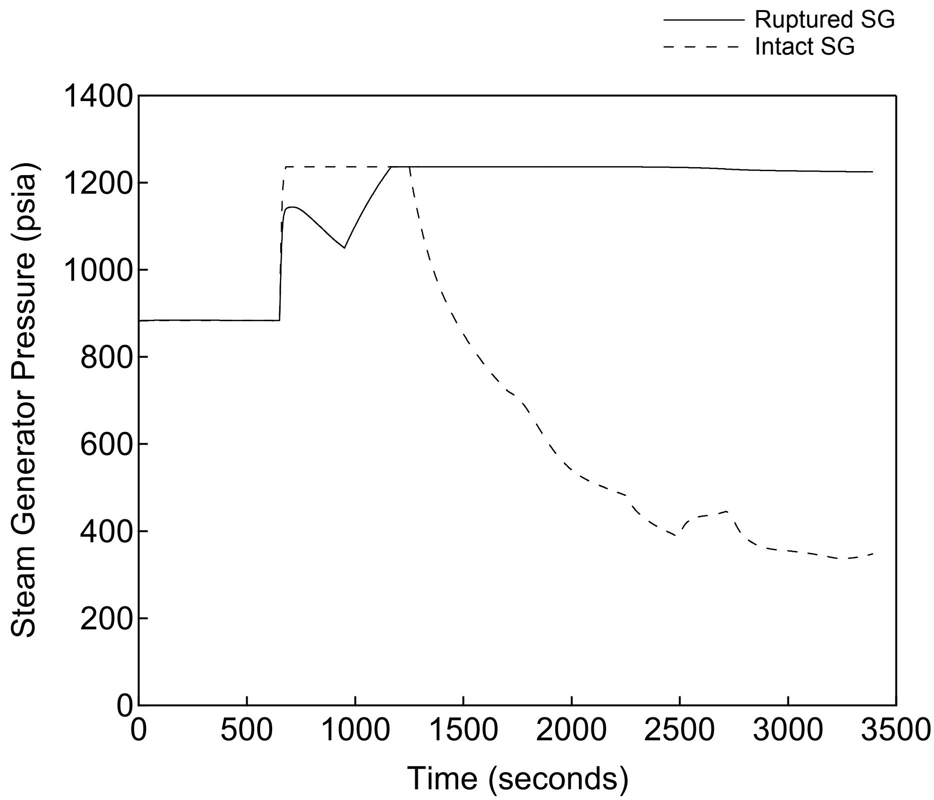


Figure 15.6.3-17 Steam Generator Pressure versus Time

Steam Generator Tube Rupture
- SG Overfill Analysis

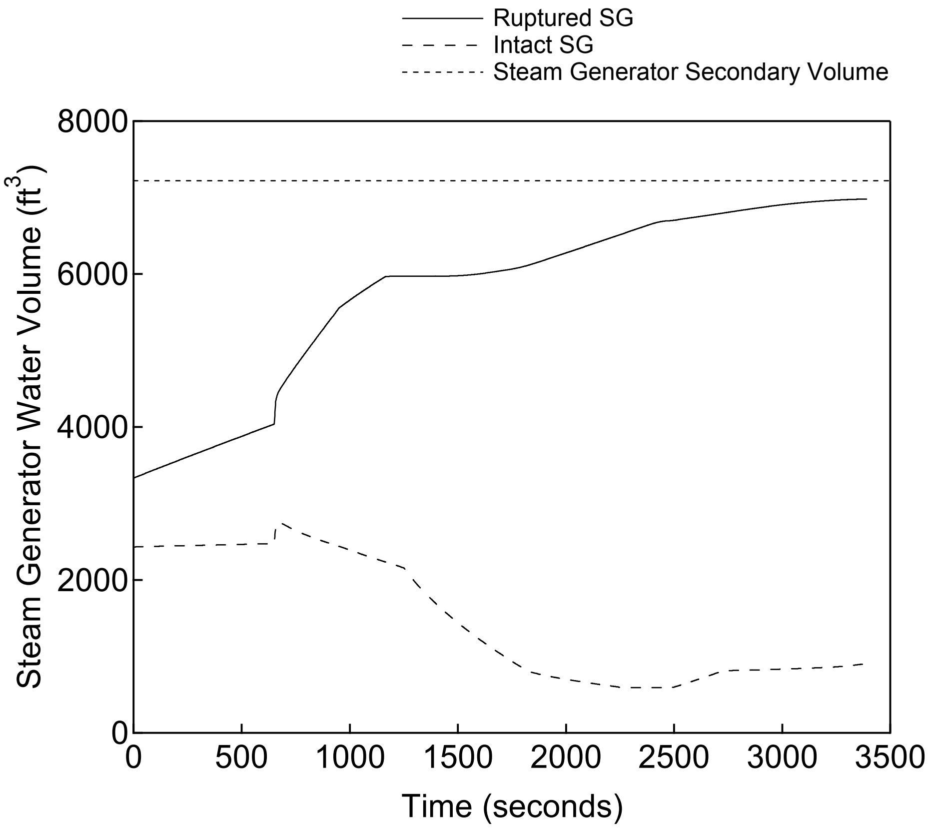


Figure 15.6.3-18 Steam Generator Water Volume versus Time

Steam Generator Tube Rupture Event
-SG Overfill Analysis

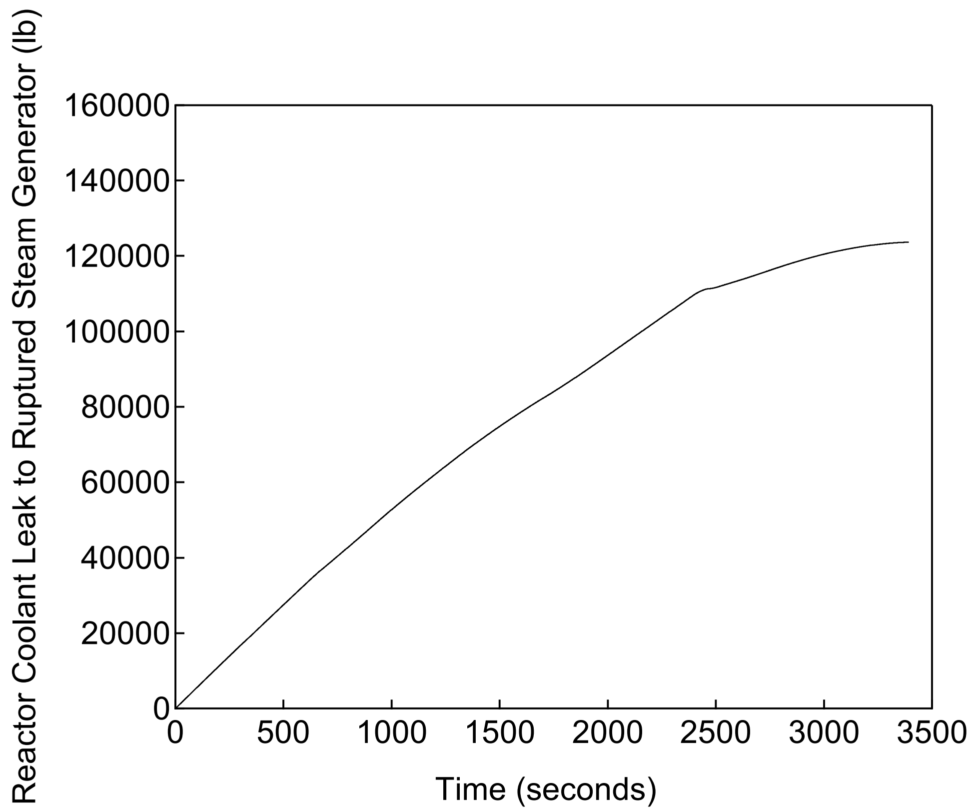


Figure 15.6.3-19 Integrated Primary-to-Secondary Break Flow versus Time

Steam Generator Tube Rupture
-SG Overfill Analysis

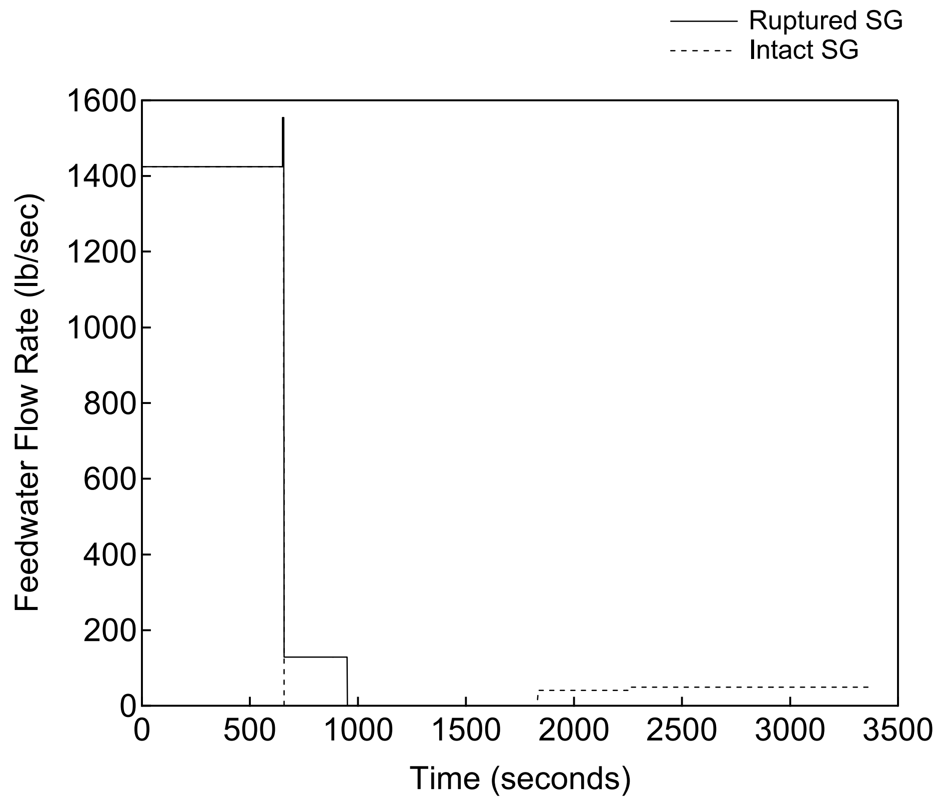


Figure 15.6.3-20 Feedwater Flow Rate versus Time

Steam Generator Tube Rupture
-SG Overfill Analysis

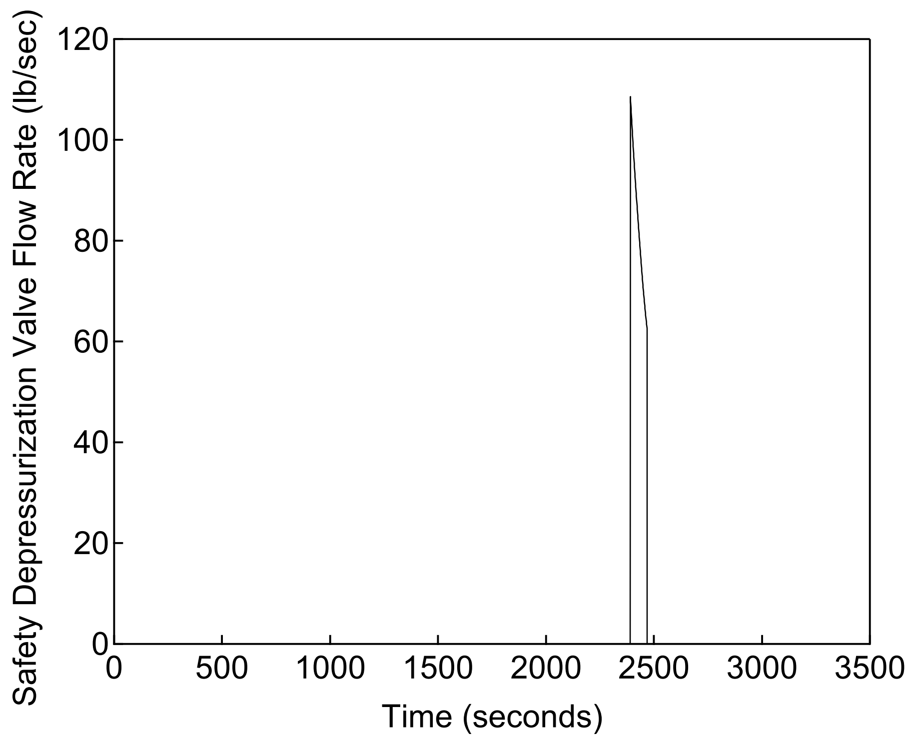


Figure 15.6.3-21 Safety Depressurization Valve Flow Rate versus Time

Steam Generator Tube Rupture
-SG Overfill Analysis

15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

This section is not applicable to the US-APWR.

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

Loss-of-coolant accidents (LOCAs) are postulated accidents (PAs) that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system. The coolant loss occurs from piping breaks in the reactor coolant pressure boundary (RCPB) up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system (RCS).

Various size breaks were examined to determine the conditions of the RCS, reactor core, and containment vessel and to demonstrate that the emergency core cooling system (ECCS) has the capability to mitigate each LOCA. For the US-APWR, the spectrum of breaks is categorized under large break and small break LOCAs, for the purposes of reporting bounding results. A large break is defined as a break with a total cross-sectional area equal to or greater than 1.0 ft^2. A small break is defined as a piping break within the RCPB with a total cross sectional area up to 1.0 ft^2.

The small break LOCA reported in this section is a large enough break that the charging pumps of the chemical and volume control system (CVCS) cannot provide sufficient makeup water to the RCS; therefore, the ECCS would be actuated. For very small breaks where the charging pumps have the capability to make up for leakage, the pressurizer level and pressure would be sustained and the ECCS would not be actuated.

In the transient and accident analyses for the US-APWR, both large break and small break LOCAs are classified as the PAs. They are not expected to occur during the life of the plant, but postulated as a conservative design basis. The event frequency conditions are described in Section 15.0.0.1.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Description of Large Break LOCA

The pipe break for the large break LOCA is assumed to occur in a cold leg piping located between the outlet of the reactor coolant pump (RCP) and the corresponding reactor vessel (RV) inlet nozzle, as this break places the most severe performance requirement on the ECCS. The double-ended cold leg guillotine (DECLG) and split breaks, with a total cross-sectional area equal to or greater than 1.0 ft^2, are analyzed. The RCS loop taken for the break is the one with pressurizer on it.

In this large break LOCA analysis, loss-of-offsite power (LOOP) is assumed. The LOOP occurs coincident with the break. The primary effect on this transient is that AC power will be lost to the RCPs and they will coastdown. The LOOP scenario is more severe as core flow decreases earlier and the safety injection (SI) pumps start later than in the offsite power available scenario. Because the LOOP cases are more severe, only those results are reported in this document.

As a result of the break, the coolant from RCS is rapidly lost, the cooling capability for the reactor core is reduced, and the RCS pressure decreases rapidly. The reactor trip and subsequent borated water injection from the accumulator complement void formation in causing a rapid reduction to a power level corresponding to fission product decay heat.

The large break LOCA is generally divided into three phases in which specific phenomena are occurring. They are the blowdown phase, refill phase, and reflood phase.

Blowdown phase

The blowdown phase covers the period from the initial pipe break to the time where the RCS pressure is equal to the containment pressure.

Initially, subcooled liquid is discharged through the break at a rate that exceeds the capacity of the RCPs in coast down mode. As a result, core flow reverses and the fuel rods go through DNB resulting in rapid cladding heat up. Reactor power decreases due to voiding in the core. Water flashes to steam starting in the upper plenum and core and continuing to the lower plenum and downcomer.

In the early stage of the blowdown, the RCPs in the intact loops are still delivering single-phase liquid to the core. As a result, there will be a temporary upward flow through the core. As the loops become two-phase, the RCP performance degrades. The cooling effect due to upward flow may not be sufficient if the break is large and the pumps performance degrades rapidly.

As the RCPs driven flow decreases, the break flow begins to dominate and core flow reverses again. Liquid, entrained liquid, and steam flows provide core cooling. As the RCS pressure continues to fall to the containment pressure, the break flow and core flow are reduced. Consequently, the core begins to heat up. When the RCS pressure drops below the accumulator injection pressure, borated water is injected into the vessel.

Refill phase

The refill phase starts from the end of blowdown phase until the lower plenum is refilled up to the bottom of the core. During this phase the core experiences a nearly adiabatic heatup as the lower plenum is filled with borated water supplied by the accumulators of the safety injection system (SIS). The accumulators operate in the high flow rate mode in this phase, which is similar to existing conventional PWR. The accumulator flow is sufficient to fill the downcomer and initiate reflood of the core.

Reflood phase

The reflood phase covers the period from the end of the refill phase to final quenching of the core. The accumulators automatically switch from the high to low injection rate as the water level in the accumulators fall. Core cooling function is maintained by the small injection flow rate and flow from the SI pumps. The injected borated water begins to quench the lower part of the core. As the quench front progresses, the location of the highest cladding temperature moves higher in the core. Eventually, the entire core is covered with a two-phase mixture and cooled.

(1) Reactor Trip Signals

A reactor trip signal occurs due to one of the following signals for this event:

- Low pressurizer pressure
- Low reactor coolant flow
- Over temperature ΔT

(2) Engineered Safety Features Actuation Signals

The engineered safety features (ESF) actuation signals are comprised of the ECCS actuation signals, the main steam line isolation signals, the containment vessel isolation signals, and the containment vessel spray actuation signals.

The ECCS actuation signal is actuated on one of the following signals for this event:

- High containment pressure
- Low pressurizer pressure

The main steam line is isolated on one of the following signals for this event:

- High main steam line pressure negative rate
- High-high containment pressure

The containment spray system (CSS) is actuated on the high-3 containment pressure signal.

The containment is isolated on one of the following signals for this event:

- ECCS actuation signal
- Containment spray actuation signal

The turbine trips automatically following the reactor trip. The RCPs trip automatically, initiated by both the ECCS actuation signal and the reactor trip with a delay time. RCP coastdown occurs in the blowdown phase.

After the reactor and turbine trips, heat from the core, hot internals, and the vessel continue to be transferred to the coolant and then to the secondary system. Since the secondary system heat sink is temporarily lost due to the turbine trip, the secondary system pressure increases. In the case of LOOP, the emergency power source (EPS) supplies electrical power to the essential components of the ECCS. Hence, the design functions are maintained.

As a result of a high containment pressure, the main steam lines are automatically isolated. After the isolation of main feedwater system, the ECCS actuation signal initiates flow to the secondary side by starting the emergency feedwater (EFW) pumps.

(3) Emergency Core Cooling System Functions During a LOCA

The US-APWR ECCS consists of the accumulator system, the high head injection system (HHIS) and emergency letdown system. The ECCS injects borated water into the RCS following a postulated LOCA to cool the reactor core, to prevent damage to the fuel cladding, and to limit the zirconium-water reaction of the fuel cladding to a very small amount.

Each of the four RCS loops has an accumulator connected to the respective cold leg. When the RCS pressure falls below the accumulator initiating pressure of 600 psia, the accumulators begin to inject borated water into the RCS cold legs. Each accumulator has an internal passive flow damper, which automatically switches the injection flowrate. When the water level is above the top of a standpipe within an accumulator, water enters the flow damper through both inlets at the top of the standpipe and at the side of the flow damper and thus the accumulator injects water with a large flowrate. When the water level drops below the top of the standpipe, the water enters the flow damper only through the side inlet and thus injects water at a lower flowrate. The accumulators are attached to the cold legs.

The HHIS consists of four independent safety trains, each containing an SI pump and the associated valves and piping. The SI pumps are aligned to take suction from the refueling water storage pit (RWSP) and deliver borated water directly to the downcomer through the direct vessel injection (DVI) nozzles located below the cold leg inlet nozzles on the RV. The RWSP is located within the lowest portion of the containment vessel and collect the water from the break and the containment sprays. The RWSP provides a continuous borated water source for the SI pumps avoiding the need to switch the pump suction from a storage water tank to the containment recirculation sump. The SI pumps start automatically upon receipt of the ECCS actuation signal.

The accumulators initially inject large flow rate, then automatically reduced to lower flow rate as the water level in the accumulators drop below the level of the internal standpipe. The reduced flow from the accumulators, together with the DVI flow from the SI pumps is sufficient to maintain the downcomer level provide flow to the core during the reflood phase. The combined performance of the accumulator system and the HHIS is sufficient to eliminate the need of low head injection pumps.

(4) Containment Spray System Functions During a LOCA

The containment spray system (CSS) consists of four independent trains, each containing a containment spray/residual heat removal (CS/RHR) heat exchanger, a CS/RHR pump, spray nozzles, piping and valves. The CSS takes borated water taken from the RWSP then sprays it into the containment vessel to maintain the pressure of the containment to be below the design pressure and restore it to approximately atmospheric pressure. The CSS is automatically actuated on the high-3 containment pressure signal. The CS/RHR heat exchangers provide long term cooling by removing heat from the containment to further reduce the pressure.

During a LOCA, the RWSP is well protected against debris wash down. Containment drains (transfer pipes) into the RWSP are protected from large debris by vertical debris bars, capped by a ceiling plate. The suction strainers, and the CSS and SI suctions are located as such that they are protected from clogging. Detailed design descriptions are given in Section 6.2.2.2.

Continued operation of the SI pumps supplies borated water during long term cooling. Core temperatures are reduced to long term, steady state levels associated with the dissipation of residual heat generation. During long term cooling, the HHIS is designed to inject into both the RCS hot legs and the reactor vessel downcomer to avoid an unacceptably high concentration of boric acid (H_3BO_3) in the core.

15.6.5.2.2 Description of Small Break LOCA

The small break LOCA is assumed primarily to occur in a cold leg piping located between the outlet of the RCP and the corresponding RV inlet nozzle, as this break places the most severe performance requirement on the ECCS. The DECLG, split and the direct vessel injection (DVI) line breaks, with a total cross sectional area up to 1.0 ft^2 are analyzed. The RCS loop taken for the DECLG and split breaks is the one with pressurizer on it.

In this small break LOCA analysis, LOOP is assumed to occur in concurrent with the reactor trip. The LOOP scenario is more severe as core flow decreases earlier and the SI pumps start later than in the offsite power available scenario. Because the LOOP cases are more severe, only those results are reported in this document.

Compared with the large break, the phases of the small break LOCA prior to recovery occur over a longer time period. In order to identify various phenomena, the small break LOCA can be divided into five phases: blowdown, natural circulation, loop seal clearance, boil-off, and core recovery. The duration of each phase depends on the break size and the performance of the ECCS. The following discussion of these five phases assumes the small break is located at the cold leg. The phases during small break LOCA can be described as follows:

<u>Blowdown phase</u>

Upon initiation of the break, the RCS primary side rapidly depressurizes until flashing of the hot coolant into steam begins. Reactor trip is initiated on the low pressurizer pressure setpoint of 1860 psia. Closure of the condenser steam dump valves isolates the SG secondary side. As a result, the SG secondary side pressure rises to the safety valve set point of 1296 psia, and steam is released through the safety valves. The ECCS actuation signal is generated at the time the pressurizer pressure decreases to the low pressurizer pressure setpoint of 1760 psia and safety injection initiates, after a time delay. Then the RCPs automatically trip, after 3 seconds delay, upon the ECCS actuation signal resulting from the low pressurizer pressure. The coolant in the RCS remains in the liquid phase throughout most of the blowdown period, although toward the end of the period, steam begins to form in the upper head, upper plenum, and hot legs. The rapid depressurization ends when the pressure falls to just above the saturation pressure of the SG secondary side, which is at the safety valve set point. The break flow in the RCS is single-phase liquid throughout the blowdown period.

Natural Circulation phase

When the blowdown phase ends, two-phase natural circulation is established in the RCS loops with the decay heat being removed by heat transfer (condensation and convection) to the SG secondary side. The EFW is initiated to maintain the secondary side inventory. As more coolant is lost from the RCS through the break, steam accumulates in the downhill side of the SG tubes and the crossover leg. The natural circulation phase will continue until there is insufficient driving head on the cold leg side of the loops, due to the accumulation of steam in loops between the top of the steam generator tubes and the loop seals.

Loop Seal Clearance phase

The third phase is the loop seal clearance period. With the loop seals present, the break remains covered with water. The RCS water inventory continues to decrease and steam volume in the RCS increases. The relative pressure in the core increases, which, together with the loss of coolant inventory through the break, causes the liquid levels in the core and the SG to continue to decrease. If, during this process, the core mixture level drops below the top of the core, the cladding will experience a dryout and the cladding temperature in the upper part of the core will begin to rise. When the liquid level of the downhill side of the SG is depressed to the elevation of the loop seals, the seals clear and steam in the RCS is vented to the cold legs. Break flow changes from a low-quality mixture to primarily steam. This relieves the back-pressure in the core and the core liquid level is restored to the cold leg elevation by flow from the downcomer.

Boil-off phase

After the loop seals clear, the RCS primary side pressure falls below that of the secondary side due to the increase of the break flow quality, resulting in a lower mass flowrate but a higher volumetric flow through the break. The vessel mixture level may decrease as a result of the core boiling in this phase, if the RCS pressure is too high for the injection system to make up for the boil-off rate. The core might uncover before the RCS depressurizes to the point where the SI pumps (and accumulator, when the RCS pressure drops to a sufficiently low value) deliver ECCS water to the RCS at a rate higher than the break flow.

Core Recovery phase

As the RCS pressure continues to fall, the combined SI and the accumulator flowrates eventually exceed the break flow. The vessel mass inventory increases and the core recovery is established. In a small break LOCA, the accumulator injection to the core begins before the reactor coolant is completely discharged into the containment vessel, and the RCS pressure is still above the containment pressure. For a small break LOCA, the PCT occurs when the core is at a relatively high pressure, and the break flow is choked. Therefore, the containment pressure in the small break LOCA does not affect the PCT.

TMI action item II.K.3.5 "Automatic RCP Trip during a LOCA" requires RCP trip following all small breaks. Restart of the RCPs is based on explicit guidance in the emergency operating procedures (EOPs) dealing with a safe restart of the pumps. In the case of LOOP, the RCPs automatically trip after 3 seconds time delay. When offsite-power is available, the RCPs automatically trip, after 3 seconds time delay, on the ECCS actuation signal resulting from the low pressurizer pressure. Hence, the requirement is met.

In the small break LOCA, the RCS pressure may not fall below the pressure that allows water injection from the accumulators. In this case, the HHIS alone provides the core cooling function. Continued operation of the SI pumps supplies borated-water during long term cooling. Core temperatures are reduced to long term, steady state levels associated with the dissipation of residual heat generation.

15.6.5.2.3 Description of Post-LOCA Long Term Cooling

There are two considerations in the post-LOCA long term cooling that must be addressed: maintaining long term decay heat removal and the potential for boric acid (H_3BO_3) precipitation. After the quenching of the core at the end of reflood phase, continued operation of the ECCS supplies borated water from the RWSP to remove decay heat and to keep the core subcritical. Borated water from the RWSP is initially injected through DVI lines (RV injection mode). If left uncontrolled, boric acid (H_3BO_3) concentration in the core may increase due to boiling and reach the precipitation concentration. Boric acid precipitation in the core could affect the core cooling. To prevent the boric acid precipitation, the operator switches over the operating DVI lines to the hot leg injection line (simultaneous RV and hot leg injection mode).

In the case of a hot leg break, almost all ECCS water injected through DVI lines passes through the core and exits from the break point. As a result, the boric acid concentration in the core does not increase. Even after the switchover, sufficient ECCS water passing through the core for decay heat removal is assured, and that simultaneously prevents any increase in boric concentration in the core.

In the case of a cold leg break, the ECCS water through DVI lines is not effective in flushing the core. As the result, boric acid concentration in the core may increase. After the switchover, almost all ECCS water injected into the hot leg passes the core. Therefore, the boric acid concentration in the core decreases.

The main objective of the post LOCA long term cooling evaluation is to determine the switchover time from RV injection mode to the simultaneous RV and hot leg injection mode to prevent the boric acid precipitation, hence the long-term cooling is assured.

15.6.5.3 Core and System Performance

15.6.5.3.1 Evaluation Model

The reactor is designed to withstand thermal effects caused by a LOCA event including the double-ended severance of the largest RCS pipe. The reactor core and internals together with the ECCS are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core is preserved following the accident. The ECCS, even when operating during the injection mode with the most severe single active failure, is designed to meet the requirements of 10 CFR 50.46. The requirements are:

a. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

b. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

d. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In this best-estimate large break LOCA analysis, the analysis method and inputs are identified and assessed to estimate the uncertainty of the calculated results. This uncertainty is accounted for, in order to obtain a high probability that the criteria (a) through (c) above are not exceeded.

15.6.5.3.1.1 Large Break LOCA Evaluation Model

Large Break LOCA Calculation Methodology

The 10 CFR 50.46 permits the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. In particular, best estimate thermal-hydraulic models may be used to predict the peak cladding temperature (PCT), local maximum cladding oxidation (LMO), and maximum core wide cladding oxidation (CWO). The regulation requires an assessment of the uncertainty of the best estimate calculations and that this uncertainty be included when comparing the results of the calculations to the acceptance criteria of 10 CFR 50.46. Further guidance for the use of best estimate codes is provided in Regulatory Guide 1.157 (Ref. 15.6-5).

The code scaling, applicability, and uncertainty (CSAU) evaluation methodology (Ref. 15.6-6) presented an approach for applying a best estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis. This methodology has been applied to three and four-loop PWR plants (Ref. 15.6-7) using a response surface technique for the uncertainty treatment.

The Automated Statistical Treatment of Uncertainty Method (ASTRUM), was developed using the WCOBRA/TRAC code (Ref. 15.6-8). This methodology uses a statistical sampling method, in which all parameters are simultaneously varied. The necessary number of cases to calculate the 95th percentile PCT, LMO, or CWO with 95% confidence is determined based on statistical theory. The ASTRUM is used for the US-APWR large break LOCA analysis.

WCOBRA/TRAC (M1.0) Evaluation Model

The WCOBRA/TRAC (M1.0) code is a modified version of WCOBRA/TRAC. The applicability of the modified code for the US-APWR large break LOCA analysis is discussed in the Topical Report (Ref. 15.6-9).

The WCOBRA/TRAC code combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift flux equations used in the loops to allow a complete and detailed simulation of a PWR. Also the WCOBRA/TRAC code has ability to represent important reactor components such as fuel rods, steam generators, RCPs and so on.

The main confirmation points of applicability to the US-APWR are as follows:

- Empirical correlations to model the advanced accumulator characteristics are included.

- Metal heat release and bypass flow within the neutron reflector is modeled as a separate channel with heat structure.

ASTRUM Analysis Process

The confirmatory calculations are performed before the uncertainty evaluation to set the limiting data for some parameters based on a plant-specific basis. These parameters are identified from the conventional PWR sensitivity studies as potential contributors to uncertainty. They are categorized into three groups: nominal without uncertainty, bounded, and nominal with uncertainty. The results of the confirmatory calculations are used to define conditions for reference transient calculation.

The reference transient calculation is performed to evaluate the typical large break LOCA characteristics. The reference transient calculation incorporates the nominal values for initial conditions, power distributions, and global and local parameters for the DECLG break. Some bounding parameters are fixed and selected to obtain a conservative estimate of PCT. One such parameter is the containment pressure, which affects the PCT and contains an uncertainty, as described in Chapter 6, Section 6.2.

Applying the Wilks' equation (Ref. 15.6-26), it needs 59 ASTRUM runs to obtain the 95th percentile for one parameter (i.e. PCT) with 95% confidence. The number of runs (N) for three parameters (i.e., PCT, LMO and CWO) with 95th percentile and 95% confidence is 124, obtained using the following equation:

$$\beta \le 1 - \sum_{k=0}^{2} {}_{N}C_{k} \alpha^{N-k} (1-\alpha)^{k}$$

where: α = 0.95 (95th percentile) β= 0.95 (95% confidence), k is the number of evaluation parameter, and N is the number of runs. The detail procedure to yield the 124 runs is described in the Topical Report (Ref.15.6-9) and Reference 15.6-15.

Applying ASTRUM to calculate the total uncertainty in the PCT and other parameters, all the uncertainty parameters are sampled simultaneously in random in the WCOBRA/TRAC runs. Local parameters are those that affect the local fuel response at the hot spot. The local uncertainty is incorporated in the HOTSPOT code (Ref.15.6-8) to evaluate the PCT.

15.6.5.3.1.2 Small Break LOCA Evaluation Model

The small break LOCA analysis is performed using the M-RELAP5 code (Ref. 15.6-13), a modified version of the RELAP5-3D, which has multi-dimensional thermal-hydraulics and kinetic modeling capability. One-dimensional modeling with M-RELAP5 is used for LOCAs with break sizes less than 1.0 ft^2.

The following modifications were made to the M-RELAP5 code to incorporate 10 CFR 50.46 and 10 CFR Part 50, Appendix K requirements that are also in accordance with the TMI Action Item II.K.3.30 and II.K.3.31.

- Addition of ANS-1971 x 1.2 fission product decay curve
- Addition of Baker-Just correlation (not steam-limited) for metal-water reaction rate calculations
- Addition of ZIRLO™ burst model
- For choked-flow calculation, the Moody model (steam quality > 0.01) and the Henry-Fauske model (steam quality < 0.01) are incorporated to model the discharge
- Return to nucleate and transition boiling heat transfer modes are prevented for the initial blowdown phase

Several M-RELAP5 modeling techniques are used to address specific US-APWR design features:

- Empirical correlations to model the advanced accumulator characteristics are included.

- Safety injection (SI) water temperature rises because the makeup water from the RWSP is recirculated. Temperature rise in the RWSP water is modeled.

A full spectrum of break sizes up to 1.0 ft^2 and various locations are analyzed (Ref. 15.6-16). The spectrum analysis is performed to find out the limiting PCT break size.

15.6.5.3.1.3 Post-LOCA Long term Cooling Evaluation Model

An analysis method with appropriate evaluation model is applied to control the boric acid precipitation and to assure post long term cooling after small and large break LOCAs. Figure 15.6.5-41 shows the evaluation models of post-LOCA long term cooling. These models are similar to the model described in References 15.6-10 through 15.6-13

<u>Fundamental Calculation Method</u>

The fundamental method of boric acid concentration evaluation during the post-LOCA long term cooling is as follows:

(1) Assumptions

- Only cold-leg break is modeled, because boric acid precipitation would not occur in the case of a hot leg break.
- Boric acid only flows in liquid phase. Vapor phase does not contain any boric acid.
- Two volumes are modeled. The first volume includes the core, lower plenum and upper plenum as boric acid condensation volume. The second volume is the RWSP volume as the main source of borated water.

  In this evaluation, the first volume is defined as the “Mixing Volume”.
- Void fraction is considered in estimating the inventory of mixing volume.
- The void fraction in the mixing volume is calculated by the modified Yeh’s correlation (Ref. 15.6-27).
- Boric acid mixes uniformly.
- Core decay heat is modeled to calculate core evaporation and void fraction.
- Two modes are simulated. The first is RV injection mode. The second is the simultaneous RV and hot leg injection mode.

(2) Initial Conditions

- Calculation is initiated at the beginning of reflood phase.
- The inventory of mixing volume contains a portion of the injected borated water from accumulators.
- The remaining portion of the accumulators inventory spills out into the RWSP.
- The volume of RWSP consists of: its original inventory, accumulators’ spillage and RCS coolant.

(3) Calculation Procedure

Boric acid concentration is calculated by the following procedure:

- RV injection mode

a. Core evaporation rate and void fraction are calculated.
b. The mixing volume makeup flow rate that compensates for the core evaporation and reduces void fraction is calculated.
c. Boric acid concentration in the mixing volume is calculated by the following equation:

$$CB_{MV} = \frac{MB_{MV} + \left(W_{makeup} \times CB_{RWSP}\right) \times dt}{MF_{MV} + \left(W_{makeup} - W_{boil}\right) \times dt}$$

where

CB_{MV}	Boric acid concentration in the mixing volume
MB_{MV}	Boric acid mass in the mixing volume
MF_{MV}	Boric acid solution mass in the mixing volume
CB_{RWSP}	Boric acid concentration in the RWSP volume
W_{makeup}	Mixing volume makeup flow rate
W_{boil}	Core evaporation rate
dt	Time step

d. Then, the boric acid concentration in RWSP volume is calculated by

$$CB_{RWSP} = \frac{MB_{RWSP} - \left(W_{makeup} \times CB_{RWSP}\right) \times dt}{MF_{RWSP} - \left(W_{makeup} - W_{boil}\right) \times dt}$$

Where

MB_{RWSP}	Boric acid mass in the RWSP
MF_{RWSP}	Boric acid solution mass in the RWSP

At a certain time, this RV injection mode is switched over into the simultaneous RV and hot leg injection mode.

- Simultaneous RV and hot leg injection mode

a. Core evaporation rate and void fraction are calculated.
b. All hot leg injection water flows into the mixing volume. Then, boric acid concentration of the mixing volume is calculated as follows:

$$CB_{MV} = \frac{MB_{MV} + \left(W_{hotleg} \times CB_{RWSP}\right) \times dt}{MF_{MV} + \left(W_{hotleg} - W_{boil}\right) \times dt}$$

where

W_{hotleg}: Hot leg injection flow rate

c. Mixing volume flushing flow rate is calculated by

$$W_{flush} = W_{hotleg} - \left\{W_{boil} + \left(dMF_{MV} / dt\right)\right\}$$

where

W_{flush} Mixing volume flushing flow rate

dMF_{MV} The increase of liquid mass caused by the reduction of void fraction in one time step.

d. Next, boric acid concentration in the RWSP is calculated by

$$CB_{RWSP} = \frac{MB_{RWSP} - \{W_{hotleg} \times CB_{RWSP} - W_{flush} \times CB_{MV}\} \times dt}{MF_{RWSP} - (W_{hotleg} - W_{flush}) \times dt}$$

Boric acid concentration in the core flushing flow is the same as that in the mixing volume.

Range of Mixing Volume

To specify the mixing volume, the following assumptions are used:

- Mixing volume consists of core, upper plenum and lower plenum.
  - All volumes of the core region are included in the mixing volume.
  - Upper plenum volume below hot leg bottom elevation is included in the mixing volume.
  - Half of the lower plenum is included in the mixing volume.
- Mixing volume does not include any volume of neutron reflector region.

Decay Heat

The decay heat of 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards: "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors", October 1971) is used in accordance with 10CFR50 Appendix K requirements.

Borated Water Source

The RWSP, accumulator, and RCS are considered as the sources of borated water. The initial boric acid concentration is assumed to be the maximum allowed for operating conditions. The water mass in the RWSP and accumulators is assumed to be the maximum allowed for operating conditions, because large quantity of borated water in the sources yield higher concentration of boric acid in the mixing volume. A minimum amount of RCS water mass is assumed because boric acid concentration in the RCS is lower than that in the RWSP and accumulators. Boric acid concentration is also considered in calculating liquid mass density.

Effects of System Pressure

The effects of system pressure are as follows.

- Higher system pressure gives a lower void fraction in the core and consequently, more water mass in the mixing volume.
- Higher system pressure increases the boiling rate because of the decrease in the

latent heat.

- SI system injection flow rate decreases with an increase in system pressure.

The first item implies that a higher system pressure reduces boric acid concentration in the mixing volume, while the second one yields a reverse effect. In the evaluation, the atmospheric pressure is assumed for the large break LOCA and a higher pressure for the small break.

Criterion of Boric Acid Precipitation

From Reference 15.6-28, the boric acid precipitation criterion is conservatively assumed to be 29.27 wt.%, which is the precipitation concentration in the atmospheric pressure. Core pressure is higher than the atmospheric pressure, due to the downcomer head and the flow-resistances around the loop. Therefore, the core boiling temperature and the boric acid solubility will be higher than the assumed values. Furthermore, no credit is taken for the RWSP pH additive that increases the boric acid solubility. Hence, this criterion is conservative.

15.6.5.3.2 Input Parameters and Initial Conditions

15.6.5.3.2.1 Large Break LOCA

Table 15.6.5-1 lists the major plant parameter inputs identified for use in the large break LOCA analysis. An initial transient run was made with mostly nominal values, or in some cases, a conservative one. Confirmatory WCOBRA/TRAC runs were performed by varying these limiting parameters over their normal operational ranges to determine the limiting value. The limiting values were used for the reference transient. The other parameters, which are not limiting parameters, are treated as randomly sampled over their operating range in the ASTRUM calculations. Table 15.6.5-1 also lists the major uncertainty parameters and ranges to perform the ASTRUM runs for large break LOCA of the US-APWR based on the operating ranges and other aspects.

- The limiting single failure in the large break LOCA analysis is assumed, which is the loss of one train of ECCS and a second train out of service for maintenance; In this case, only two SI pumps are available.
- Minimum ECCS safeguards are assumed, which results in the minimum delivered ECCS flow available to the RCS.
- Minimum containment pressure is applied for conservatism as described in Section 6.2.1.5.

15.6.5.3.2.2 Small Break LOCA

Spectrum analysis is performed to determine a limiting break size within the small break LOCA category. In addition, sensitivity analyses are reported in Reference 15.6-16, which covers the entire spectrum of break size, break orientation and break location, also noding, time-step size and input sensitivity studies. The sensitivity analyses are performed by complying with the requirements set forth in 10 CFR Appendix K to Part 50 on ECCS Evaluation Models. The objective is performed to determine the effects of various modeling assumption on the calculated PCT, LMO and CWO. Three small break LOCA cases are reported in this section. They are as follows:

- 7.5-inch upside break, which is the limiting break for PCT during the loop-seal clearance phase.
- 1-ft^2 upside break, which is the limiting break for PCT during the boil-off phase.
- 3.4-inch break, which is a DVI line break, with only 1 train of SI system is assumed to operate.

The major plant parameters inputs used in the Appendix-K based small break LOCA analysis are listed in Table 15.6.5.2. The top-skew axial power shape is chosen because it provides the distribution of power versus core height that maximizes the PCT. Figure 15.6.5-13 shows the hot rod power shape used to conduct the small break LOCA analysis. The hot rod power shape considers the axial off-set limits of the core design, and is conservative compared to the limiting large-break LOCA power shape. The beginning of life (BOL) hot assembly burnup provides the maximum (conservative) initial stored energy in the fuel for the SBLOCA event. In addition, for the hot rod, an initial highest pellet temperature is also assumed for conservatism.

In addition to the conditions in Table 15.6.5-2, the following conditions are also applicable to the SBLOCA.

- The limiting single failure in the small break LOCA analysis is assumed, which is the loss of one ECCS train, with one additional train out of service for maintenance; In this case, only two SI pumps are available.
- Minimum ECCS safeguards are assumed, which results in the minimum delivered ECCS flow available to the RCS.
- LOOP is assumed to occur simultaneously with the reactor trip, resulting in the delay of SI pumps and EFWS operations. RCP trip is assumed to occur 3 seconds after the reactor trip, as described in Section 15.0.0.7.
- Shutdown reactivities resulting from fuel temperature and void are given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors as shown in Table 15.6.5-2. Control rod insertion is considered to occur and assumed in the analysis.

15.6.5.3.2.3 Post-LOCA Long Term Cooling

The major input parameters used in the long term cooling evaluation are listed in Table 15.6.5-3. In this evaluation, atmospheric pressure is assumed as the lowest possible system pressure during a large break LOCA. The pressure of 120 psia, which corresponds to the boric acid congruent melting temperature of 339.8°F, is assumed as the highest possible system pressure during a small break LOCA. The initial boric acid concentrations in the RWSP, accumulator, and RCS are assumed to be maximum. Water inventory of RWSP and accumulator are assumed to be maximum because much mass of borated water source makes the concentration in mixing volume higher. RCS water mass is assumed to be minimum because RCS boric acid concentration is lower than RWSP and accumulator.

Safety injection temperature is assumed to be maximum to maximize the core evaporation rate. For a large break LOCA, the assumed injection temperature is the saturation temperature at atmospheric pressure. In the case of a small break LOCA, this temperature is assumed as the RWSP maximum temperature reached during a LOCA. In the post-LOCA long term cooling analysis, the limiting single failure is assumed, which is the loss of the entire train of one ECCS train, with one additional train out of service for maintenance; In this case, only two SI pumps are available.

Operator actions are credited to perform the switchover from the RV injection mode to the simultaneous RV and hot-leg injection mode. The timing of operator action is determined by the solubility limit of boric acid concentration in the core.

15.6.5.3.3 Results

15.6.5.3.3.1 Large Break LOCA Analysis Results

The Result of Reference Transient Calculation

The reference transient calculation is performed based on the confirmatory calculation results in order to obtain the conservative estimation. Figures 15.6.5-1 through 15.6.5-7 present the results of the reference case for the best estimate large break LOCA analysis. The transient is initiated from the end of a steady-state run. The sequence of events for the reference case large break LOCA is listed in Table 15.6.5-6, which shows the plant actions (e.g. trips, etc) and those phenomena observed in the calculation (e.g., end of blowdown, etc).

(1) Blowdown phase

During the first few seconds of the transient, the core water inventory decreases rapidly. During the blowdown phase, the initial stored energy is the main contributor to the temperature rise and boiling. The decay heat is a secondary contributor. The RCPs are presumed to trip concurrent with the break in the LOOP scenario. Consequently, DNB occurs and the cladding temperature rises quickly even though the core power decreases. The hot rod cladding temperature at the limiting elevation for large break LOCA is shown in Figure 15.6.5-1. At seven seconds into the transient, an ECCS actuation signal is generated due to the low pressurizer pressure. In the early blowdown phase, an upward flow takes place in the core removing the core decay heat by way of two-phase heat transfer. About 13 seconds into the transient, the accumulator begins to inject water at a high rate into the cold leg regions.

Figure 15.6.5-2 shows the hot assembly exit vapor, entrainment, and liquid flowrates transients. This figure displays the flow rates for the vapor, entrained liquid and continuous liquid at the top of the hot assembly.

The core pressure transient is illustrated in Figure 15.6.5-3. Following the break, the vessel rapidly depressurizes during the subcooled break flow. The pressure reduction rate then decreases as boiling begins in the vessel and the break flow becomes two-phase. As the RCS pressure falls and approaches the containment atmosphere pressure, the break and core flows reduce accordingly. The blowdown phase ends at 31 seconds.

(2) Refill phase

Figure 15.6.5-4 presents the transient of liquid level in the lower plenum. During the refill phase, core heat up occurs because the primary heat transfer mechanism is convection to steam. The lower plenum is filled with borated water supplied by the accumulators. At approximately 37 seconds, the lower plenum fills to the bottom of the core, which ends the refill period and begins the reflood period.

(3) Reflood phase

The reflood phase starts 37 seconds from the beginning of the break. In this phase, coolant enters the core from the bottom, and the core collapsed liquid level increases. The transient of collapsed liquid level in each of the four downcomer quadrants is presented in Figure 15.6.5-5. The collapsed liquid level in each of four core channels is shown in Figure 15.6.5-6.

The accumulator water level drops below the top of the internal standpipe at 56 seconds and switches from high to low flowrate injection. The accumulator and SI system flow rate transients are shown in Figure 15.6.5-7. PCT occurs at 59 seconds at about 10-ft elevation. Steam generation and liquid entrainment in the flooded portion of the core help to cool the upper part of the core and reduce the cladding temperature.

At 125 seconds, the SI pumps start to inject water into the vessel. By 190 seconds, the cladding temperature at the PCT location has gradually decreased to the point of minimum film boiling temperature. Then, the temperature rapidly decreases to the saturation temperature at about 190 seconds. This ends the reflood phase at about 220 seconds, in which the core is recovered by water.

<u>ASTRUM Results and Comparison with the 10 CFR 50.46 Criteria</u>

A series of WCOBRA/TRAC calculations are performed to determine the 95th percentile PCT, LMO and CWO with 95% confidence. This PCT and other parameters are accomplished by performing 124 ASTRUM runs by randomly selecting from those parameters that are allowed to vary in the reference transient (see Table 15.6.5-1). The same WCOBRA/TRAC runs are used to obtain the 95th percentile at 95% confidence for the LMO and the CWO.

Figures 15.6.5-9 through 15.6.5-12 depict the limiting case values of PCT, LMO, and CWO with 95th percentile and 95% confidence.

Figure 15.6.5-8 shows the axial power shape operating space envelope used by the ASTRUM methodology. In the figure, PBOT is the integrated power fraction in the lower 3rd of the core, while PMID means the integrated power fraction in the middle 3rd of the core.

Figure 15.6.5-9 shows the PCT scatter plot as a function of the effective break area. The effective break area is calculated by multiplying the coefficient of discharge (C_D) with the sample value of the break area, normalized to the cold leg cross sectional area. The C_D is implemented to account for the uncertainty of the break flow model. The PCT is a conservative estimate of the 95th percentile PCT with a 95% confidence level. The figure shows cases for both DECLG and split breaks. The limiting PCT transient corresponds to the DECLG breaks. The figure clarifies that the DECLG break is found to be more limiting than the limiting size split break.

Figure 15.6.5-10 shows the cladding temperature transient of limiting PCT case, which is predicted with Run 72 and is equal to 1763°F. Table 15.6.5-7 lists the sequence of events for the limiting case large break LOCA. The PCT occurs at 60 seconds during the reflood phase. After reaching the PCT, core reflooding progresses and cladding temperature decreases. The cladding at the PCT location is quenched at 180 seconds. Finally, the whole core quenching is established at 200 seconds.

Figure 15.6.5-11 shows the cladding temperature transient at the limiting elevation for the LMO limiting case. The PCT values corresponding to the CWO is plotted against time in Figure 15.6.5-12. As a conservative approach, the value of CWO is selected as the most limiting oxidation value for the rod within the hot-assembly. Table 15.6.5-8 presents the calculated 95th percentile PCT, LMO, and CWO.

Based on the above analysis, the requirements of 10 CFR 50.46 are satisfied, and summarized as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F. The 95th percentile results of 1763°F (Run 72) presented in Table 15.6.5-8 indicates that this regulatory limit is met.

2. The calculated total oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation. The 95th percentile result of 3.5% (Run 103) maximum local cladding oxidation presented in Table 15.6.5-8 indicates that this regulatory limit is met.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. The 95th percentile result of below 0.2% (Run 30) maximum core wide cladding oxidation presented in Table 15.6.5-8 indicates that this regulatory limit is met.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. The calculations of PCT, LMO and CWO above imply

that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The analyses are carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling is achieved.

Based on the analysis, the application of ASTRUM for the best-estimate analysis of the large break LOCA shows that the acceptance criteria of 10 CFR 50.46 are satisfied for the US-APWR. In addition, it is confirmed that 2 (two) safety injection trains are capable of satisfying the design cooling function for any large break LOCA, assuming a single failure of one train, and another train out of service for maintenance.

15.6.5.3.3.2 Small Break LOCA Analysis Results

Details for the limiting small break LOCA are presented in this section. The results for other cases are documented in detailed in Technical Report (Ref. 15.6-16).

Results of 7.5-inch Small Break LOCA Analysis

The sequence of events for the 7.5-inch small break LOCA is presented in Table 15.6.5-9. Depressurization of the RCS (Figure 15.6.5-14) causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated when the low pressurizer pressure setpoint of 1860 psia is reached. The reactor trips at 9.3 seconds, then the power decreases (Figure 15.6.5-15). Control rod insertion starts at 11 seconds, which is concurrent with the turbine trip and main steam isolation. Voiding in the core also causes the reactor power to decrease.

The liquid and vapor discharges out of the break are shown in Figure 15.6.5-16. During the earlier part of the transient, the effect of the break flow is not strong enough to overcome the upward flow through the core that is maintained by the coasting RCPs. The ECCS actuation signal occurs at 12 seconds when the low pressurizer pressure setpoint is reached. This is immediately followed by the RCPs trip just before 13 seconds. The main feedwater flow is isolated at 17 seconds. To limit the pressure build up in the secondary system, the main steam safety valves open at 80 seconds. The upper region of the core begins to uncover at 122 seconds. Figure 15.6.5-17 shows the accumulator and safety injection mass flowrates transient. The HHIS begins to inject borated water to the reactor core at 130 seconds. The accumulators begin injecting borated water into the cold-leg at about 300 seconds.

As a result of the loop-seal clearance, the core is recovered at 143 seconds. Figure 15.6.5-18 shows the RCS inventory transient. The downcomer liquid collapsed level and core/upper plenum liquid collapsed level are shown in Figures 15.6.5-19 and 15.6.5-20, respectively.

Figure 15.6.5-21 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and the average rod in the hot assembly that contains the hot rod. The PCT of 774°F occurs at 136 seconds. This figure demonstrates that the PCT is substantially lower than 2200°F.

Figure 15.6.5-22 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-10 presents the calculated PCT, LMO, and CWO results for the limiting 7.5-inch small break LOCA. This case is the limiting break for PCT during the loop-seal clearance phase.

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F. The PCT of 774°F presented in Table 15.6.5-10 indicates that this regulatory limit has been met.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. The result of 0.2% maximum local cladding oxidation presented in Table 15.6.5-10 indicates that this regulatory limit has been met.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. The maximum core wide cladding oxidation is lower than 0.2 % as presented in Table 15.6.5-10 in compliance with regulatory limit.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met since the PCT does not exceed 2200°F. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The analyses were carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling has been achieved.

Results of 1-ft^2 Small Break LOCA Analysis

The sequence of events for the 1-ft^2 break, which is a 13.5-inch equivalent diameter small break LOCA is presented in Table 15.6.5-11. This is the limiting break for PCT during the boil-off phase.

Figure 15.6.5-23 depicts the pressure transient in the pressurizer. Depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated at 6.9 seconds when the low pressurizer pressure setpoint is reached. LOOP is assumed at the same time with the reactor trip. The reactor power then decreases (Figure 15.6.5-24) following the reactor trip. Control rod insertion and main steam flow isolation occur at 8.7 seconds. The RCPs trip at 9.9 seconds, indicating 3 seconds delay from the reactor trip. Main feedwater flow is isolated at 15 seconds. Because secondary system pressure build up does not occur, the main steam safety valves remain closed.

The liquid and vapor discharges from the break are shown in Figure 15.6.5-25. Early in the transient, the effect of the break flow is not strong enough to overcome the upward flow through the core that is maintained by the coasting RCPs. Upward flow through the core is maintained. However, the flow rate is not sufficient to prevent partial uncovery in the core.

The ECCS actuation signal is generated when the low pressurizer pressure setpoint is reached at 8 seconds.

Figure 15.6.5-26 shows the accumulator and safety injection mass flow rates. The accumulators begin injecting borated water into the cold-leg at 90 seconds. The HHIS begins to inject borated water to the reactor core at 126 seconds. As a result of ECCS injection, the mass inventory is recovered. Figure 15.6.5-27 shows the RCS inventory transient. The downcomer liquid collapsed level and core/upper plenum liquid collapsed level transients are shown in Figures 15.6.5-28 and 15.6.5-29, respectively. Figure 15.6.5-30 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and for the average rod in the hot assembly that contains the hot rod. This figure shows that the PCT of 1317°F occurs at 170 seconds. The PCT is significantly lower than 2200°F.

Figure 15.6.5-31 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-12 presents the 1-ft^2 upside break, which is a 13.5-inch equivalent diameter small break LOCA.

1. The PCT of 1317°F presented in Table 15.6.5-12 indicates that this regulatory limit has been met.

2. The result of 0.2% maximum local cladding oxidation presented in Table 15.6.5-12 indicates that this regulatory limit has been met.

3. The maximum core wide cladding oxidation is lower than 0.2% as presented in Table 15.6.5-12, in compliance with regulatory limit.

4. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. The analyses were carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling has been achieved.

Results of the DVI-Line Small Break LOCA Analysis

The sequence of events for the DVI-line break, which is a 3.4-inch equivalent diameter small break LOCA is presented in Table 15.6.5-11. This case assumes the injection of only one SI pump.

Depressurization of the RCS (Figure 15.6.5-32) causes fluid to flow into the loops from the pressurizer resulting in a decrease in the pressurizer level. A reactor trip signal is generated when the low pressurizer pressure setpoint is reached at 26 seconds. The reactor power then decreases (Figure 15.6.5-33) following the reactor trip. Control rod insertion starts at 28 seconds, simultaneous with the turbine trip and main steam isolation. The RCP trips at 29 seconds, which is 3 seconds after the reactor trip.

The liquid and vapor discharges out of the break are shown in Figure 15.6.5-34. Downward flow does not occur in this particular case. Upward flow through the core is maintained. The core flow is sufficient to prevent any uncovery of the core.

The ECCS actuation signal is initiated when the low pressurizer pressure setpoint of 1760 psia is attained at 35 seconds. In this case, the HHIS alone provides the core cooling function. Figure 15.6.5-35 shows the accumulator and safety injection mass flow rates. Figure 15.6.5-36 shows that the RCS inventory increases. The downcomer liquid collapsed level transient and core/upper plenum liquid collapsed level transient are shown in Figures 15.6.5-37 and 15.6.5-38, respectively.

Figure 15.6.5-39 shows the PCT at all elevations for the hot rod at the maximum allowed linear heat rate and the average rod in the hot assembly that contains the hot rod. This figure shows that the PCT does not occur in the DVI-line break, indicating that the core keeps covered throughout the transient.

Figure 15.6.5-40 shows the flow rates for the vapor and continuous liquid at the top of the hot assembly.

The results show that the limits set forth in 10 CFR 50.46 are met as discussed below. Table 15.6.5-14 presents the DVI-line break, which is a 3.4-inch equivalent diameter small break LOCA.

1. For the DVI-line break, no heatup occurs. This obviously demonstrates that the regulatory limit has been met.

2. The result of 0.2% maximum local cladding oxidation presented in Table 15.6.5-14 indicates that this regulatory limit has been met.

3. The maximum core wide cladding oxidation is not observable because core uncovery does not even occur.

4. The calculations of PCT, LMO and CWO above imply that the core geometry remains amenable to cooling. Therefore, this regulatory limit is met.

5. The analyses were carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature to assure that long term cooling has been achieved.

Based on the analysis, the acceptance criteria of 10 CFR 50.46 are satisfied for the US-APWR. In addition, it is confirmed that two safety injection trains are capable of satisfying the design cooling function for any small break LOCAs, assuming a single failure of one train, and another train out of service for maintenance. Concluding the small break LOCA analysis, Table 15.6.5-15 lists the spectrum of peak cladding temperatures.

15.6.5.3.3.3 Post-LOCA Long Term Cooling Evaluation Results

Results of the Large Break LOCA

Figure 15.6.5-42 shows the calculated time-history of the core boric acid concentration and the solubility limit used for this calculation. In the figure, the solid line indicates that the boric acid concentration gradually increases as time advancing. The dotted line imposes the criterion of boric acid precipitation. This implies that the switchover to the hot leg injection mode must be performed before the precipitation limit is reached. The calculation indicates that a switchover at around four hours after the LOCA assures that the boric acid concentration remains below the solubility limit. After the switchover, the boric acid concentration decreases. In contrary, the dashed line shows that the concentration would increase beyond the precipitation limit if the switchover were not performed. Figure 15.6.5-42 also shows the dilution effect of the hot leg injection flow after the switchover.

Results of the Small Break LOCA

In the case of a small break LOCA, the SI flowrate is relatively small compared with the large break LOCA because RCS pressure remains high. The simultaneous RV and hot leg injection may affect the dilution behavior of the boric acid in the core. In the small break LOCA, two cases are considered with regard to the break area.

If the break size is small, the RCS pressure is maintained high and retained in a subcooled condition due to the SI system operation. In this case, the boiling of core may not occur and two-phase natural circulation is established. This situation prevents the boric acid build up in the core.

If the break size is relatively large, RCS depressurizes to relatively low pressure. Therefore, it is necessary to calculate the boric acid concentration in the core for the long term cooling evaluation in this case.

The congruent melting temperature of boric acid is 339.8°F, which is slightly lower than the saturation temperature at 120 psia (341.3°F). Therefore, cases at pressures higher than 120 psia need not be considered and the bounding case for boric acid precipitation is at 120 psia. Small break evaluation is the same as that used for the large break LOCA , except for the assumed system pressure.

Figure 15.6.5-43 shows the calculated time-history of the core boric acid concentration. The solid line indicates that the gradual increase of boric acid concentration is terminated by the switchover performed at four hours, before the precipitation limit is reached. Accordingly, the boric acid concentration reduces. The dashed line implies that the boric acid concentration continues to increase if the switchover were not carried out at four hours. Figure 15.6.5-43 also shows the dilution effect of the hot leg injected flow after the switchover.

Core Cooling after Switchover to Hot Leg

Evaluation is also performed to clarify the effect of early switchover from RV injection mode to the simultaneous RV and hot leg injection mode. If switchover is performed too early, then the injected water to the hot legs is circulated around the RCS loops by entrainment and there may not be sufficient water for core cooling and boron dilution in the core. Entrainment threshold calculations similar to those reported in Reference 15.6-10 demonstrates that significant hot leg entrainment will not occur after 100 minutes. Therefore, the evaluation demonstrates that both hot leg injection and DVI are sufficient to provide core-cooling flow at four hours after the LOCA.

15.6.5.4 Barrier Performance

The Barrier Performance is discussed in detail in the Chapter 6, Section 6.2 on the Containment System. In general, it discusses the evaluation of the containment vessel pressure and temperature transients that may affect the performance of the barriers, other than fuel cladding, that restrict or limit the transport of radioactive material from the fuel to the public during and after a LOCA.

15.6.5.5 Radiological Consequences

The radiological consequences evaluation for this event is based on the alternative source term (AST) guidance documented in Reference 15.6-4. The large break LOCA is the design basis case for determining radiological consequences for LOCA transients.

The release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity as fuel damage occurs due to the loss of coolant.

15.6.5.5.1 Evaluation Model

Mathematical models used in the analysis are described in the following sections:

- The offsite and onsite doses are calculated with the RADTRAD code. Direct radiation doses in the main control room (MCR) from the containment, radioactive plume and the MCR emergency filtration unit are calculated with the MicroShield code (Ref. 15.6-25). Assumed source information of source for direct radiation doses in the MCR is described in Section 6.4.2.5.

- The χ/Q values used in the analysis are described in Section 15.0.3.3.
- The total effective dose equivalent (TEDE) doses to a receptor at the exclusion area boundary (EAB) and outer boundary of the low-population zone (LPZ) are analyzed using the models described in Section 15.0.3.1 and Appendix 15A.

The potential release paths to the environment are from:

- The containment low volume purge system until the purge valves are closed
- Containment leakage from accident initiation
- ESF system leakage from accident initiation

Figure 15A-2 depicts the leakage sources to the environment modeled in the dose computation.

Additionally, radionuclide decay of the nuclides is credited prior to release to the environment. No decay is credited for activity in environment.

15.6.5.5.1.1 LOCA Consequence Model

Source Terms

All of the reactor coolant inventory are assumed to be released to the containment at the initiation of the LOCA The reactor coolant is assumed to have initial concentration levels at the Technical Specification limits of 300 μCi/g dose equivalent (DE) Xe-133 and 1.0 μCi//g DE I-131. Iodine spikes are not considered per Reference 15.6-4.

For the design-basis accident (DBA) LOCA, all fuel assemblies in the core are assumed to be affected. The release of activity from the damaged fuel takes place in two stages. First is the gap release, which is assumed to occur at 30 second after the initiation of the accident. The early in-vessel phase immediately follows the gap release phase and is the phase when the bulk of the activity releases associated with the accident occur.

The initial fission product inventory in the core is given in Table 15.0-14. The core inventory release fractions into the containment are prescribed by Reference 15.6-4 and are shown in Table 15.0-15. The release durations are also prescribed by Reference 15.6-4 and are shown in Table 15.0-16. Onset is the time following the initiation of the accident (i.e., time =0) and is immediately followed by the early in-vessel phase. The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase.

The chemical forms of the iodine released from the fuel to the containment are prescribed by Reference 15.6-4 to be:

- Cesium iodide (Particulates) 0.95

- Elemental iodine 0.0485
- Organic iodide 0.0015

With the exception of elemental iodine and organic iodide and noble gases, fission products are assumed to be in particulate form.

The pH of the RWSP water is assumed to be maintained at 7.0 or greater. By maintaining the pH above 7.0, the assumed iodine species split fractions given above remain valid. Several pH adjustment baskets containing sodium tetraborate decahydrate are placed in the containment to maintain the desired post-accident pH conditions in the RWSP water. (See Section 6.3.2.2.5.)

The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment as it is released. This distribution is adjusted for internal compartments that have limited containment spray. The release into the containment is assumed to terminate at the end of the early in-vessel phase.

Airborne Radioactivity Removal

Expected radioactivity removal mechanisms that are credited in the analysis are:

- Noble gases – radioactive decay
- Elemental iodine – radioactive decay, natural deposition, charcoal filter
- Organic iodide – radioactive decay, charcoal filter
- Particulates – radioactive decay, natural deposition, CSS, high-efficiency particulate air (HEPA) filter

Radioactive decay is credited for fission products remaining within containment. If a fission product escapes to the environment, no credit is taken for radioactive decay.

Reduction in airborne radioactivity in the containment by natural deposition and by the CSS is credited. Acceptable natural deposition and CSS models for removal of iodine and aerosols are described in References 15.6-18, 15.6-19, and 15.6-20. These natural deposition and CSS models are incorporated into the analysis code RADTRAD (Ref. 15.6-21). RADTRAD is used to calculate the removal of airborne radioactivity in the US-APWR containment.

Elemental iodine is removed by natural deposition on the containment wall and other objects in containment. However, natural deposition is conservatively credited to occur on the inside surface of containment only. A conservative natural deposition removal coefficient calculation is used and is based on NUREG-0800, SRP 6.5.2 (Ref. 15.6-18).

Removal of particulate iodine by natural deposition is determined based on the Powers model (10th percentile), as shown in NUREG/CR-6189 (Ref. 15.6-19). Also, the containment spray "washout" removal coefficient for particulate iodine is calculated using NUREG-0800, SRP 6.5.2 (Ref. 15.6-18).

The evaluation of the containment sprays should address areas within the containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour.

Decontamination Factor (DF) for the containment atmosphere achieved by the containment spray system is time dependent and is determined based on NUREG-0800, SRP 6.5.2 (Ref. 15.6-18). Credit for elemental iodine removal is assumed to continue until the DF of 200 (See Appendix 15A.1.2) is reached in the containment atmosphere.

In addition to removal of airborne radioactivity by natural deposition and by sprays, removal of airborne activity by filters is considered. Decay of fission products and in-growth of daughter products are also considered. The transport pathway models include filters, and air leakage. Doses at the EAB, LPZ, and the MCR are also calculated.

Release paths

Radioactive material can escape from the containment to the environment by three different pathways for the large break LOCA. Releases occur from the containment purge line prior to containment isolation, containment leakage, and ECCS equipment leakage outside the containment. Containment leakage releases consist of unfiltered leakage and leakage filtered by the annulus emergency exhaust system. The doses from these release paths are summed to obtain the total dose for the LOCA. The releases are assumed to be ground level releases.

It is assumed that containment purge is in operation when the LOCA occurs. Prior to containment isolation, radionuclides released into the containment from the break can escape through this pathway until the purge system isolation valves are closed. The volume of gas escaping is calculated based on a release rate of 20,700 cfm and a valve closure time of 15 seconds from accident initiation. Radionuclides release from the containment low volume purge system assumes that 100% of the radionuclide inventory in the RCS liquid is released to the containment at the initiation of the LOCA. No credit is taken for the filters in this purge line.

The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate for the first 24 hours. Per Reference 15.6-4, the leak rate may be reduced to 50% of the peak leak rate after the first 24 hours. The containment integrated leak test verifies that the leak rate is less than the allowable leakage rate specified in 10 CFR 50, Appendix J.

The annulus emergency exhaust system prevents uncontrolled radioactive release from the containment penetrations and safeguard components to the environment. This system has two annulus emergency exhaust filtration units, which maintain the penetration areas and safeguard component areas at a negative pressure, during accident conditions. The annulus emergency exhaust system automatically initiates on a

ECCS actuation signal. This system has HEPA filters and particulates are removed by the filters.

ESF systems that recirculate RWSP water outside of the containment are assumed to leak during their intended operation. These ESF systems include the containment spray system, residual heat removal system, and the safety injection pump. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The only borated water source for the US-APWR ESF recirculation systems is the RWSP, which is located within the containment. The following assumptions are used for evaluating the consequences of leakage from ESF components outside the containment.

With the exception of noble gases which are assumed to escape to the containment atmosphere, all the fission products released from the fuel to the containment (as defined in Table 15.0-15) are assumed to instantaneously and homogeneously mix in the RWSP water at the time of release from the core.

- The leakage is taken as two times (Ref.15.6-4) the sum of the simultaneous leakage from all components in the ESF recirculation systems above. The leakage is assumed to start at the earliest time the recirculation flow occurred in these systems and ended at the latest time the releases from these systems are terminated.
- With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.
- The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in the release activity by the ESF filter system is credited as this system serves those building areas where ESF equipment leakage can occur. The ESF filter system is evaluated based on the guidance of RG 1.52 (Ref. 15.6-22).

The analysis duration is 30 days for containment and ECCS leakage per Reference 15.6-4.

Dose calculation

The EAB dose is calculated for the 2-hour period over which the highest doses would be incurred by an individual located at the EAB. Because of the delays associated with the core damage for this accident, the first 2-hour of the accident are not the worst 2 hour interval for accumulating a dose.

The LPZ boundary dose is calculated for the 30-day duration of the accident.

For both the EAB and LPZ dose determinations, the calculated doses are compared to the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

The dose calculation models are provided in Section 15A.3 for the determination of doses resulting from activity which releases the environment.

15.6.5.5.1.2 Main Control Room Consequence Model

The release from the LOCA has the potential to expose personnel in the MCR. The TEDE analysis considered all sources of radiation that will cause exposure to MCR personnel. The sources include:

- Contamination of the MCR atmosphere by the intake of the radioactive material contained in the radioactive plume released from the facility,
- Contamination of the MCR atmosphere by the infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope (CRE),
- Direct radiation from the external radioactive plume released from the facility,
- Direct radiation from radioactive material in the containment,
- Direct radiation from radioactive material in the MCR emergency filtration unit.

The radioactive material releases and radiation levels used in the main control room dose analysis are based on the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.

Credit for engineered safety features that mitigate airborne radioactive material within the MCR are assumed according to the guidance given in the Reference 15.6-24. Such features included main control room isolation or pressurization, or intake or recirculation filtration.

When radioactivity enters the MCR, the MCR heating, ventilation, and air conditioning (HVAC) system switches over to the pressurization mode. The main control room HVAC system which works during normal operation is not a safety-class system but provides defense in depth.

The main control room HVAC system provides passive pressurization of the MCR from a filtered air intake to prevent in-leakage of contaminated air to the MCR during the accident. The main control room HVAC system automatically transfers to emergency operation mode (pressurization mode) on a ECCS actuation or a high radiation signal. The MCR is accessed by a vestibule entrance, which restricts the volume of any contaminated air that can enter the MCR from ingress and egress. The equivalent inflow of unfiltered air due to expected ingress/egress has been determined to be 120 cfm.

Assumed filter efficiency of MCR emergency filtration units is based on RG 1.52 (Ref. 15.6-22) and Generic Letter 99-02 (Ref.15.6-23).

Reference 15.6-4 provides guidance on calculating the consequences to the main control room receptor. The dose receptor is a hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual is assumed to be 3.5 x 10^{-4} m^3/s.

MCR doses are calculated using dose conversion factors identified in Regulatory Position 4.1 of Reference 15.6-4. The deep dose equivalent (DDE) from photons are corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression is used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room.

$$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$

The MCR dose calculation models are provided in Section 15A.4 for the determination of doses resulting from activity which enters the CRE.

15.6.5.5.1.3 Technical Support Center Consequence Model

The radioactive material releases and radiation levels used in the technical support center (TSC) dose analysis used the same source term, transport, and release assumptions used for determining the MCR TEDE values. The TSC dose calculation models are the same as the MCR dose calculation model. That is, ratio of ventilation flow rate to TSC volume is the same value as that of the MCR. Also, the efficiency of HEPA filter and charcoal absorber of the TSC are the same as those of the MCR. The distances from release points to receptors are almost the same between the TSC and the MCR. Therefore, the radiological consequences in the TSC are represented by those in the MCR.

15.6.5.5.2 Input Parameters and Initial Conditions

Major input parameters for the consequence analysis during the LOCA are summarized in Table 15.6.5-4, Table 15.0-10, and Tables 15.0-12 through 15.0-16. Also, the major input parameters for the MCR consequence analysis during the LOCA are summarized in Table 15.6.5-5, and Tables 15A-18 through 15A-23.

Other assumptions relating to the transport, reduction, and release of radioactive material to the environment are those covered in Appendix A of RG 1.183 (Ref. 15.6-4).

15.6.5.5.3 Results

The doses calculated for the EAB and the LPZ boundary are listed in Table 15.6.5-16. The TEDE doses for the limiting 2 hours are calculated to be 13 rem at the EAB and 13 rem at the LPZ outer boundary. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The doses calculated for the MCR personnel due to airborne activity entering the MCR are listed in Table 15.6.5-16. Also listed on Table 15.6.5-16 are the doses due to direct shine from the activity in the containment, from the radioactive plume and from the MCR emergency filtration unit. The total of the four dose pathways is within the dose criteria of 5 rem TEDE as defined in GDC 19.

15.6.5.6 Conclusions

The US-APWR satisfied all criteria for the postulated LOCA transient:

- The best-estimate analysis of the large break LOCA demonstrates that the acceptance criteria of 10 CFR 50.46 are satisfied.

- The conservative analysis of the small break LOCA, which is based on the Appendix K, demonstrates that the acceptance criteria of 10 CFR 50.46 are satisfied.

- The switchover to the simultaneous RV and hot leg injection mode at four hours after a LOCA prevents boric acid precipitation in the core, and the post-LOCA long term cooling is assured.

- The EAB and LPZ doses are shown to meet the 10 CFR 50.34 dose guidelines.

- The dose for the MCR personnel is shown to meet the dose criteria given in GDC 19.

- The requirements of the TMI Action Plan items are met.

Table 15.6.5-1

US-APWR Major Plant Parameter Inputs Used in the Best-Estimate Large break LOCA Analysis

	Reference Case	**ASTRUM run conditions**
Plant physical configuration		
Fraction of SG tube plugged	10% (maximum)	10% (maximum)
Hot assembly location	Under the open hole	Under the open hole
Power-related Parameters		
Core power	4451MWt (100%)	98% $\leq P_{core} \leq$ 102% of 4451 MWt
Peaking factor (F_Q)	2.6	$F_Q \leq 2.6$
Axial power distribution	Top skewed	Figure 15.6.5-8
Peripheral assembly power	0.2 (lower bound)	0.2 (lower bound)
Hot assembly burnup	Beginning of life (BOL)	Beginning of life (BOL)
Fuel assembly type	17 X 17 ZIRLO™ cladding	17 X 17 ZIRLO™ cladding
Initial RCS Fluid Condition		
RCS average temperature	583.8°F	583.8-4.0°F $\leq T_{AVG} \leq$ 583.8+4.0°F
Pressurizer pressure	2250 psia	2250-30 psia $\leq P_{RCS} \leq$ 2250+30 psia
Primary coolant flow	112,000 gpm/loop (thermal design flow)	112,000 gpm/loop (thermal design flow)
Accumulator temperature	95°F	70°F $\leq T_{ACC} \leq$ 120°F
Accumulator pressure	655 psia	600 psia $\leq P_{ACC} \leq$ 710 psia
Accumulator water volume	2152 ft^3	2126 $ft^3 \leq V_{ACC} \leq$ 2179 ft^3
Accident Boundary Condition		
Break location	Cold leg (in the loop with pressurizer)	Cold leg (in the loop with pressurizer)
Break type	Double-ended guillotine break	Split break and double-ended guillotine breaks
Discharge coefficient	1.0	0.8 – 1.4
Offsite Power	Not available	Not available
Number of SI pumps available	2	2
Safety Injection flow rate	Minimum	Minimum
Safety Injection temperature	76°F	45°F $\leq T_{SI} \leq$ 120°F
Safety Injection delay	118 sec	118 sec
Containment pressure	Bounded (minimum)	Bounded (minimum)

Table 15.6.5-2

US-APWR Major Plant Parameter Inputs Used in the Appendix-K based Small Break LOCA Analysis

Parameters	Values
Core and Fuel Rod Condition	
Core Power	102% of rated power (4540 MWt)
Peaking factor	F_Q = 2.6
Axial power shape	Top-skew (double humps), as shown in Figure 15.6.5-13.
Hot assembly burnup	Beginning of life (BOL)
Fuel assembly type	17 X 17 ZIRLO™ cladding
Plant Operating Condition	
Fraction of SG tube plugged	10% (maximum)
RCS average temperature	Nominal value + 4°F (587.8°F)
Pressurizer pressure	Nominal value + 30 psia (2280 psia)
Primary coolant flow	Thermal design flow (112,000 gpm/loop)
RV upper head temperature	Nominal (T_{cold})
Pressurizer level	Nominal
Accumulator temperature	Maximum (120°F)
Accumulator pressure	Minimum (600 psia)
Accumulator volume	Nominal (2150 ft^3)
Accident Boundary Condition	
Break location	Cold leg
Break type	Split
Break sizes	• 7.5-inch diameter break • 1.0 ft^2 break • 3.4-inch diameter DVI-line break
Offsite power	Not available
Reactor trip signal	Low pressurizer pressure
Reactor trip signal delay time	1.8 seconds
RCP trip (at LOOP)	3 seconds after reactor trip
ECCS actuation	Low pressurizer pressure
Safety injection delay	Maximum (118 seconds)
Number of available SI pumps	2 pumps for cold leg break 1 pump for DVI line break
Safety injection flow	Minimum
Safety injection water temperature	RWSP temperature rise is modeled

Table 15.6.5-3

US-APWR Major Plant Parameter Inputs Used in the Post-LOCA Long Term Cooling Analysis

Parameters	Values
System pressure	Atmospheric pressure (for large break LOCA)
	120 psia (*) (for small break LOCA)
Core Power	102% of rated power (4540 MWt)
Decay Heat	1971 ANS, infinite operation plus 20%
Boric Acid Source	
RWSP	
Boric Acid Concentration	Maximum (2.4 wt.%)
Volume	Maximum (89,000 ft^3)
Water Density	Maximum (Density at 39°F)
Accumulator	
Boric Acid Concentration	Maximum (2.4 wt.%)
Volume	Maximum
Density	Maximum (Density at 39°F)
RCS	
Boric Acid Concentration	Maximum (1.3 wt.%)
Volume	Minimum
Density	Minimum (Density at T_{hot} + 4°F)
ECC Water Temperature	Saturation temperature at atmospheric pressure (Large Break LOCA)
	RWSP maximum temperature reached during a LOCA (Small Break LOCA)
Operator Actions	Credited (**)

Notes:

(*) Corresponding to the boric acid congruent melting temperature of 339.8°F

(**) To perform the switchover from RV injection mode to the simultaneous RV and hot leg injection mode.

Table 15.6.5-4

US-APWR Major Input Parameters Used in the LOCA Consequence Analysis (Sheet 1 of 2)

Parameters	Value
Core thermal power level (MWt)	4540 (2% above the design core thermal power)
reactor coolant radionuclide inventory	
Noble gas concentration	300 μCi/g DE Xe-133
Iodine concentration	1.0 μCi/g DE I-131
Particulate concentration	Based on 1% fuel defect (See Table 11.1-2.)
reactor coolant mass (lb)	646,000
Radionuclide release from damaged core	
Core activity at start of accident	See Table 15.0-14.
Release fractions to containment	See Table 15.0-15.
Release timing and durations	See Table 15.0-16.
Iodine species distribution	
• Cesium iodide (%)	95
• Elemental (%)	4.85
• Organic (%)	0.15
Containment purge release data	
Containment purge flow rate (cfm)	20,700
Duration of purge from accident initiation (s)	15
Release characteristics	100% of reactor coolant inventory is released to the containment at the initiation of the LOCA

Table 15.6.5-4

US-APWR Major Input Parameters Used in the LOCA Consequence Analysis (Sheet 2 of 2)

Parameters	Value
Containment leakage release data	
Containment volume (ft^3)	2,800,000
Containment leak rate (%/d), 0-24 hr	0.15
Containment leak rate (%/d), > 24 hr	0.075
Leakage fraction to penetration areas (%)	50
Leakage fraction to environment (%)	50
Filter efficiency for particulates in annulus emergency exhaust system (%)	99
Penetration areas negative pressure arrival time (min)	4
Containment spray system initiation time (min)	5
Containment spray flow rate (lb/h)	2,650,000
Elemental iodine deposition removal coefficient	See Section 15A.1.2.
Removal coefficient for particulates	See Section 15A.1.2.
DF limit for elemental iodine removal	See Section 15A.1.2.
ESF system leakage release data	
Recirculation water mass (lb)	3,540,000
Recirculation water leakage rate (lb/h)	17.6
Start time of recirculation water leakage (min)	0
Flash fraction (%)	10
Accident period (d)	30
χ/Q	See Tables 15.0-13 and 15A-22.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.6.5-5

US-APWR Major Input Parameters Used in the Main Control Room Consequence Analysis for the LOCA

Parameters	Value
CRE volume (including MCR) (ft^3)	140,000
Occupancy frequency 0 to 24 hrs 24 hrs to 96 hrs 96 hrs to 720 hrs	 1.0 0.6 0.4
Unfiltered inleakage via ingress/egress (cfm)	120
Main control room HVAC system Time delay to switch from normal operation to emergency CRE air filtration mode (s) Unfiltered air intake flow during normal operation (cfm) Filtered air intake flow (cfm) Filtered air recirculation flow (cfm) Filter efficiency ● Elemental iodine (%) ● Organic iodine (%) ● Particulates (%)	 180 1800 1,200 2,400 95 95 99

Table 15.6.5-6

Sequence of Events for Reference Case Large Break LOCA

Events	Time (sec)
Break occurs, coincident with LOOP	0.0
RCP trip	0.0
Reactor trip due to low pressurizer pressure setpoint	0.0
ECCS actuation signal	7.0
Accumulator high flow rate injection begins	13
End of blowdown	31
End of refill	37
Accumulator low flow rate injection begins	56
PCT occurs	59
High Head Injection System begins	125
PCT Elevation quenched	190
End of transient, core covered	220

Table 15.6.5-7

Sequence of Events for Limiting Case Large Break LOCA (95th Percentile PCT with 95% Confidence)

Events	Time (sec)
Break occurs coincident with LOOP	0.0
RCP trip	0.0
Low pressurizer water level reached for reactor trip	0.0
ECCS actuation signal	7.0
Accumulator high flow rate injection begins	13
End of blowdown	32
End of refill	37
Accumulator low flow rate injection begins	57
PCT occurs	60
High Head Injection System begins	125
PCT Elevation quenched	180
End of transient, core covered	200

Table 15.6.5-8

Best Estimate Large Break LOCA Core Performance Results
(95th Percentile with 95% Confidence)

Parameters	Values	Criteria
Peak Cladding Temperature (°F)	1763 (Run 72)	< 2200
Local maximum cladding oxidation (%)	3.5 (Run 103)	< 17.0
Core wide maximum cladding oxidation (%)	0.2 (Run 30)	< 1.0

Table 15.6.5-9

Sequence of Events for 7.5-inch Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip (LOOP is assumed)	9.3
Control rod insertion starts	11.1
Main steam isolation	11.1
ECCS actuation signal	11.9
RCP trip	12.3
Main feedwater isolation	17.3
Main steam safety valve open	80
Emergency Power Source initiates	115
Core upper region uncovery	122
High Head Injection System begins	130
Peak Cladding Temperature occurs	136
Core upper region recovery	143
Emergency feedwater flow begins	145
Accumulator injection begins	298

Table 15.6.5-10

Core Performance Results for 7.5-inch Small Break LOCA

	Values
Peak Cladding Temperature (°F)	774
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	less than 0.2

Table 15.6.5-11

Sequence of Events for 1-ft^2 Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip (LOOP is assumed)	6.9
ECCS actuation signal	8.3
Control rod insertion starts	8.7
Main steam isolation	8.7
RCP trip	9.9
Main feedwater isolation	14.9
Main steam safety valve open	not actuated
Accumulator injection begins	90
Core upper region uncovery	96
Emergency Power Source initiates	111
High Head Injection System begins	126
Emergency feedwater flow begins	141
Peak Cladding Temperature occurs	170
Core upper region recovery	359

Table 15.6.5-12

Core Performance Results for 1-ft^2 Small Break LOCA

Items	Values
Peak Cladding Temperature (°F)	1317
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	less than 0.2

Table 15.6.5-13

Sequence of Events for DVI-line Small Break LOCA

Events	Time (sec)
Break occurs; blowdown initiation	0.0
Reactor trip, (LOOP is assumed)	25.9
Control rod insertion starts	27.7
Main steam isolation	27.7
RCP trip	28.9
Main feedwater isolation	33.9
ECCS actuation signal	35.4
Main steam safety valve open	57
Emergency Power Source initiates	138
High Head Injection System begins	153
Emergency feedwater flow begins	168
Core upper region uncovery	not occur
Peak Cladding Temperature	lower than the initial value
Core upper region recovery	N/A

Table 15.6.5-14

Core Performance Results for DVI-line Small Break LOCA

Items	Values
Peak Cladding Temperature (°F)	lower than the initial value
Maximum local cladding oxidation (%)	0.2
Maximum core wide cladding oxidation (%)	N/A

Table 15.6.5-15

Spectrum of Peak Cladding Temperatures for Small Break LOCA

Break size and orientation	PCT
1-ft^2 at cold leg (bottom)	1174°F
13-inch at cold leg (bottom)	1154°F
12-inch at cold leg (bottom)	938°F
11-inch at cold leg (bottom)	lower than the initial temperature
10-inch at cold leg (bottom)	lower than the initial temperature
9-inch at cold leg (bottom)	lower than the initial temperature
8-inch at cold leg (bottom)	lower than the initial temperature
7.5-inch at cold leg (bottom)	761°F
7-inch at cold leg (bottom)	756°F
6.5-inch at cold leg (bottom)	lower than the initial temperature
6-inch at cold leg (bottom)	lower than the initial temperature
5-inch at cold leg (bottom)	lower than the initial temperature
4-inch at cold leg (bottom)	lower than the initial temperature
3-inch at cold leg (bottom)	lower than the initial temperature
2-inch at cold leg (bottom)	lower than the initial temperature
1-inch at cold leg (bottom)	lower than the initial temperature

Table 15.6.5-16

Radiological Consequences of the LOCA

Dose Location	TEDE Dose (rem)
EAB (0.5 to 2.5 hours)	13
LPZ outer boundary	13
MCR dose	
Airborne activity entering the MCR	4.4
Direct radiation from the containment	8.2×10^{-3}
Direct radiation from the radioactive plume	2.1×10^{-4}
Direct radiation from the recirculation filters	9.2×10^{-3}
Total	4.5

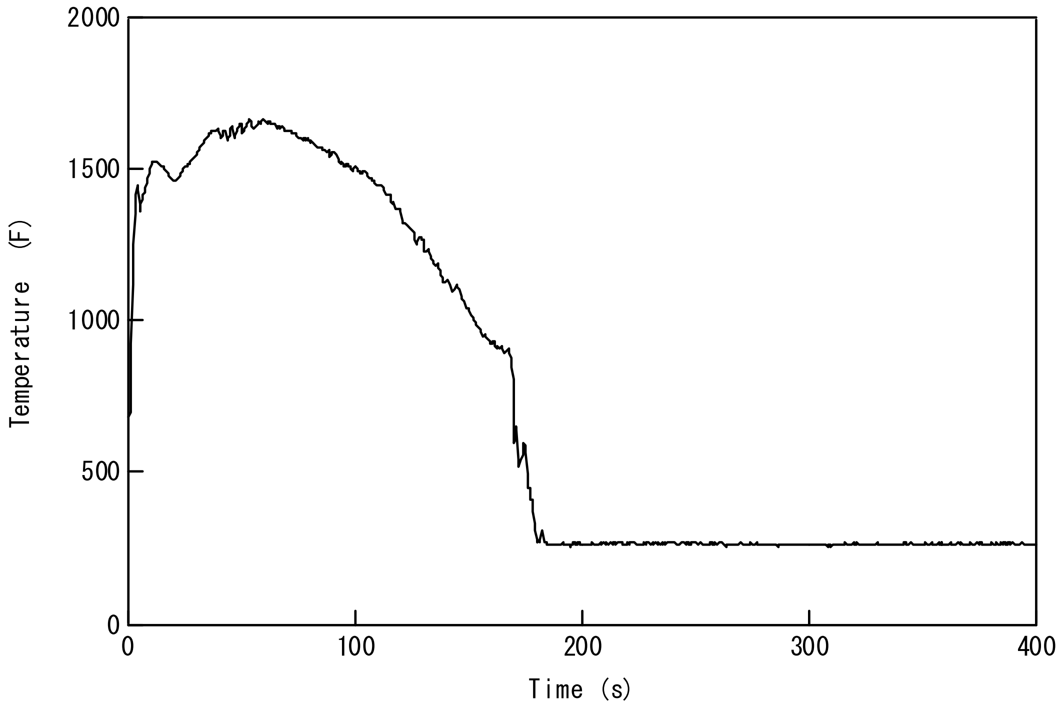


Figure 15.6.5-1 **Hot Rod Cladding Temperature at the Limiting Elevation (10 ft) for Large Break LOCA (Reference Case)**

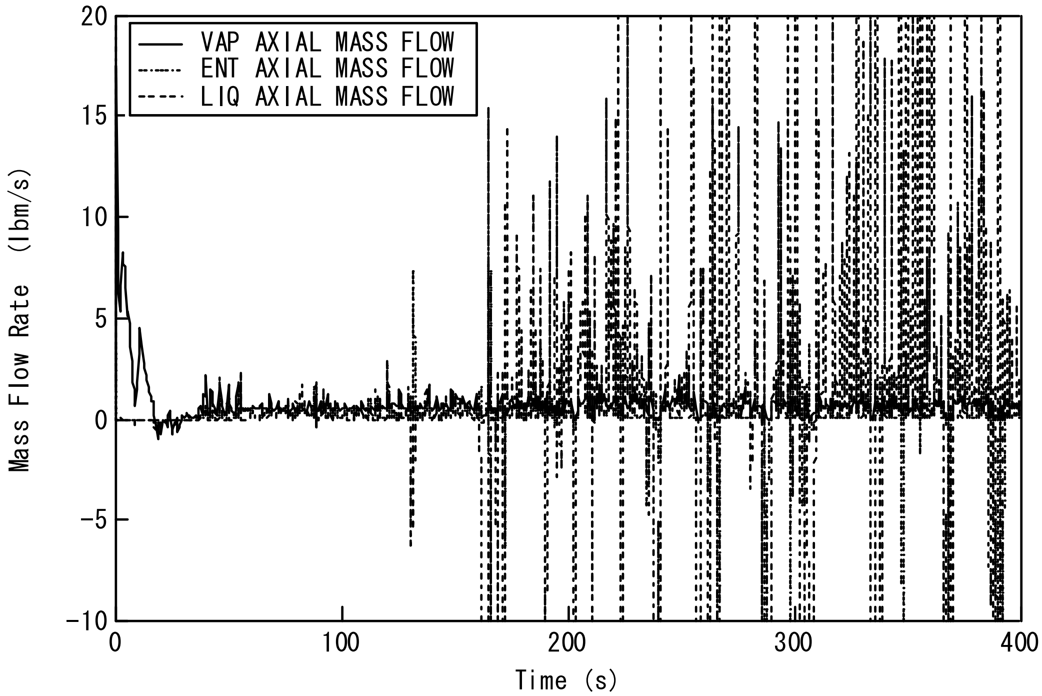


Figure 15.6.5-2 Hot Assembly Exit Vapor, Entrainment, Liquid Flow Rates for Large Break LOCA (Reference Case)

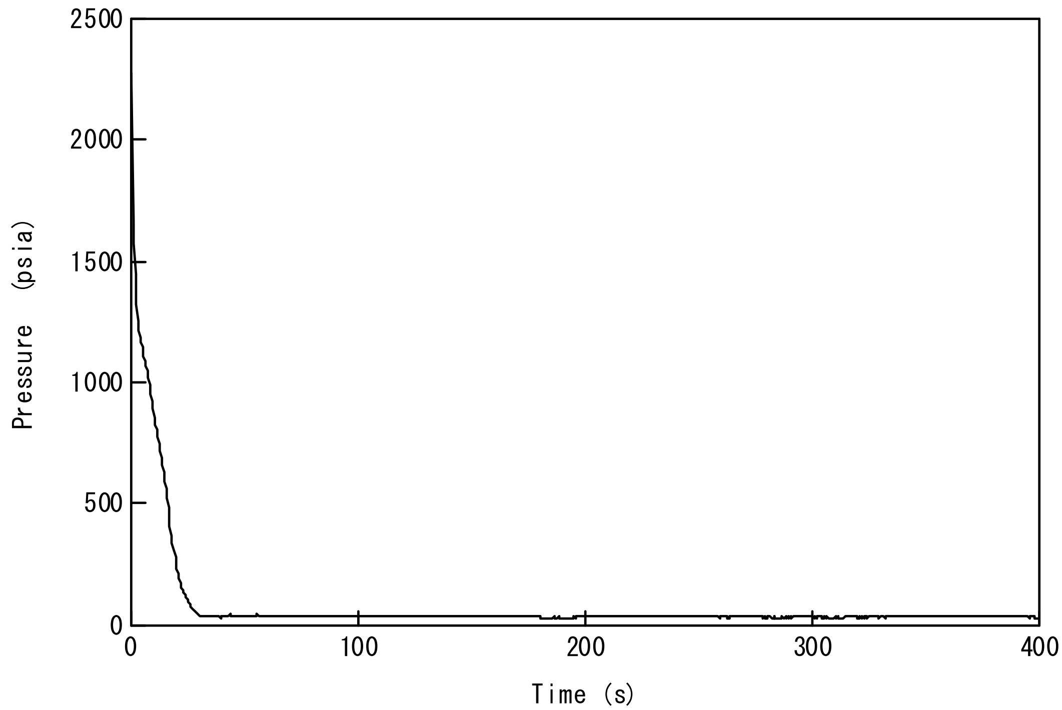


Figure 15.6.5-3 Core Pressure Transient for Large Break LOCA (Reference Case)

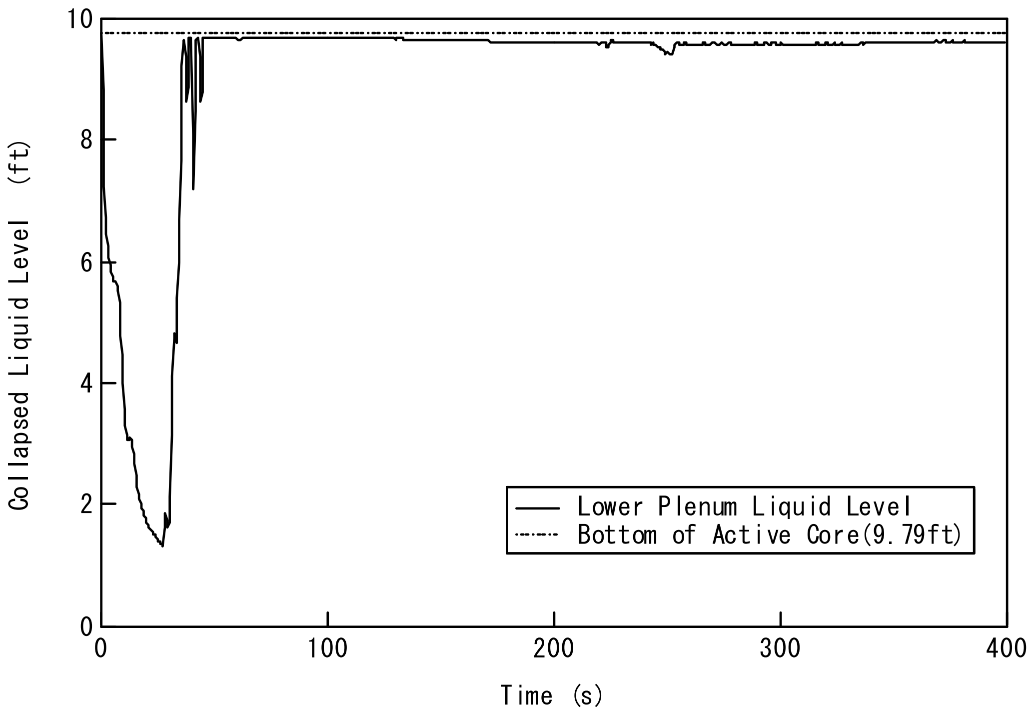


Figure 15.6.5-4 Lower Plenum Liquid Level for Large Break LOCA (Reference Case)

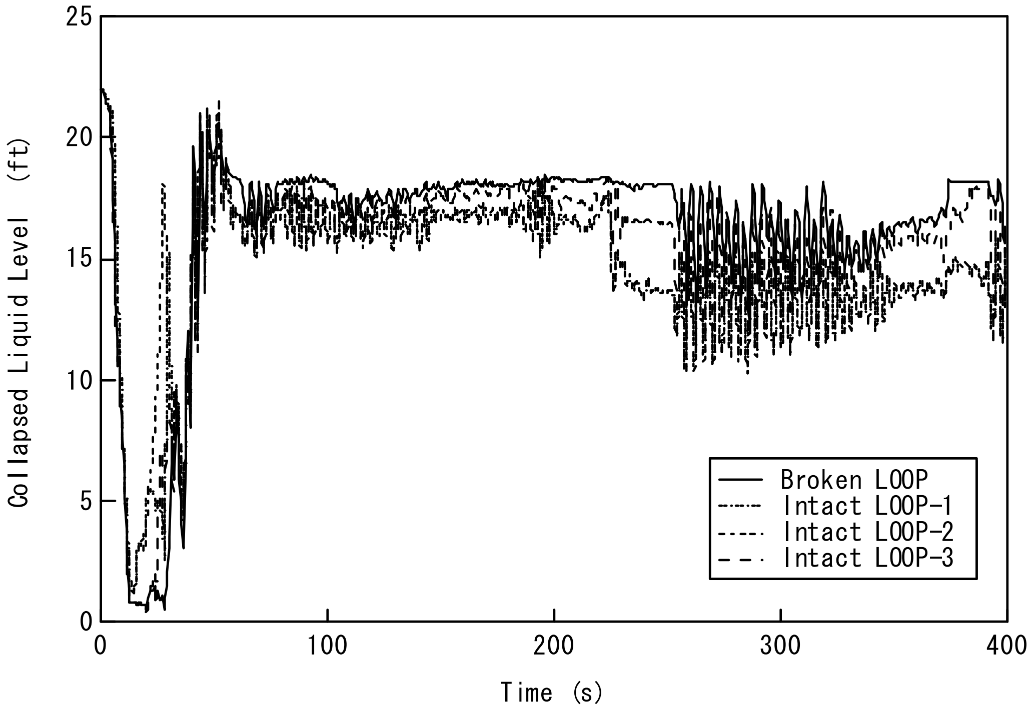


Figure 15.6.5-5 **Downcomer Liquid Level for Large Break LOCA (Reference Case)**

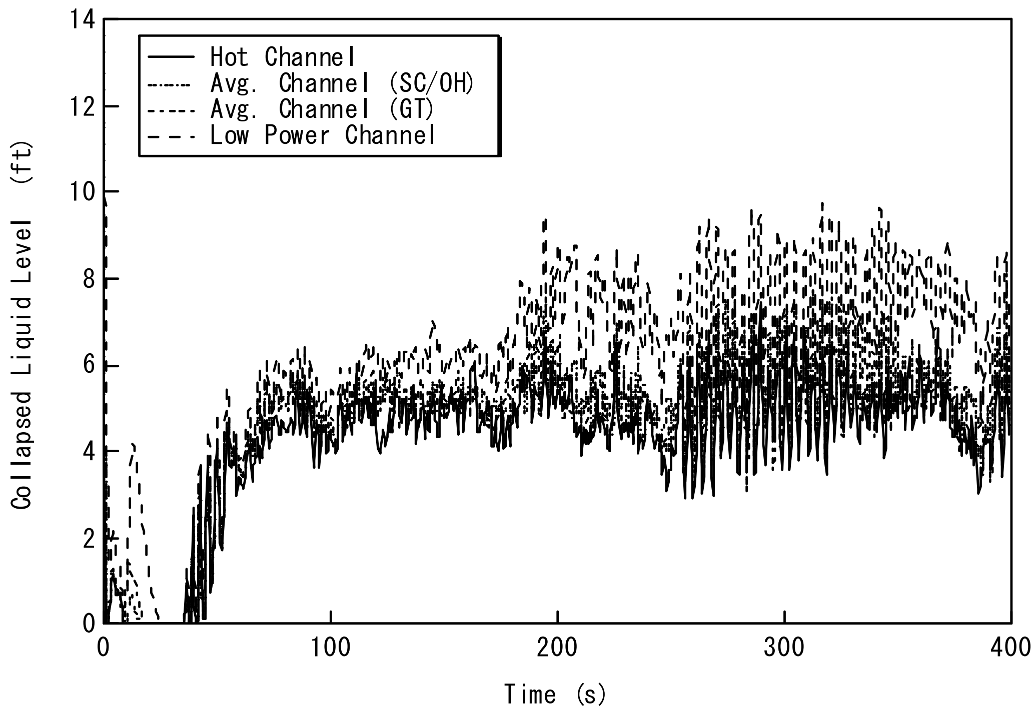


Figure 15.6.5-6 **Core Collapsed Liquid Level for Large Break LOCA (Reference Case)**

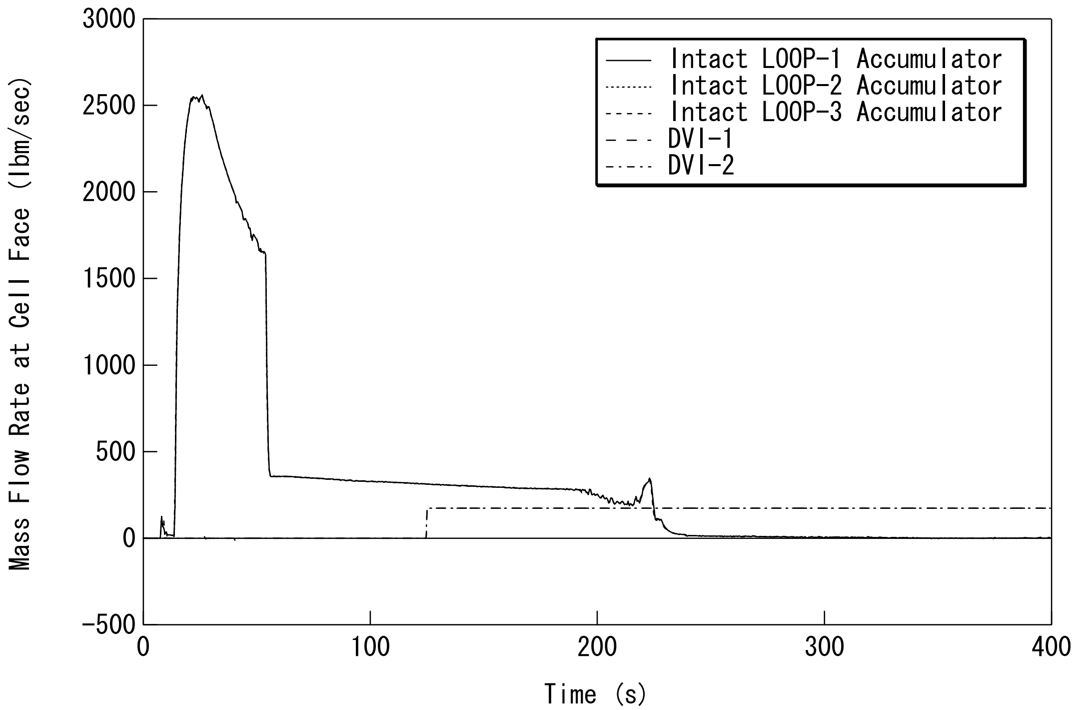
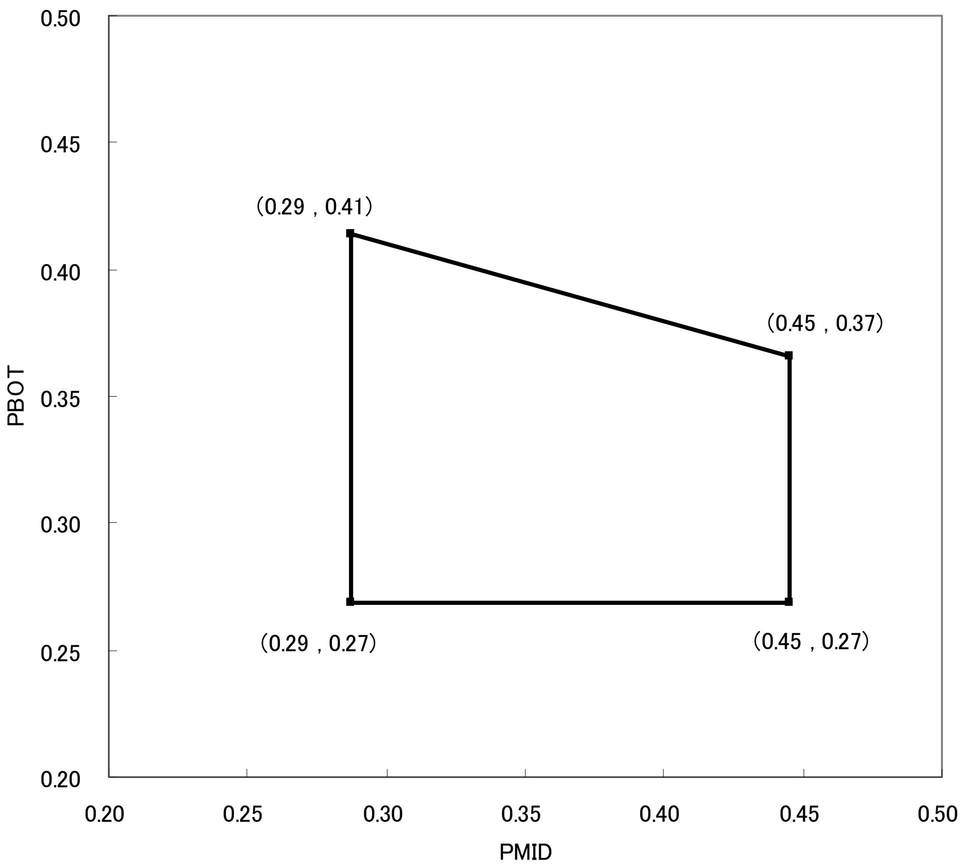


Figure 15.6.5-7 Accumulators and SI System Flowrates to DVI-1 and -2 for Large Break LOCA (Reference Case)



PBOT: integrated power fraction in the lower 3rd of the core
PMID: integrated power fraction in the middle 3rd of the core.

Figure 15.6.5-8 Axial Power Shape Operating Space Envelope for Large Break LOCA

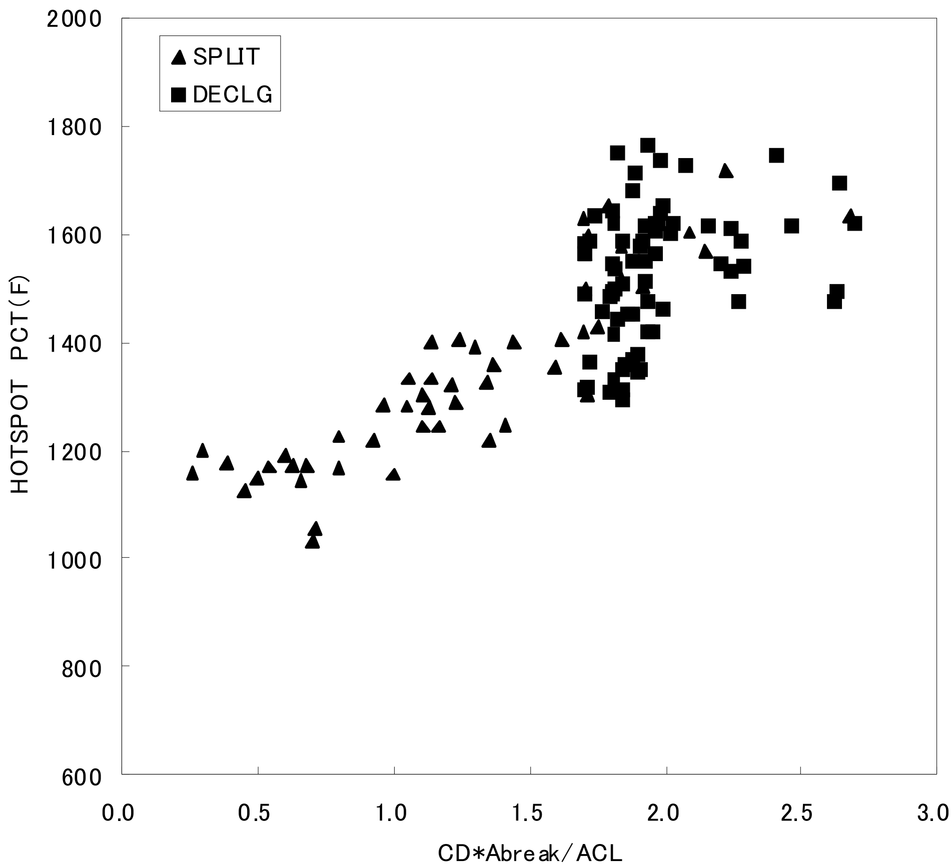


Figure 15.6.5-9 HOTSPOT PCT versus Effective Break Area Scatter Plot for Large Break LOCA

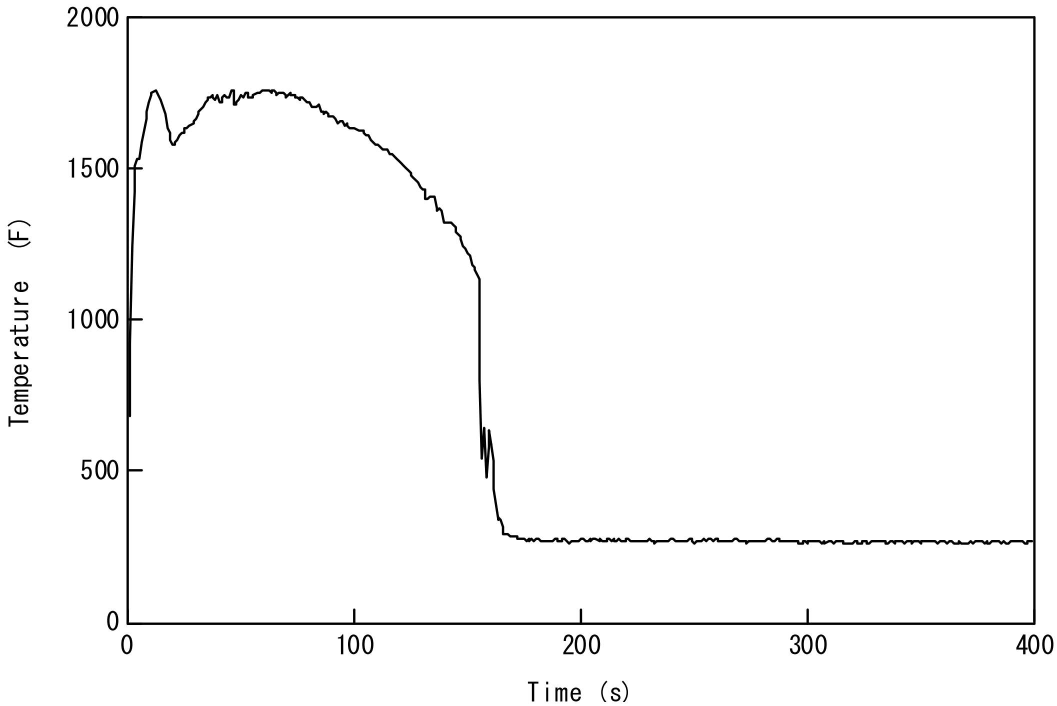


Figure 15.6.5-10 **HOTSPOT Cladding Temperature Transient at the Limiting Elevation for the PCT Limiting Case for Large Break LOCA**

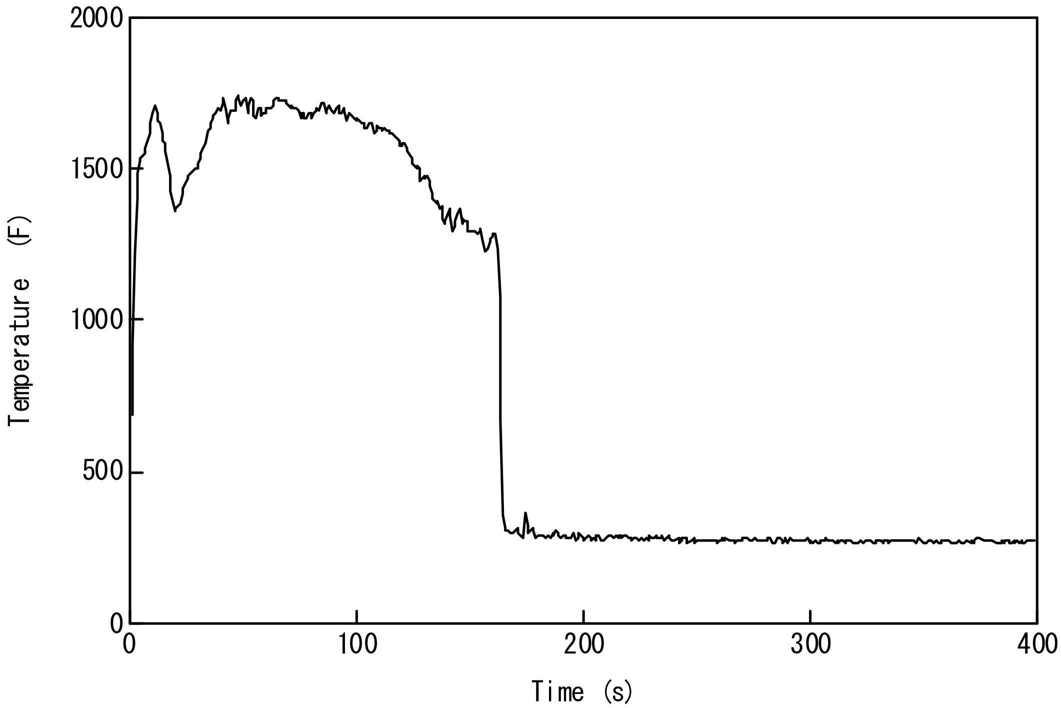


Figure 15.6.5-11 HOTSPOT Cladding Temperature Transient at the Limiting Elevation for the LMO Limiting Case for Large Break LOCA

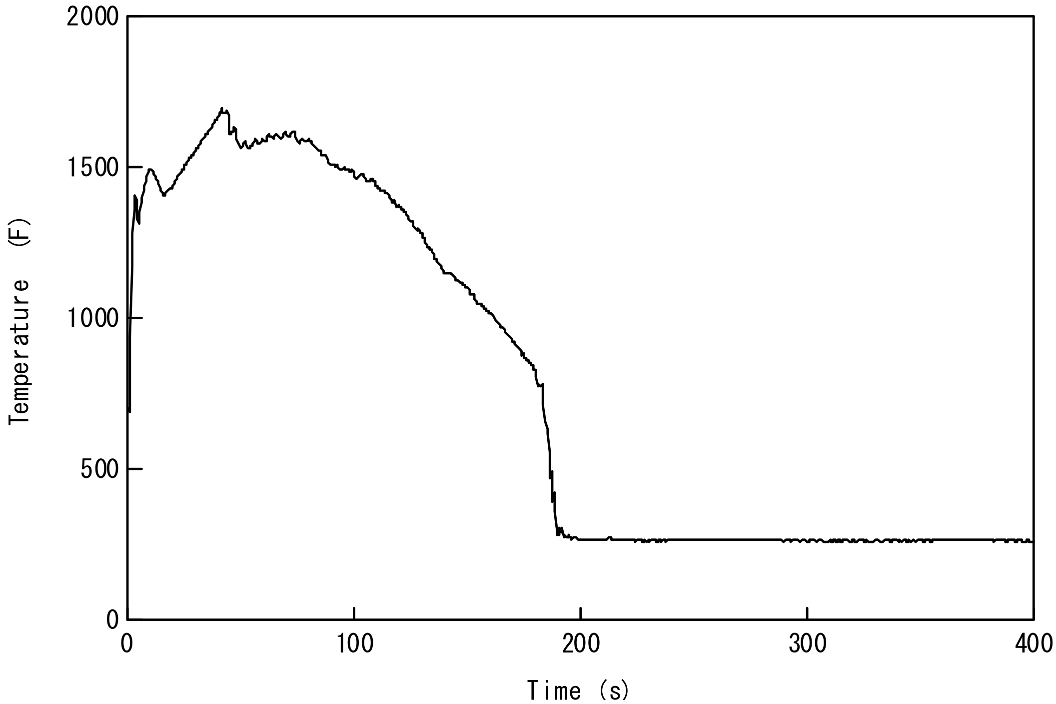


Figure 15.6.5-12 PCT Transient for the CWO Limiting Case for Large Break LOCA

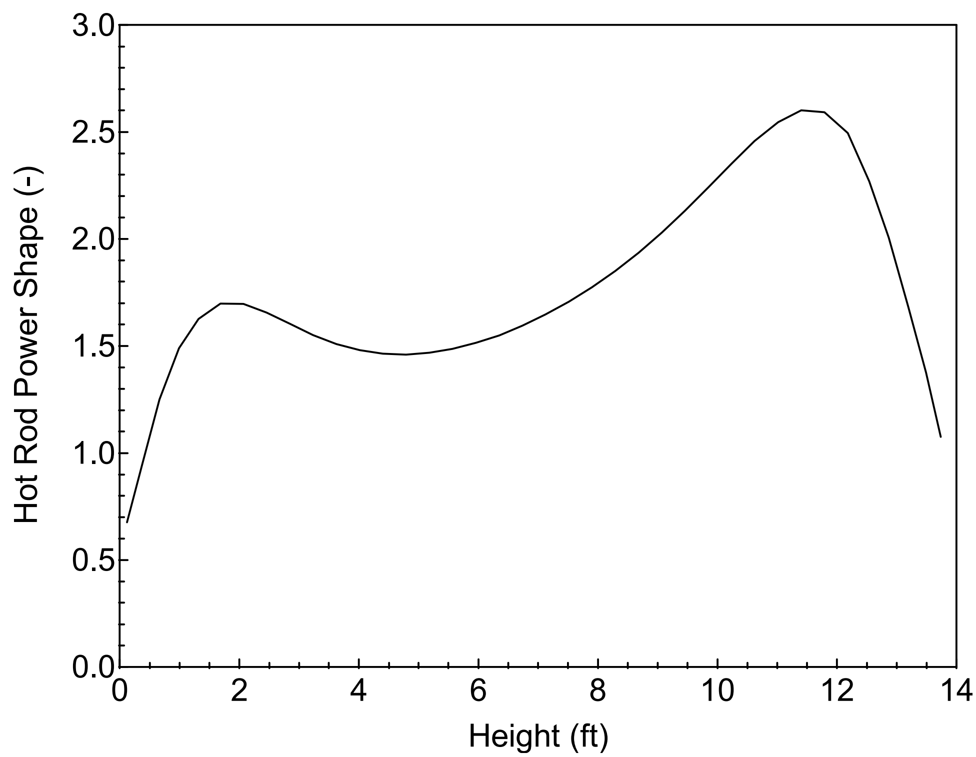


Figure 15.6.5-13 Hot Rod Power Shape Used for Small Break LOCA analysis

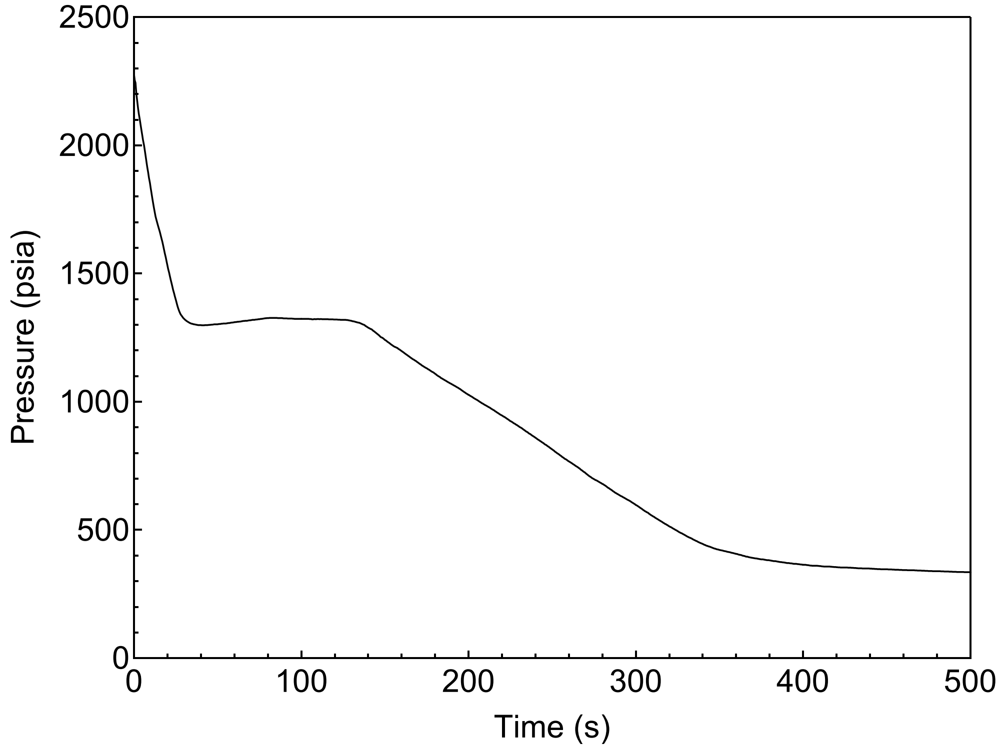


Figure 15.6.5-14 RCS (Pressurizer) Pressure Transient for 7.5-inch Small Break LOCA

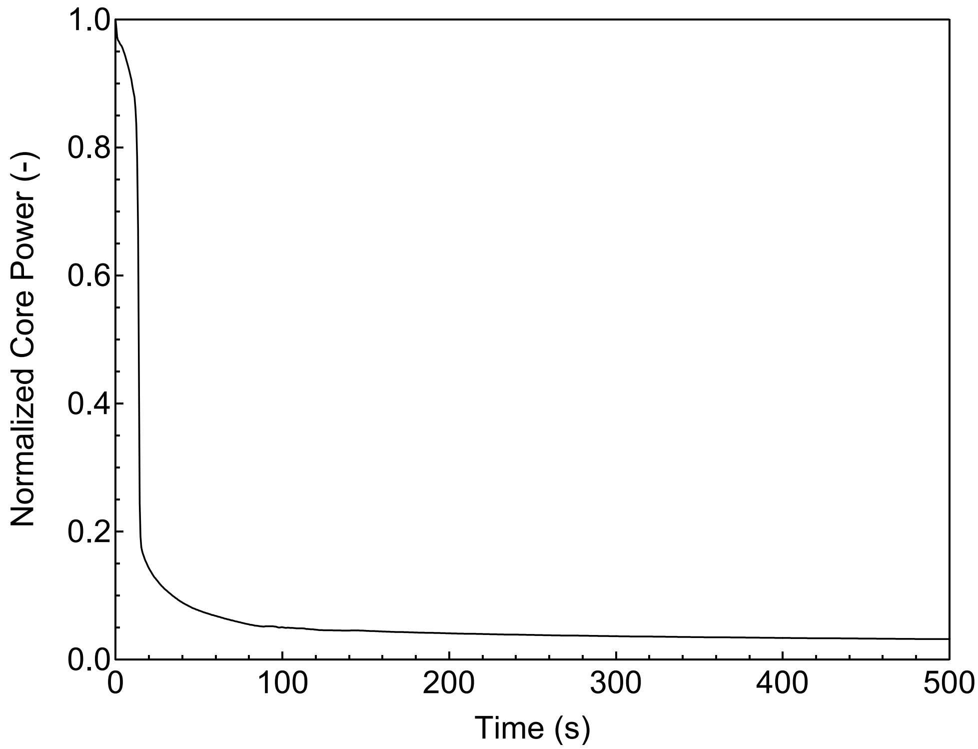


Figure 15.6.5-15 Normalized Core Power for 7.5-inch Small Break LOCA

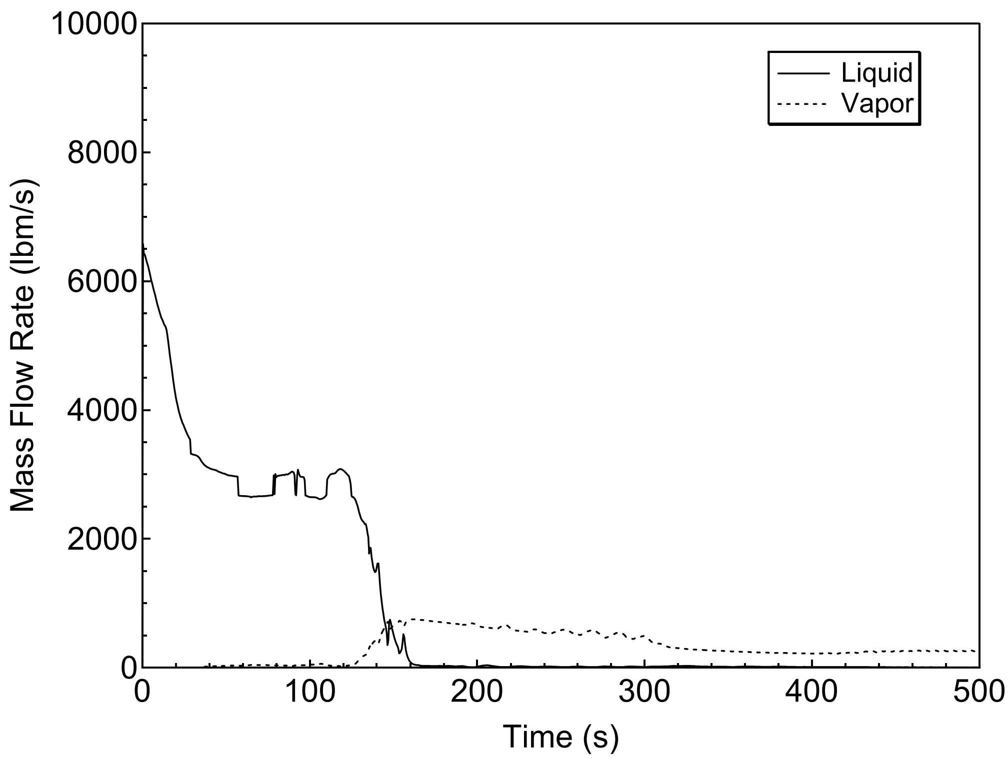


Figure 15.6.5-16 Liquid and Vapor Discharges through the Break for 7.5-inch Small Break LOCA

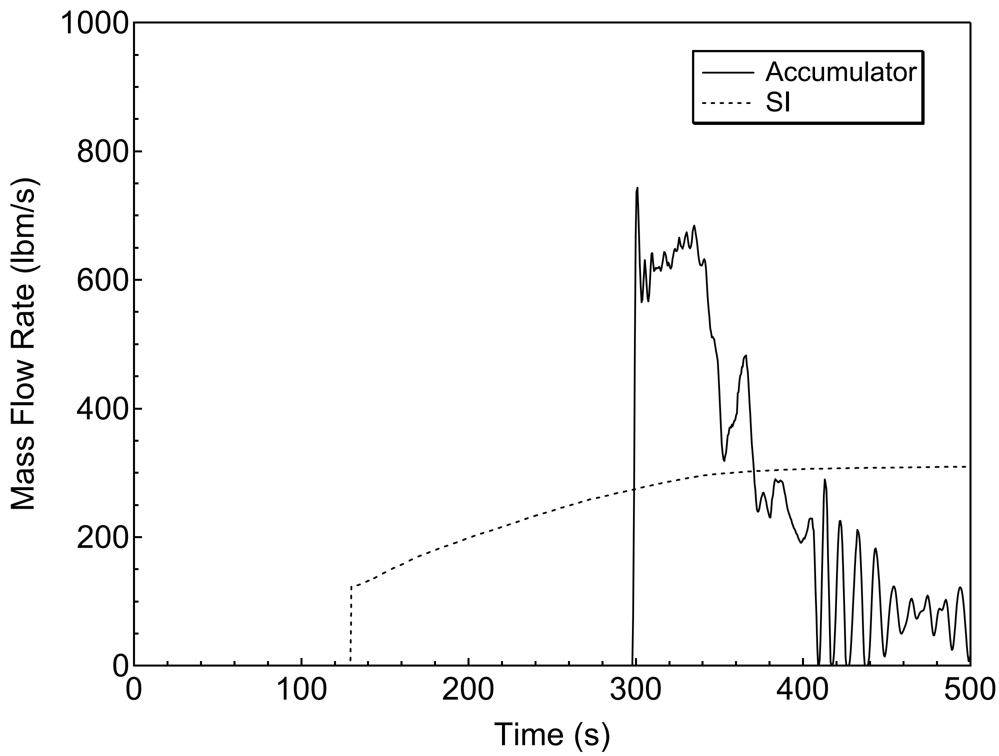


Figure 15.6.5-17 Accumulator and Safety Injection Mass Flowrates for 7.5-inch Small Break LOCA

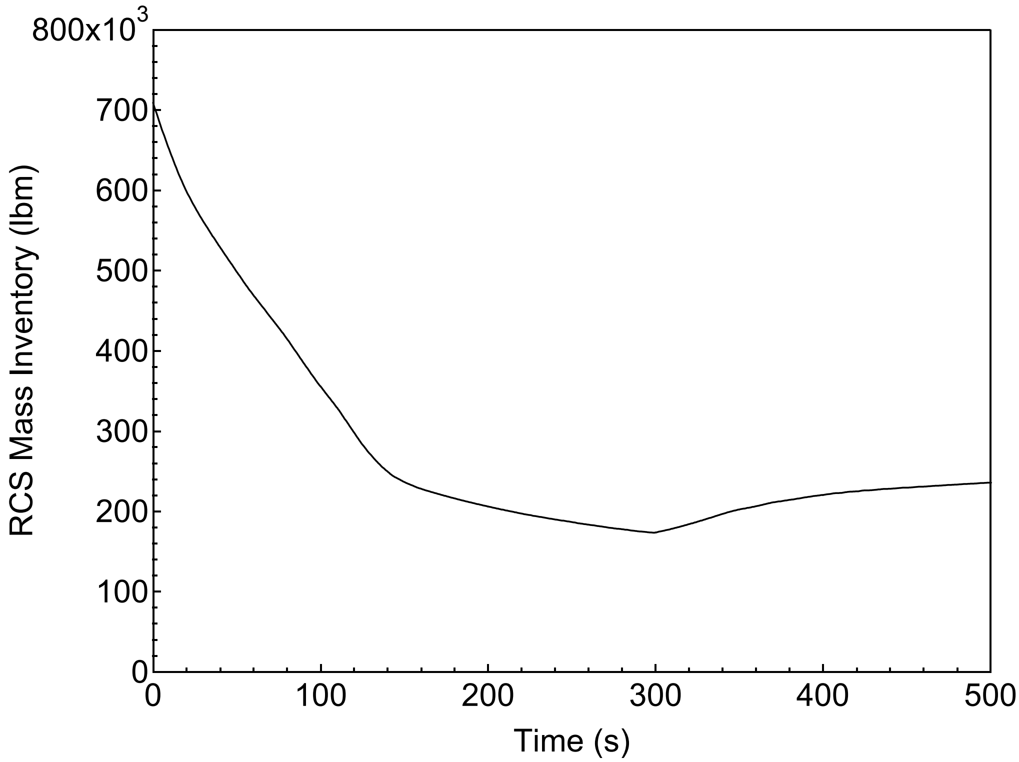


Figure 15.6.5-18 RCS Mass Inventory for 7.5-inch Small Break LOCA

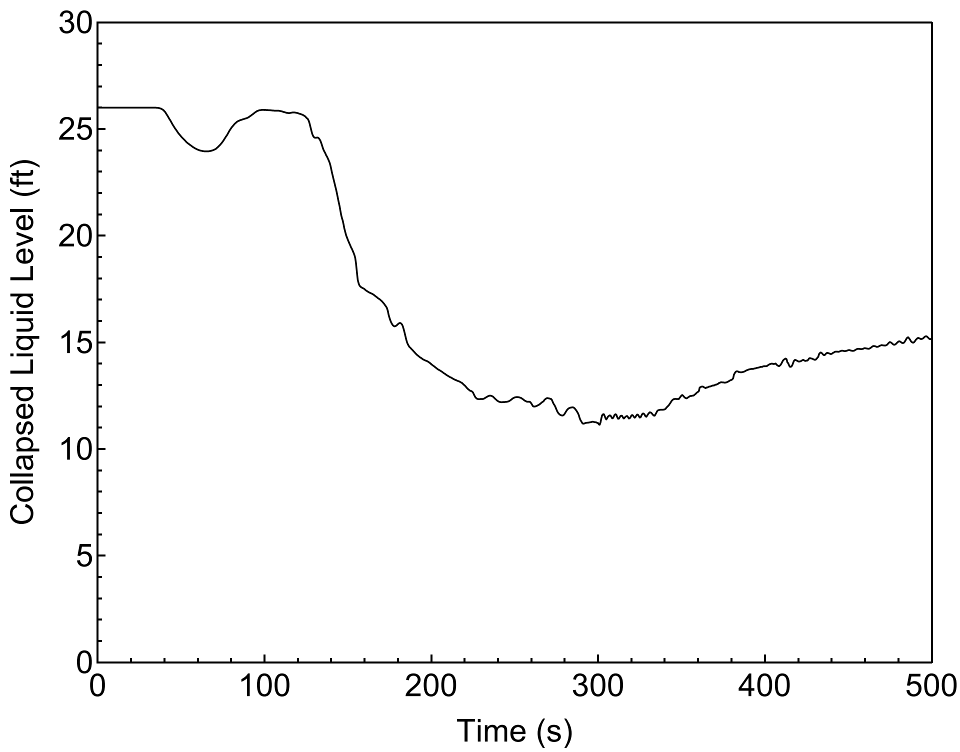


Figure 15.6.5-19 Downcomer Collapsed Level for 7.5-inch Small Break LOCA

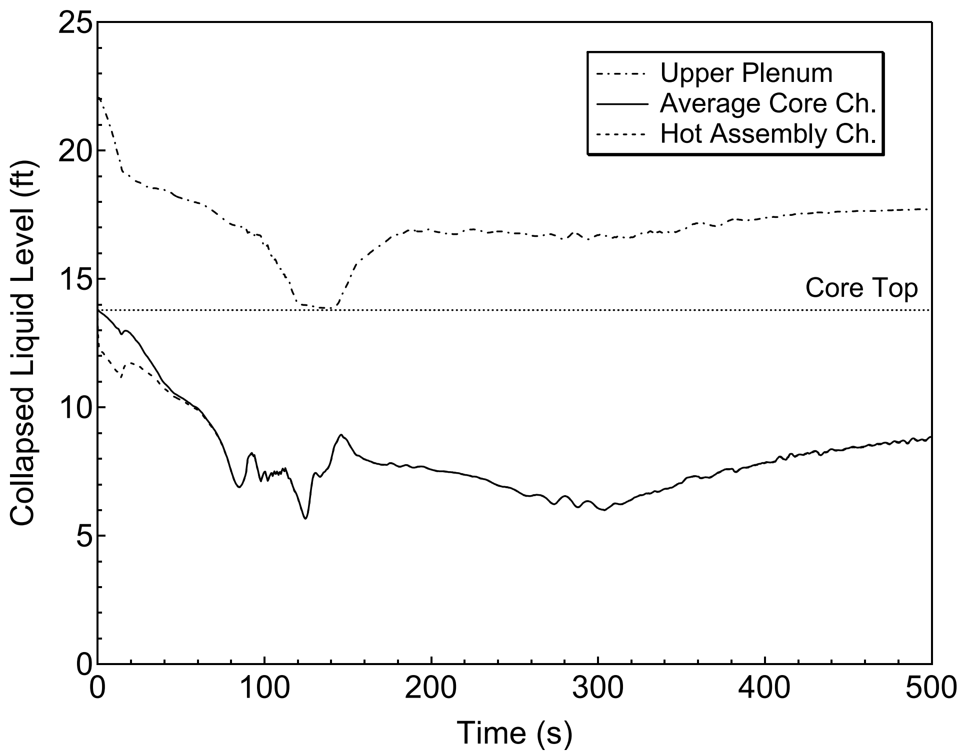


Figure 15.6.5-20 Core/Upper Plenum Collapsed Level for 7.5-inch Small Break LOCA

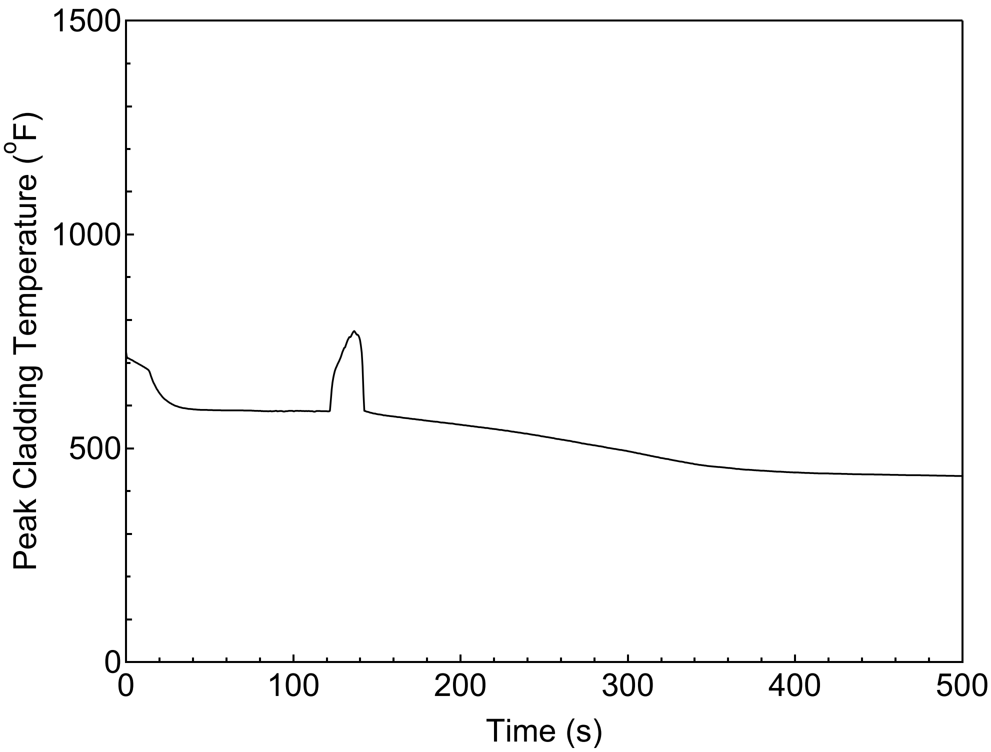


Figure 15.6.5-21 **PCT at All Elevations for Hot Rod in Hot Assembly for 7.5-inch Small Break LOCA**

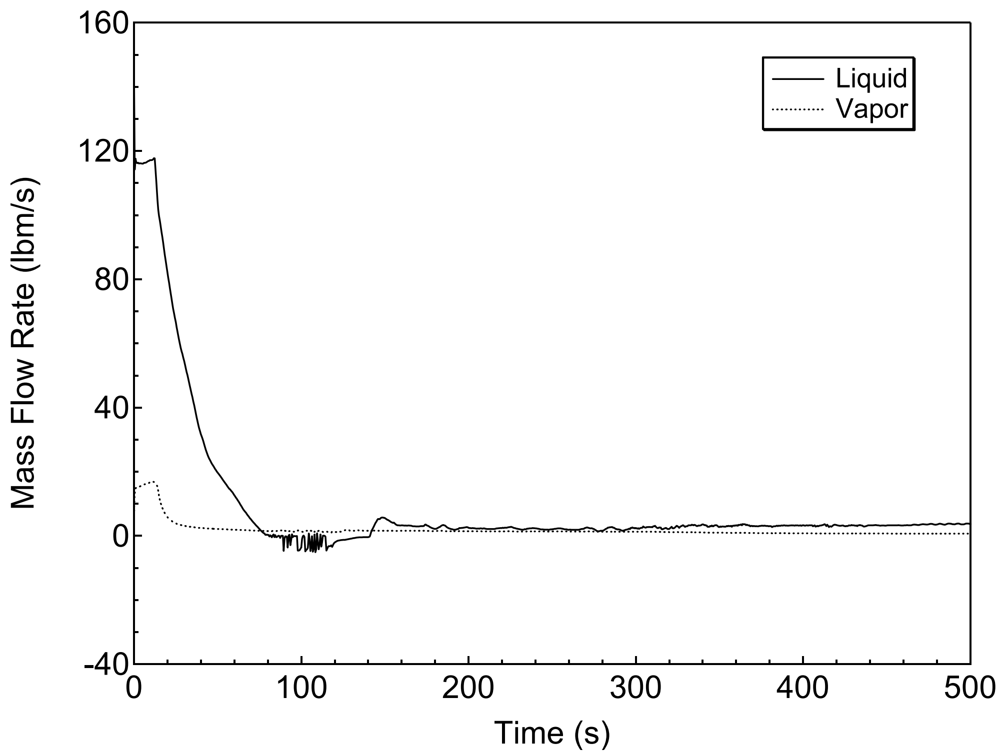


Figure 15.6.5-22 Hot Assembly Exit Vapor and Liquid Mass Flowrates for 7.5-inch Small Break LOCA

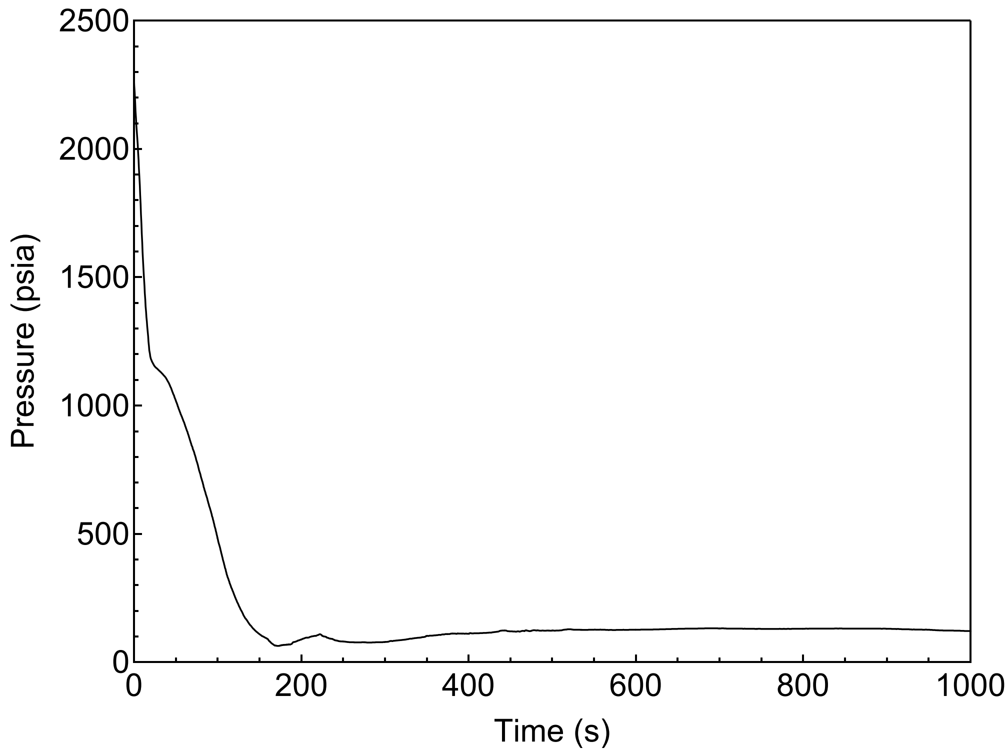


Figure 15.6.5-23 RCS (Pressurizer) Pressure Transient for 1-ft^2 Small Break LOCA

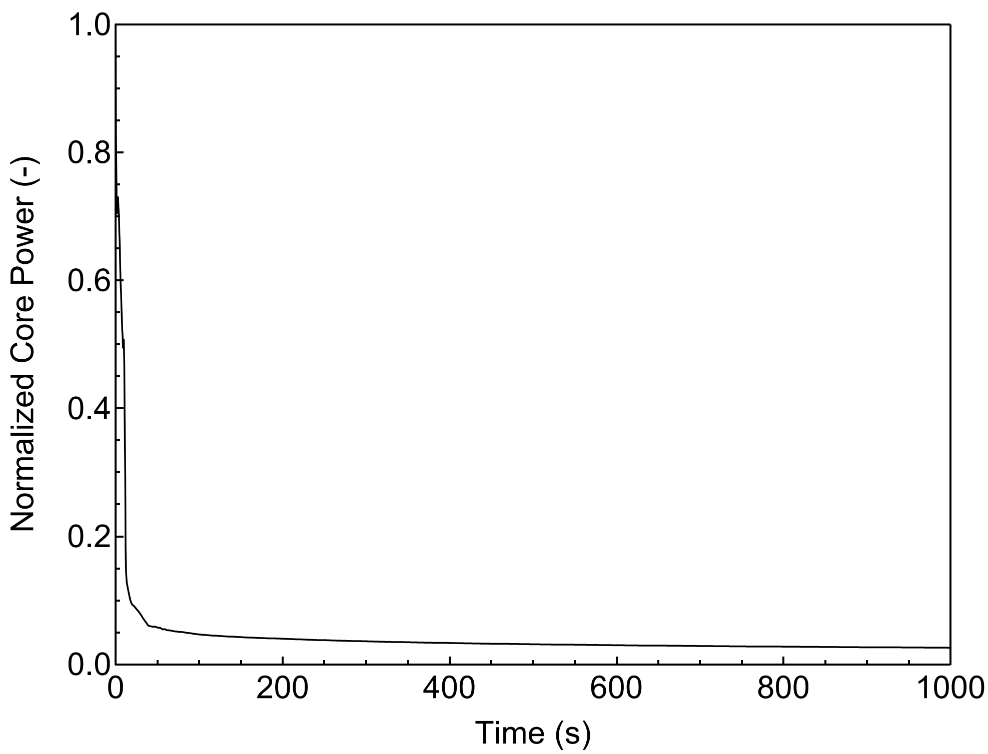


Figure 15.6.5-24 Normalized Core Power for 1-ft^2 Small Break LOCA

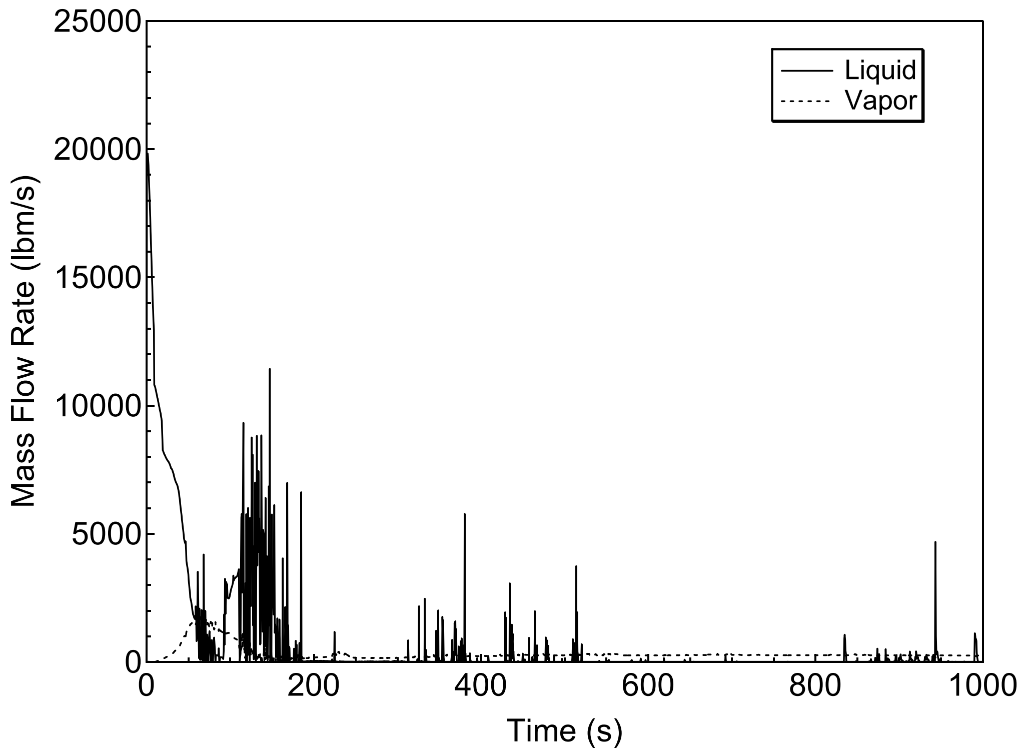


Figure 15.6.5-25 Liquid and Vapor Discharges through the Break for 1-ft^2 Small Break LOCA

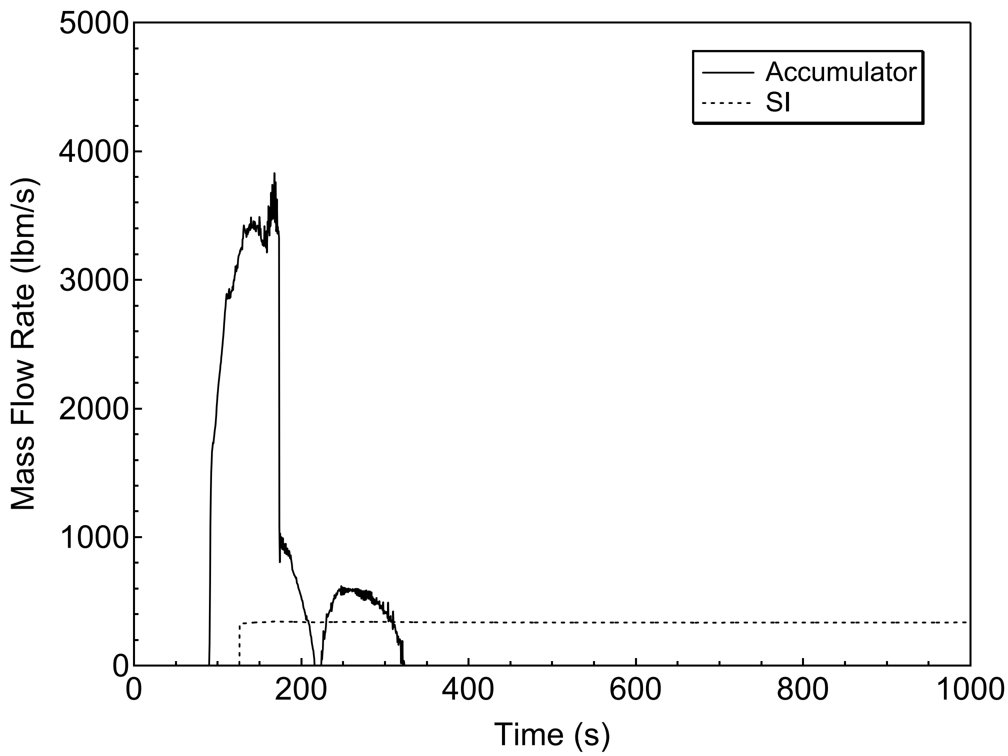


Figure 15.6.5-26 Accumulator and Safety Injection Mass Flowrates for 1-ft^2 Small Break LOCA

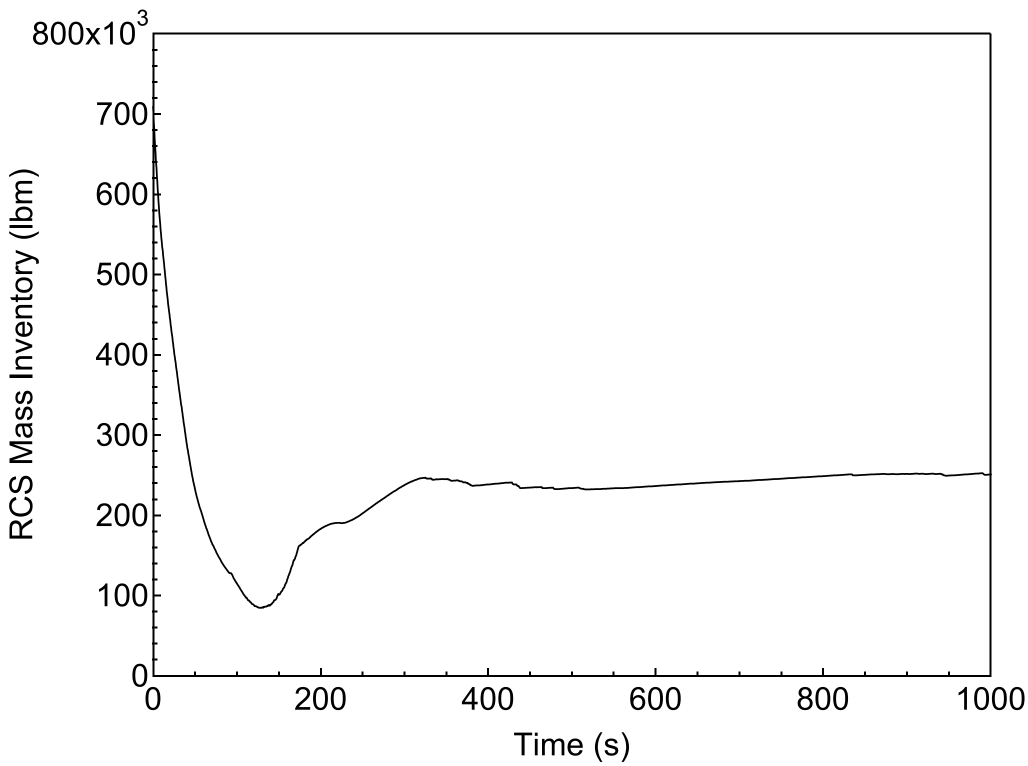


Figure 15.6.5-27 RCS Mass Inventory for 1-ft^2 Small Break LOCA

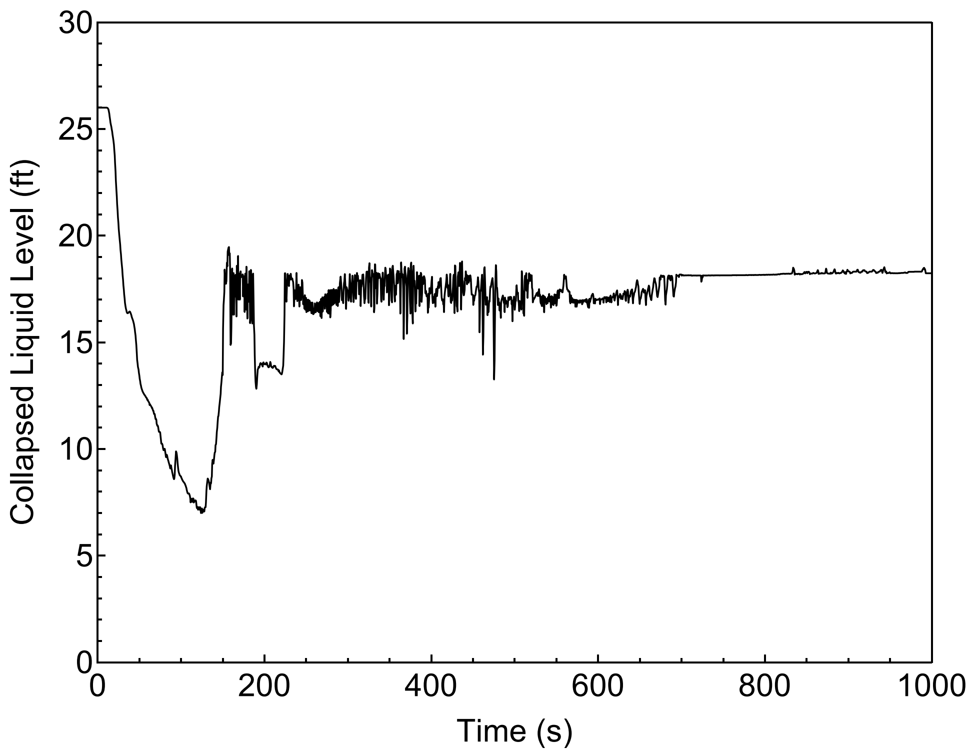


Figure 15.6.5-28 Downcomer Collapsed Level for 1-ft^2 Small Break LOCA

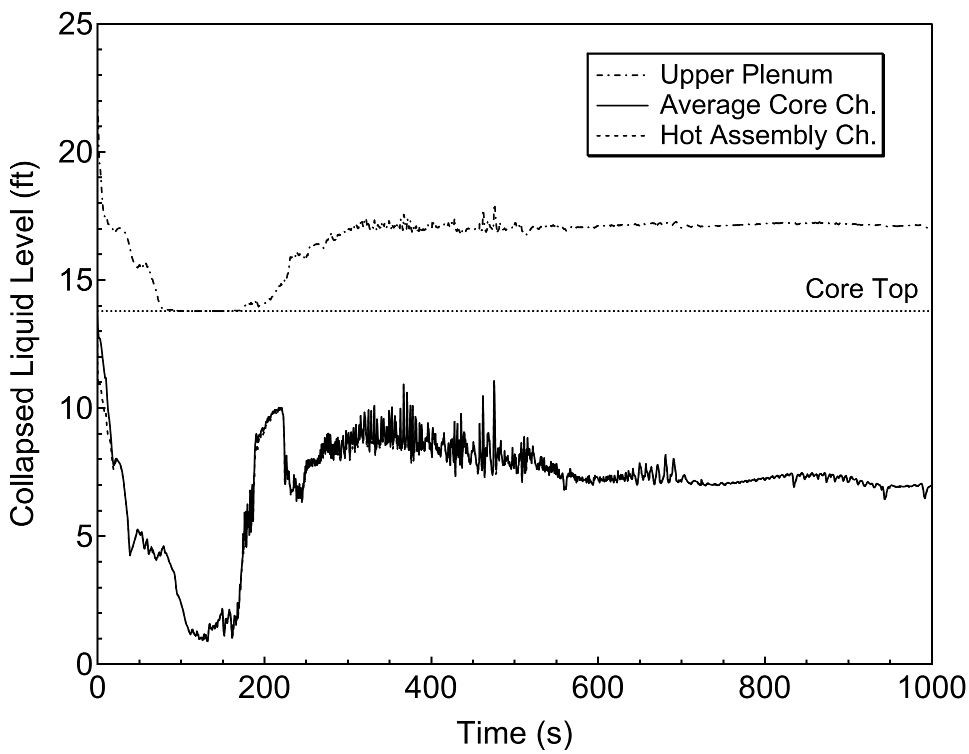


Figure 15.6.5-29 Core/Upper Plenum Collapsed Level for 1-ft^2 Small Break LOCA

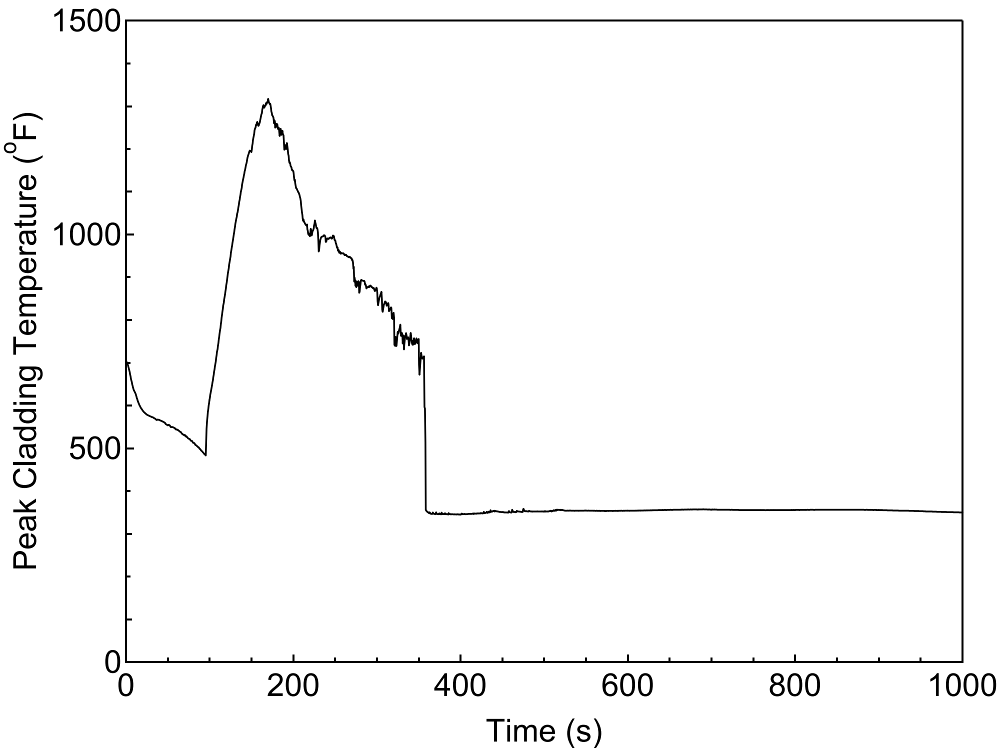


Figure 15.6.5-30 PCT at All Elevations for Hot Rod in Hot Assembly for 1-ft^2 Small Break LOCA

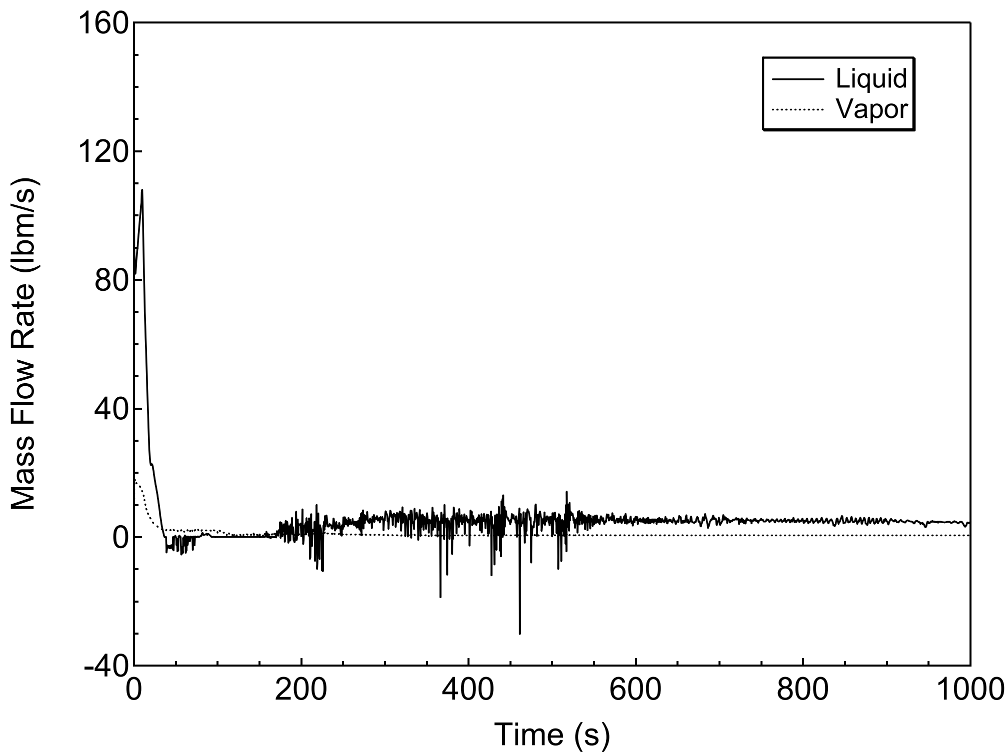


Figure 15.6.5-31 **Hot Assembly Exit Vapor and Liquid Mass Flowrates for 1-ft^2 Small Break LOCA**

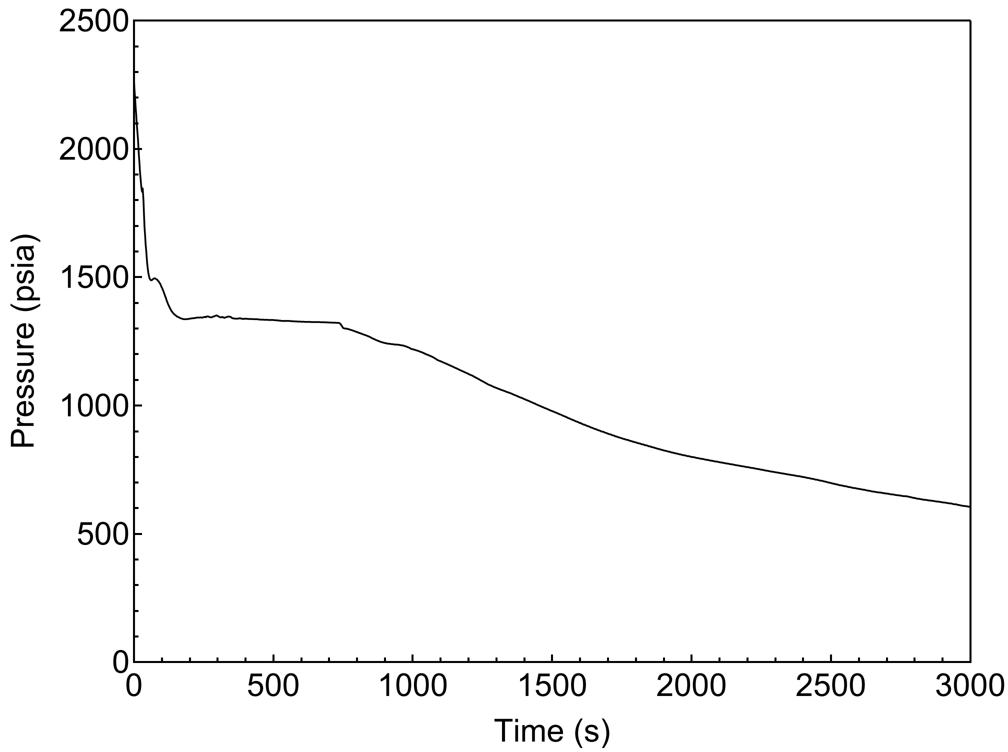


Figure 15.6.5-32 RCS (Pressurizer) Pressure Transient for DVI-line Small Break LOCA

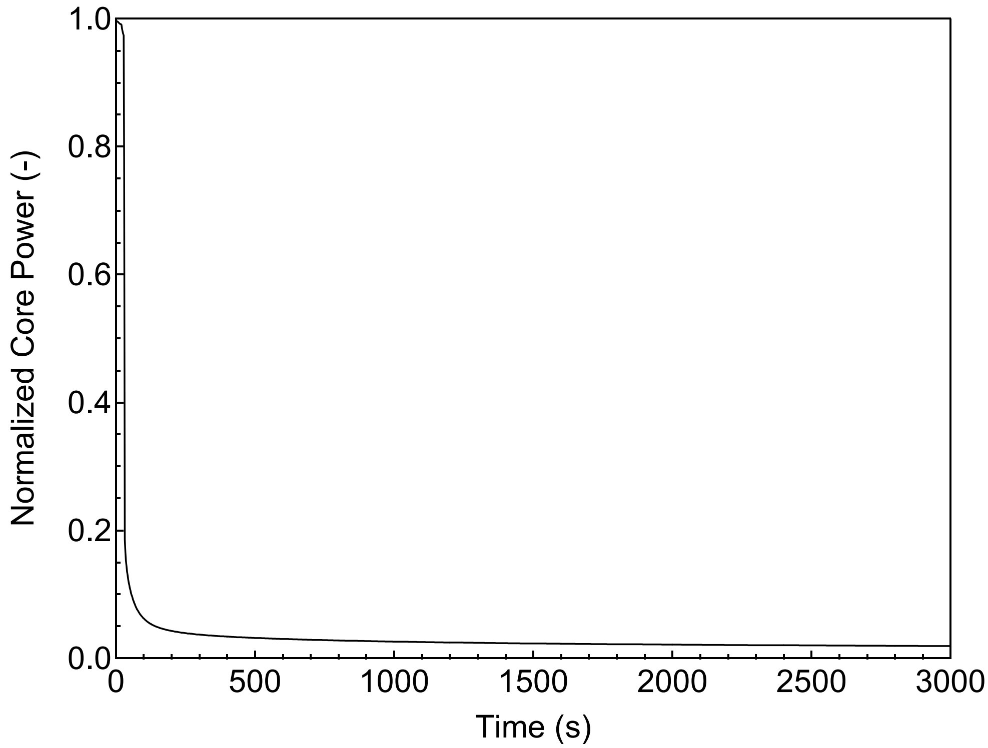


Figure 15.6.5-33 Normalized Core Power for DVI-line Small Break LOCA

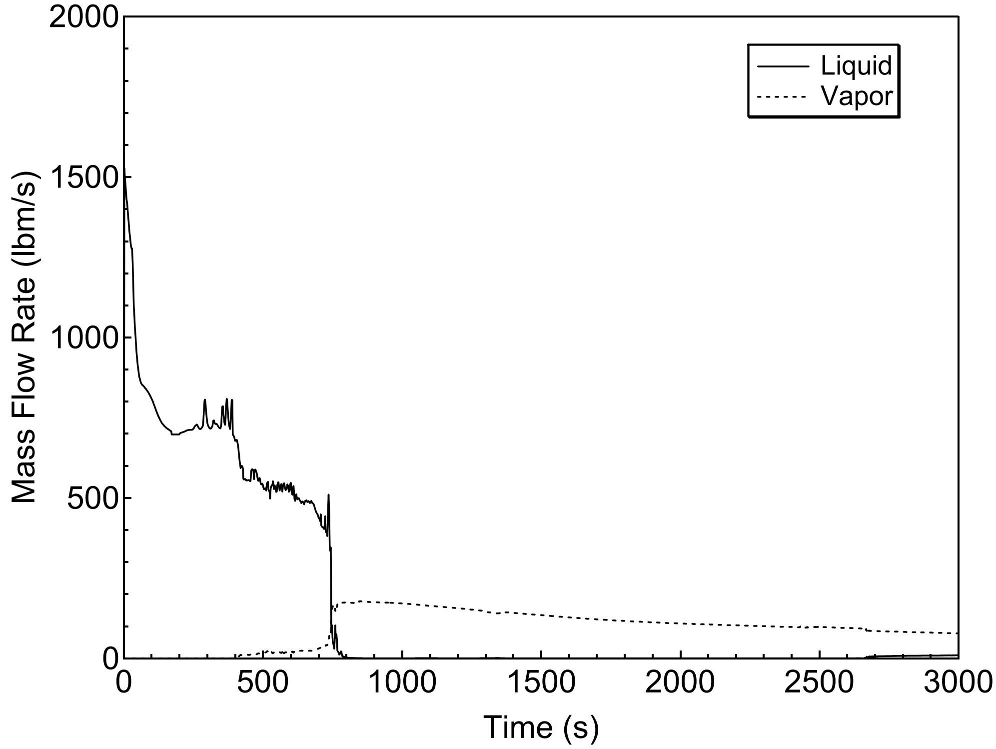


Figure 15.6.5-34 Liquid and Vapor Discharges through the Break for DVI-line Small Break LOCA

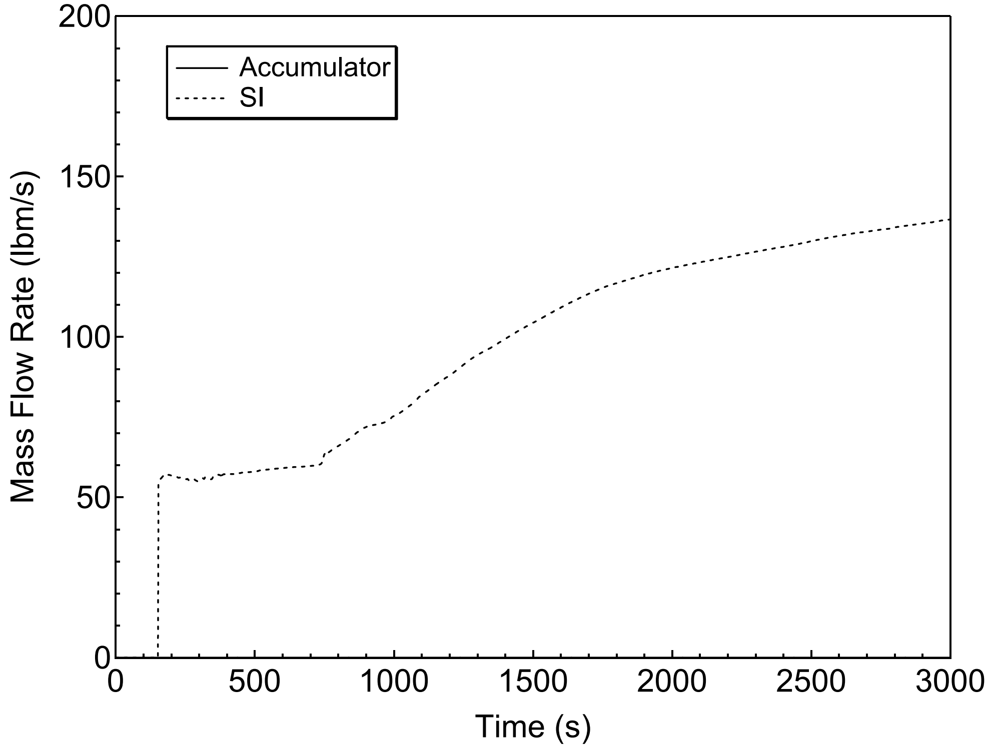


Figure 15.6.5-35 Accumulator and Safety Injection Mass Flowrates for DVI-line Small Break LOCA

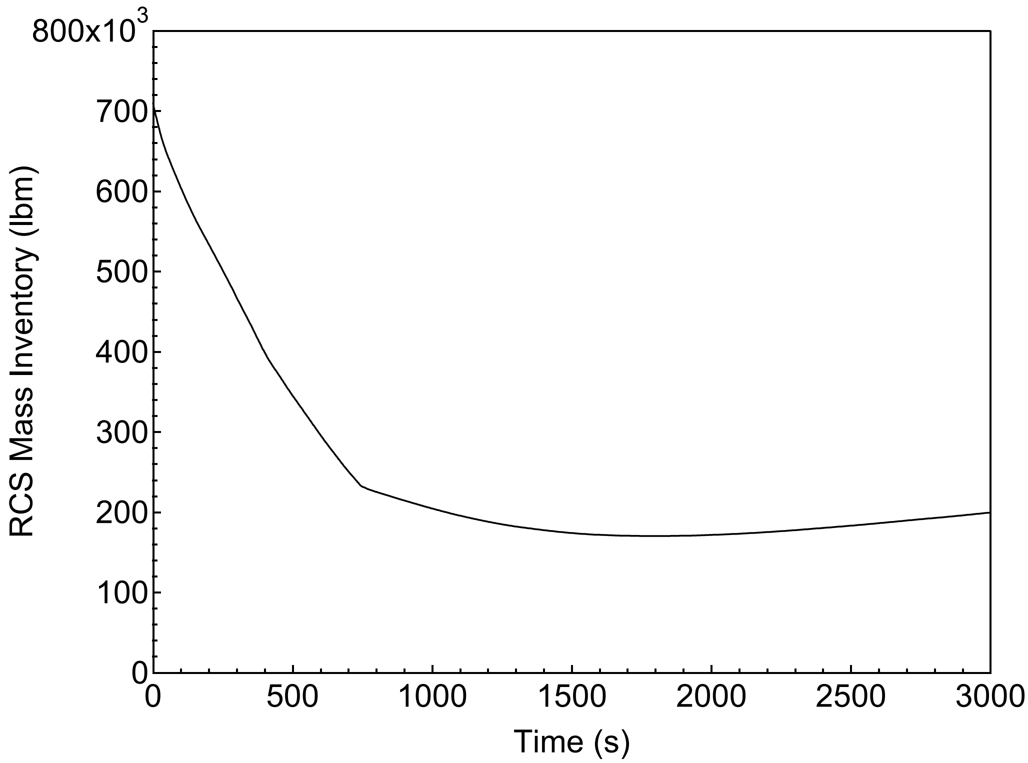


Figure 15.6.5-36 RCS Mass Inventory for DVI-line Small Break LOCA

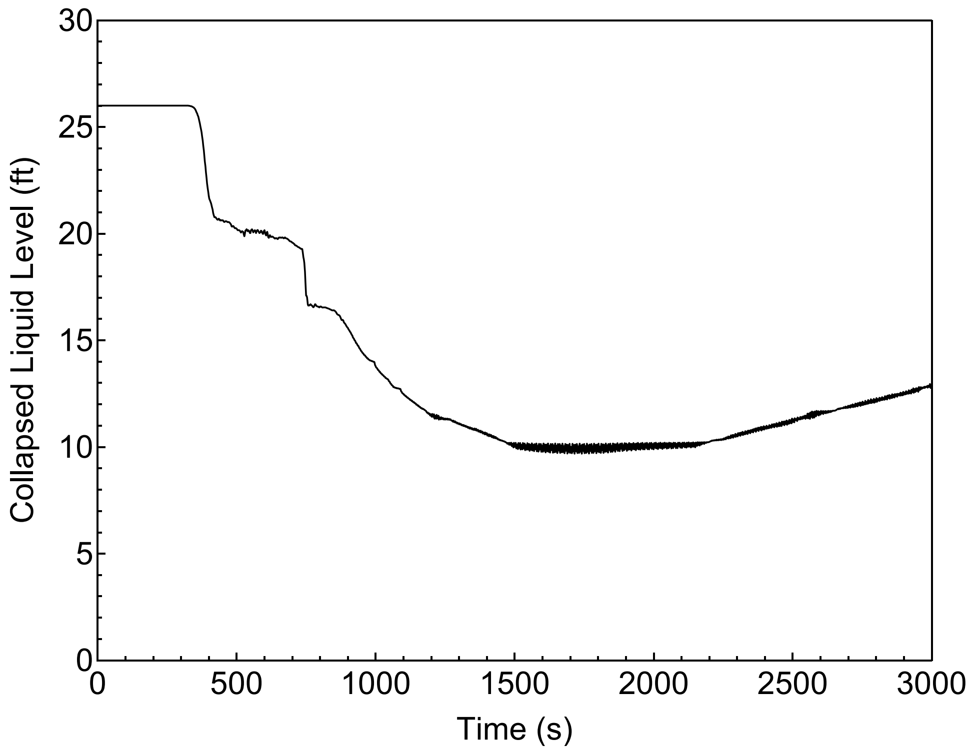


Figure 15.6.5-37 Downcomer Collapsed Level for DVI-line Small Break LOCA

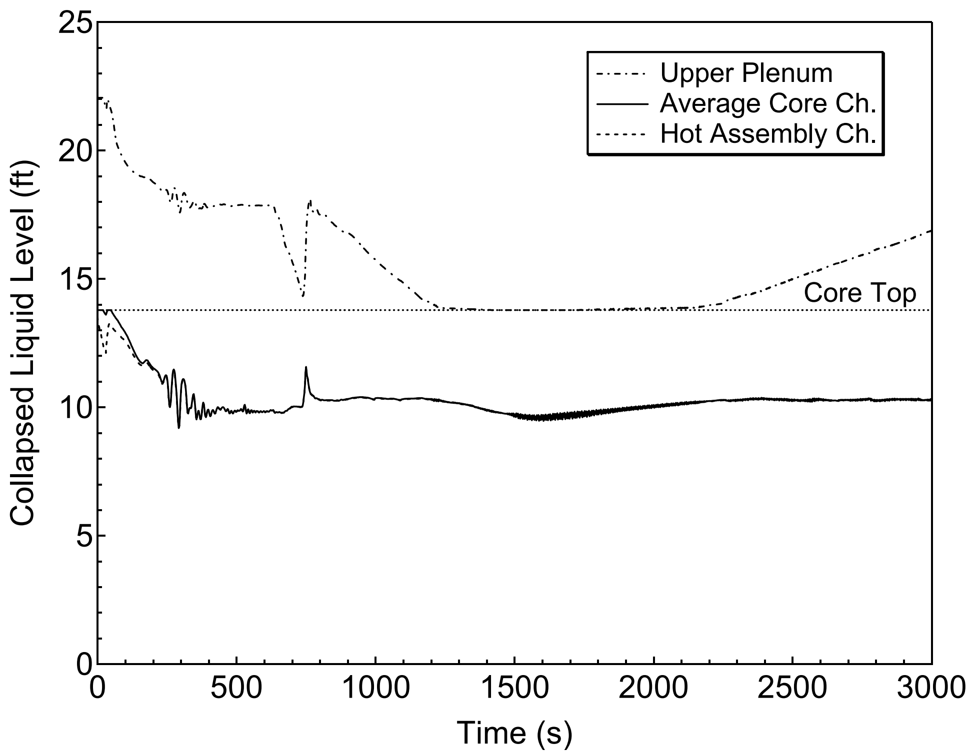


Figure 15.6.5-38 Core/Upper Plenum Collapsed Level for DVI-line Small Break LOCA

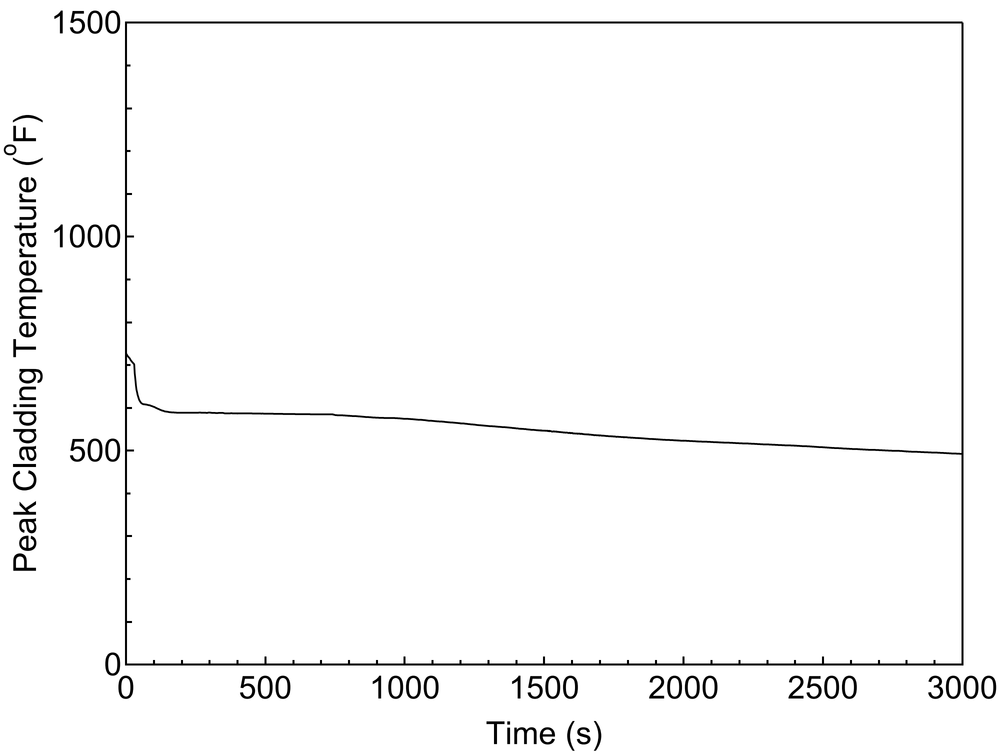


Figure 15.6.5-39 PCT at All Elevations for Hot Rod in Hot Assembly for DVI-line Small Break LOCA

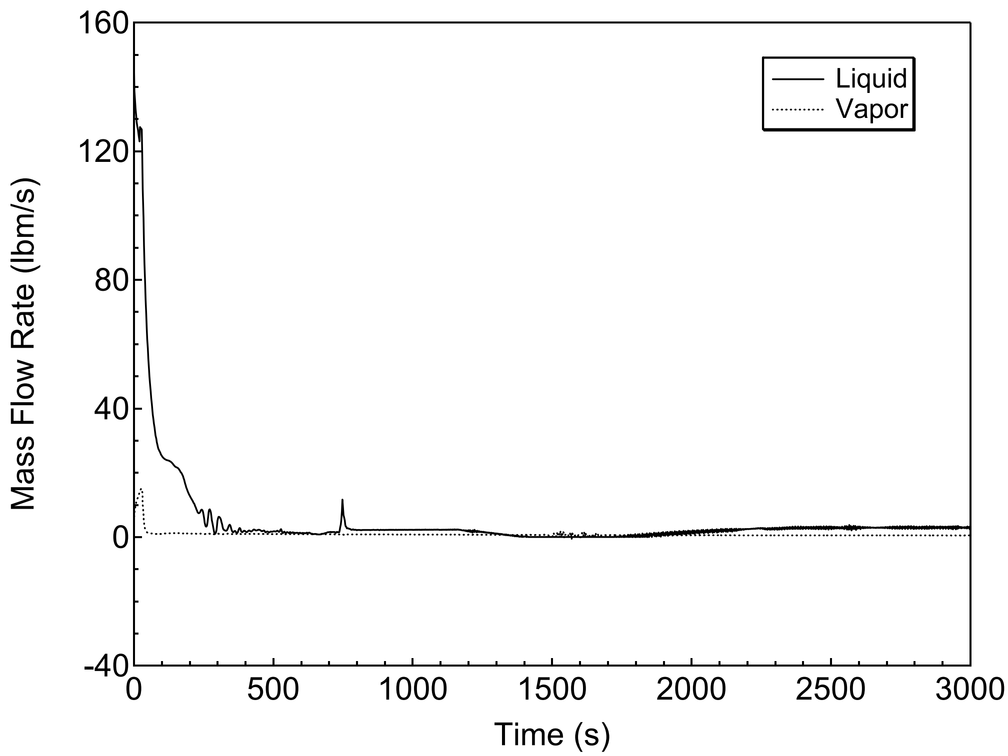
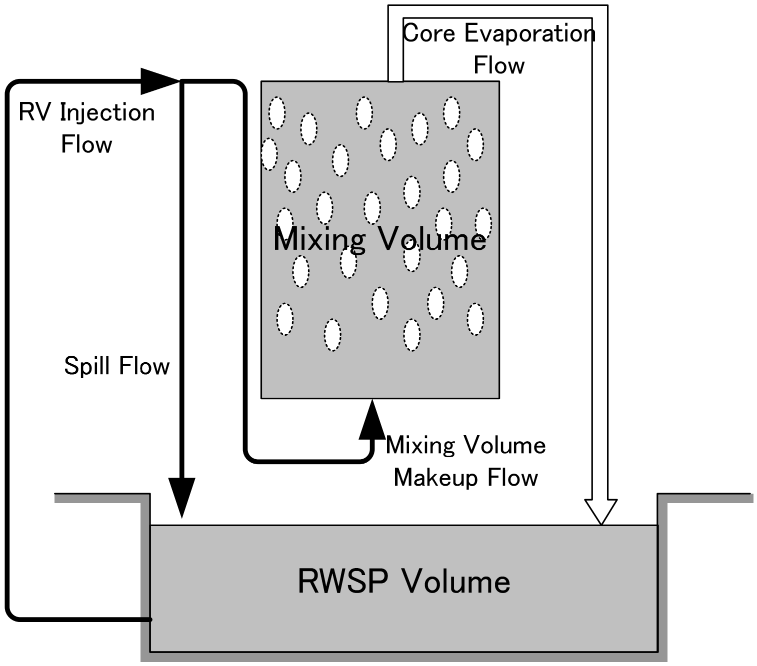
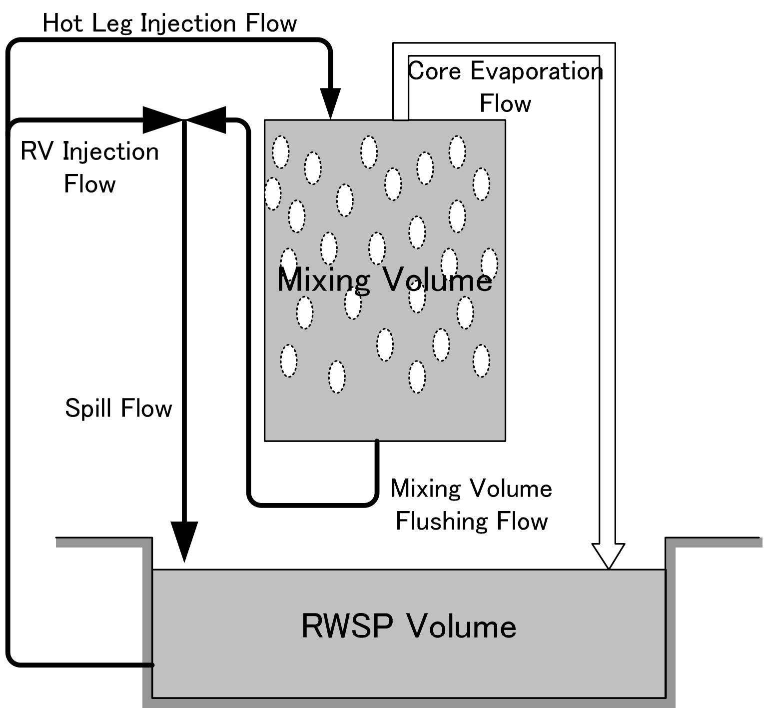


Figure 15.6.5-40 Hot Assembly Exit Vapor and Liquid Mass Flowrates for DVI-line Small Break LOCA



(a) RV injection mode



(b) Simultaneous RV and hot leg injection mode

Figure 15.6.5-41 Post-LOCA Long Term Cooling Evaluation Model

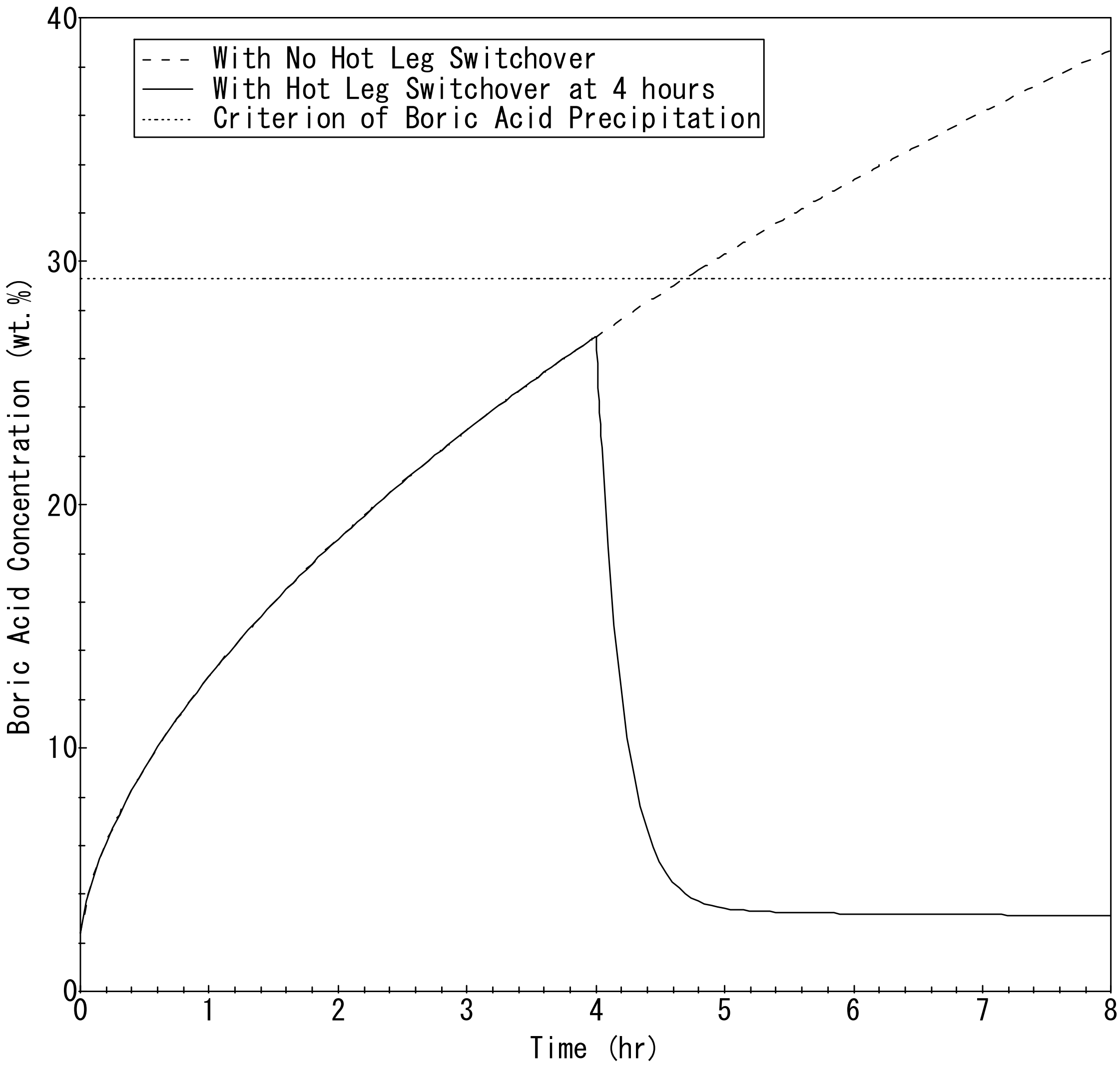


Figure 15.6.5-42 US-APWR Post LOCA Long Term Cooling Evaluation for 14.7 psia

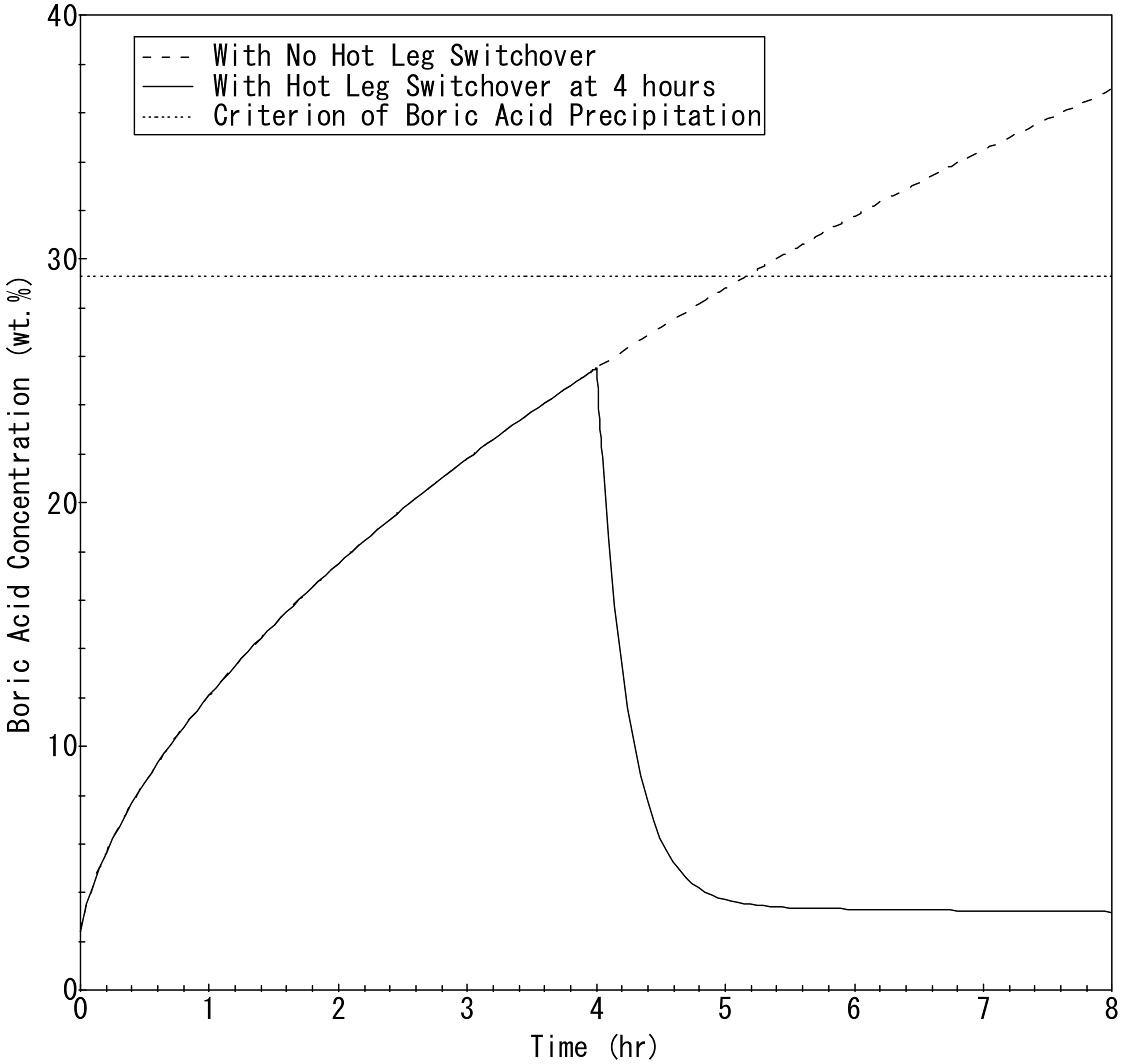


Figure 15.6.5-43 US-APWR Post LOCA Long Term Cooling Evaluation for 120 psia

15.6.6 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.6.7 References

15.6-1 Non-LOCA Methodology, MUAP-07010-P (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.6-2 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary), May 2007.

15.6-3 American National Standards Institute (ANSI) N18.2-1973 / American Nuclear Society (ANS) 18.2-1973, Nuclear Safety Criteria for the Design of Stationary PWR Plants (Historical).

15.6-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.6-5 Best-Estimate Calculations of Emergency Core Cooling System Performances, NRC Regulatory Guide 1.157, May 1989.

15.6-6 Qualifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident, NUREG/CR-5249, 1989.

15.6-7 Code Qualification Document for Best Estimate LOCA Analysis, WCAP-12945-P-A (Proprietary), Volume I, Revision 2, and Volumes II-V, Revision 1, and WCAP-14747 (Non-Proprietary), Westinghouse Electric Company, March 1998.

15.6-8 Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method, WCAP-16009-NP (Non-Proprietary) Revision 0, Westinghouse Electric Company, May 2003.

15.6-9 Large Break LOCA Applicability Report for US-APWR, MUAP-07011 (Proprietary) and MUAP-07010-NP (Non-Proprietary), July 2007.

15.6-10 Beaver Valley Power Station, Unit Nos.1 and 2 Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173, FENOC letter dated July 8, 2005, to NRC.

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15.6-23 Laboratory Testing of Nuclear Grade Activated Charcoal, U.S. Nuclear Regulatory Commission, Generic Letter 99-02, June 3, 1999.

15.6-24 ESF Atmosphere Cleanup System, U.S. Nuclear Regulatory Commission, NUREG 0800 (Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants), Chapter 6.5.1 Revision 3, March 2007.

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15.6-27 H. C. Yeh, Modification of Void Fraction Calculation, Proceedings of the Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Volume 1, Taipei, Taiwan, June 6,1988.

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15.7 Radioactive Release from a Subsystem or Component

Analyses of the following events are described in this section:

- Gaseous waste management system leak or failure
- Liquid waste management system leak or failure
- Release of radioactivity to the environment due to a liquid tank failure
- Fuel handling accident
- Spent fuel cask drop accident

15.7.1 Gas Waste Management System Leak or Failure

The US-APWR gaseous waste management system consists of waste gas compressor packages, gas surge tanks, and a noble gas holdup system. Gaseous waste radioactivity is reduced by a noble gas holdup system using a charcoal absorption bed. A waste gas dryer is used to prevent the active carbon holdup capacity from deteriorating. Also, the leakage of radioactivity out of the system and the leakage of air into the system are minimized. The radioactivity and quantity of gaseous waste released to the environment is reduced as much as is reasonably achievable and is insignificant. The radioactivity level is monitored before release.

However, the SRP includes gaseous waste management system leak or failure as part of the NRC review in Section 11.3.3. Therefore, no accident analysis for this event is provided.

15.7.2 Liquid Waste Management System Leak or Failure (Atmospheric Release)

The potential for atmospheric release of radioactivity from the US-APWR liquid waste management system is insignificant.

The SRP (NUREG-0800) no longer includes liquid waste management system leak or failure (atmospheric release) as part of the NRC review. Therefore, no accident analysis for this event is provided.

15.7.3 Release of Radioactivity to the Environment Due to a Liquid Tank Failure

Liquid waste from each building includes equipment and floor drains, a chemical drain (except strong acids), and a detergent drain. These liquids are collected separately and stored in tanks in the auxiliary building. Process waste is treated with filters and demineralizers and tested to ensure that its radioactivity is within acceptable limits before it is discharged.

While release to the environment is highly unlikely, the SRP (NUREG-0800) includes release of radioactivity to the environment due to a liquid tank failure as part of the NRC review in Section 11.2.3. Therefore, no accident analysis for this event is provided.

15.7.4 Fuel Handling Accident

It can be postulated that a fuel handling accident could occur either in the fuel handling area or inside the containment. The postulated fuel handling accident consists of an event in which the cladding of all fuel rods in one assembly is ruptured under water, and all gap activity in the fuel is released.

The fuel handling machine is designed to fail as-is. This design feature prevents the fuel handling machine from dropping a spent fuel assembly in the event of a malfunction.

In the event of a loss of air pressure, the refueling machine grapple will fail as-is and the supported fuel assembly will not be dropped. Also, the refueling machine is equipped with interlocks which prevent a load from being lifted that exceeds a preset weight. This precludes a fuel assembly from being dropped due to excessive load.

Air cleanup and water cleanup systems are available and expected to function during this event, but are not credited for mitigation of dose consequences to the public because these systems are not engineered safety features.

The fuel handling accident is classified as a postulated accident (PA) event.

15.7.4.1 Evaluation Model

The radiological consequences evaluation for this event is based on the alternative source term (AST) guidance documented in Reference 15.7-1.

Mathematical models used in the analysis are described in the following sections:

- The offsite and onsite doses are calculated with the RADTRAD code.

- The atmospheric dispersion factors (χ/Q values) used in the analysis are described in Section 15.0.3.3.

- The total effective dose equivalent (TEDE) doses to a receptor at the exclusion area boundary (EAB) and outer boundary of the low-population zone (LPZ) are analyzed using the models described in Section 15.0.3.1 and Appendix 15A.

Figure 15A-6 depicts the leakage sources to the environment modeled in the dose computation.

For evaluating the radiological consequences due to a postulated fuel handling accident, the radioactivity released from the dropped fuel assembly through either inside the spent fuel pit or inside the containment is directly released to the environment. All radioactivity is released to the environment within a 2-hour period and with no cloud depletion by ground deposition during transport to the EAB and LPZ.

15.7.4.2 Input Parameters and Initial Conditions

The major assumptions and parameters used in the analysis are itemized in Tables 15.7.4-1, 15.0-8, 15.0-13 and 15.0-14.
Other assumptions relating to the transport, reduction, and release of radioactive material to the environment are those covered in Appendix B of RG 1.183 (Ref. 15.7-1)

15.7.4.2.1 Source Term

The source term for a fuel handling accident is derived according to RG 1.183 (Ref. 15.7-1) and depends on various factors such as the inventory of the fuel assembly, the assembly peaking factor, and the assembly cooling time. The source term for the fuel handling accident is based on the discussion in the sections presented below. The important factors and the derived source term are tabulated in Table 15.7.4-1.

15.7.4.2.1.1 Fission Product Gap Fraction

Fission products build up in the fuel pellet during power operation. These fission products diffuse through the fuel matrix and deposit into the gap between the fuel and cladding. The amount of fission products in the fuel/cladding gap is dependent upon the power level of the fuel assembly and the rate of diffusion for a particular radionuclide into the fuel/cladding gap. The percent of fission product that reaches the fuel/cladding gap is known as the fission product gap fraction. If the cladding is breached during a fuel handling accident, the gaseous and volatile radionuclides of the fission product gap fraction are assumed to instantaneously escape from the fuel assembly. The fission product gap fractions for this accident are consistent with the non-LOCA fractions of fission product inventory in the gap region, as specified in RG 1.183 (Ref. 15.7-1), as given in Table 15.0-8.

15.7.4.2.1.2 Chemical Form and Release of Radioiodine

The chemical form of radioiodine released from the damaged fuel assembly is assumed to be 95% CsI, 4.85% elemental iodine, and 0.15% organic iodine. The CsI released from the fuel is assumed to dissociate in the surrounding water and instantaneously re-evolve as elemental iodine.

The minimum depth of water above the fuel assembly when handling spent fuel is 23 feet. This allows for a decontamination factor (DF) for elemental iodine of 500 and a DF for organic iodine of 1. This results in an overall effective DF of 200, and therefore 99.5% of all iodine released is retained by the water.

15.7.4.2.1.3 Radiological Decay

Radiological decay of fission products conservatively credits the minimum amount of time between reactor shutdown and the start of fuel handling activities. For the fuel handling accident, fission product decay time is conservatively assumed to be 24 hours.

15.7.4.2.1.4 Assembly Power Level

The damaged fuel assembly is assumed to be the one operated for the longest time at maximum power (102% of nominal power). Every fuel rod in the damaged assembly is assumed to have operated at the maximum fuel rod radial peaking factor. This is a conservative assumption because most fuel rods operate below the maximum fuel rod radial peaking factor.

15.7.4.2.2 Release Pathway

Although the water above the damaged fuel acts to filter the elemental iodine, it is assumed to have no effect on organic iodine or noble gases (i.e., DF of 1).

The release duration is assumed to be 2 hours. It is assumed that after the radionuclides leave the water, they are released directly to the atmosphere within a period of 2 hours.

No credit for further filtration of the released radionuclides is taken for the fuel handling accident either in the fuel handling area or inside the containment.

15.7.4.3 Results

Offsite doses are calculated at the EAB and the outer boundary of the LPZ. As shown in Table 15.7.4-2, the dose at the EAB is calculated to be 3.3 rem TEDE. At the outer boundary of the LPZ, the dose is calculated to be 1.4 rem TEDE.

These doses are well within the dose limit of 25 rem TEDE given in 10 CFR 50.34. “Well within” is defined as being less than or equal to 25% of the dose limit.

The doses for the main control room (MCR) from the fuel handling accident are bounded by the doses calculated for the loss-of-coolant accident (LOCA) event described in Section 15.6.5.5. Consequently, no doses are provided for the fuel handling accident.

15.7.4.4 Conclusions

The resultant doses are well within the guideline values of 10 CFR 50.34.

Table 15.7.4-1
Fuel Handling Accident Source Term Assumptions

Parameter	Value
Core thermal power level (MWt)	4540 (2% above the design core thermal power)
Decay (cooling) time (h)	24
Core source term Noble gases (Xe & Kr) Iodine	 See Table 15.0-14. See Table 15.0-14.
Fission product gap fraction: I-131 Kr-85 Other noble gases (Xe & Kr) Other iodine	 See Table 15.0-8. See Table 15.0-8. See Table 15.0-8. See Table 15.0-8.
Number of fuel assemblies in core	257
Amount of fuel damaged	1 assembly
Maximum rod radial peaking factor	1.78
Water decontamination factor for iodine	200
Release duration (h)	2
χ/Q	See Table 15.0-13.
Breathing rate	See Table 15.0-13.
Dose conversion factors	See Table 15.0-14.

Table 15.7.4-2
Radiological Consequences of Fuel Handling Accident

Dose Location	TEDE Dose (rem)
EAB (0 to 2 hours)	3.3
LPZ outer boundary	1.4

15.7.5 Spent Fuel Cask Drop Accident

The overhead heavy load handling system is designed with single-failure-proof cranes. The use of the single-failure-proof crane precludes the need to perform load drop evaluations. Single-failure-proof cranes are designed so that any credible failure of a single component will not result in the loss of capability to stop and hold a critical load (See Section 9.1.5). Also, the spent fuel cask handling crane is prohibited from traveling over the spent fuel. Therefore, no accident analysis is necessary for a spent fuel cask drop accident.

15.7.6 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.7.7 References

15.7-1 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, NRC Regulatory Guide 1.183, July 2000.

15.8 Anticipated Transients without Scram

15.8.1 Identification of Causes and Frequency Classification

An anticipated transient without scram (ATWS) is an anticipated operational occurrence (AOO) followed by the failure of the automatic reactor trip portion of the reactor trip system. Since the reactor trip system must satisfy the single-failure criterion, multiple failures or a common mode failure must occur to cause the assumed failure of the reactor trip.

The frequency of an AOO, in coincidence with multiple failures or a common mode failure, is much lower than any of the other events that are analyzed in the US-APWR DCD Chapter 15. Therefore, the ATWS event cannot be classified as either an AOO or design basis accident (postulated accident), and has been historically considered as a beyond-design-basis event.

15.8.2 ATWS Rule (10 CFR 50.62) Design Requirements

In the 1970s, analyses performed by both the PWR reactor vendors and NRC as part of the ATWS Rulemaking showed that although failure of the reactor trip system could transform a minor transient into a severe accident, consequences from an ATWS would be acceptable (maintain peak reactor coolant system (RCS) pressure below 3200 psia) provided that the moderator temperature coefficient of reactivity was sufficiently negative and that a turbine trip and automatic auxiliary (or emergency) feedwater flow are initiated in a timely manner. As a result, the final ATWS rule (10 CFR 50.62) required that each pressurized water reactor have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. In the US, the plant equipment providing this capability has generally been referred to as the ATWS mitigation system actuation circuitry (AMSAC), which is described in Section 7.8. Therefore, the ATWS rule is met by the US-APWR.

The goal of the ATWS rule is to reduce the core damage frequency contribution from ATWS to less than 10^{-5} per reactor year (Ref 15.8-1). The US-APWR core damage frequency is discussed in Chapter 19 of this DCD and shows that the contribution of ATWS to the total core damage frequency meets the safety goal.

15.8.3 ATWS Design for the US-APWR

The features of the AMSAC described above have been incorporated into the US-APWR design as part of the diverse actuation system (DAS), which is described in Section 7.8. In addition to the AMSAC capability (diverse turbine trip and automatic emergency feedwater actuation), the US-APWR DAS also includes diverse reactor trip signals for the following trip functions that trip the motor-generator sets as diverse means of interrupting power to the reactor trip breakers in the event the ATWS is caused by a common mode failure of the reactor trip breakers:

- Low pressurizer pressure
- High pressurizer pressure
- Low steam generator level
- Manual reactor trip initiation

Therefore, the US-APWR includes the circuitry required by the ATWS rule.

15.8.4 Conclusions

The US-APWR has met the intent of the ATWS rule for the ATWS transient through a combination of the following measures:

- Meeting the core damage frequency safety goal associated with the ATWS Rule (10 CFR 50.62 and the Reference 15.8-1 basis) as described in the PRA

- Incorporating the AMSAC capability as defined in the ATWS Rule into the US-APWR DAS

- Incorporating a diverse reactor trip into the DAS on low pressurizer pressure, high pressurizer pressure, low steam generator level, and manual pushbutton inputs

- Incorporating a diverse means of interrupting power to the reactor trip breakers in the event the ATWS is caused by a common mode failure of the reactor trip breakers

15.8.5 Combined License Information

No additional information is required to be provided by a COL applicant in connection with this section.

15.8.6 References

15.8-1 Dircks, W. J., Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events, SECY-83-293, U.S. NRC, July 19, 1983.

APPENDIX 15A EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

This appendix contains the parameters and models that form the basis of the radiological consequences analyses for the various postulated accidents. The methodology is implemented using the RADTRAD computer code (Ref. 15A.5-1).

RADTRAD Computer Code

The RADionuclide Transport, Removal, and Dose (RADTRAD) computer code is a computer model for estimating doses at offsite locations such as the exclusion area boundary (EAB) and the low-population zone (LPZ), as well as onsite locations (e.g., main control room (MCR)) due to postulated radioactivity releases from design basis accident conditions. RADTRAD calculates dose consequences for different specified time intervals based on user-input information on the amount, form, and species of the radioactive material released in the reactor plant. For US-APWR, RADTRAD is configured to use the assumptions and source terms discussed in NUREG-1465, as specified in RG 1.183.

RADTRAD uses a compartment model and simulates radioactive material transport through the containment, and related systems, structures and components. The user can account for sprays and natural removal mechanisms that would reduce the quantity of radioactive material that is transported out of the reactor complex and to various specified offsite and onsite locations. Material can flow between buildings, from buildings to the environment, or into the MCR through high-efficiency particulate air (HEPA) filters, piping, or other connectors. The code tracks the amount of radioactive material retained due to these pathways. Decay and ingrowth of progeny radionuclides can be calculated over time as the material is transported. The code has numerous options and is very flexible depending on the specific accident sequence being modeled.

The code was developed and is maintained by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission.

15A.1 General Analysis Parameters

15A.1.1 Source Terms

15A.1.1.1 Reactor Coolant Source Term

The accident dose analyses assume the following reactor coolant source term.

Iodines: 1.0 μCi/g dose equivalent (DE) I-131 based on the Tech Spec limit (Table 15A-1)

Noble Gases: 300 μCi/g DE Xe-133 based on the Tech Spec limit (Table 15A-2)

Other radionuclides: Design basis radioactive concentrations (Table 11.1-2)

The reactor coolant source term is described in Table 15A-3.

For some events, the iodine concentrations in the reactor coolant are calculated using special assumptions that ensure the calculations account for conservatively large quantities of radioactive iodine by assuming: (1) a pre-transient iodine spike and (2) a transient initiated iodine spike.

> Pre-transient iodine spike: 60 μCi/g DE I-131 (Table 15A-4)
>
> Transient initiated iodine spikes: 500 times (for the steam system piping failure and failure of small lines carrying primary coolant outside containment events) or 335 time (for the steam generator tube rupture event) greater than the release rate corresponding to the iodine concentration at the equilibrium value (1 μCi/g DE I-131) specified in the Technical Specifications (Tables 15A-5 through 15A-7)

15A.1.1.2 Secondary Coolant Source Term

The iodine and alkali metal content of the secondary coolant source term is assumed to be 10% of the maximum equilibrium reactor coolant concentration. (Tables 15A-8 and 15A-9)

15A.1.1.3 Core Source Term

The time dependent fission product inventories in the reactor core are calculated by the ORIGEN2.2 code (Ref. 15A.5-2). For core inventory calculation, it is assumed that core has 2 regions. In core inventory calculation, irradiation time for a cycle is assumed 28months. (The planned cycle duration is 24 months.) In this calculation, the average specific power 32.1 megawatts thermal per metric ton of uranium (MW/MTU) is assumed. These calculation conditions lead fission and activation products generated in fuel with burnup of about 55 gigawatt days per metric ton of uranium (GWD/MTU) in 2 cycles. The core thermal power is 102% of design thermal power. Table 15A-10 lists the fission product inventories. Table 15A-10 also identifies the radionuclide group to which each of the nuclides considered in the analysis belongs.

The cladding on previously non-leaking fuel rods can become damaged during certain non-loss-of-coolant accident (LOCA) accidents involving fuel in the reactor core. Table 15A-11 lists the fraction of the fission product inventory assumed to be in the fuel rod gap (for various nuclides and groups of nuclides) for Non-LOCA accidents. Table 15A-12 lists the elements making up the various groups of fission product nuclides.

Table 15A-13 summarizes the fraction of fission products released into containment in the (1) gap and (2) early-in-vessel release phases of LOCA analyses performed for the US-APWR for specific fission product radionuclide groups. These values are intended to be applied to the core-average inventories. Table 15A-14 summarizes the time for onset and the duration of the LOCA release phases for which core inventory fractions are listed in Table 15A-13.

15A.1.1.4 Radioactive Concentration in Containment

For LOCA and rod ejection accidents, radionuclides are released within the containment. Radionuclides within the containment escape to the environment via leakages. The peak concentrations of radioactivity in the containment for LOCA and rod ejection accident are described in Tables 15A-15 and 15A-16.

15A.1.2 Airborne Radioactivity Removal Coefficients

The following material was taken from References 15A.5-3 and 15A.5-4, and is applicable to the US-APWR.

15A.1.2.1 Elemental iodine removal by wall deposition

The removal of elemental iodine by wall deposition can be estimated by the equation:

$$\lambda_w = K_w A/V$$

where:

λ_w = first-order removal coefficient by wall deposition
A = wetted-surface area
V = containment building net free volume
K_w = a mass-transfer coefficient (all available experimental data are conservatively bounded if K_w is taken to be 4.9 meters per hour)

For the US-APWR, A = 6230 m^2, V= 81000 m^3. Therefore, λ_w = 0.376 hr^{-1}.

The iodine decontamination factor (DF) is defined as the maximum iodine concentration in the containment atmosphere divided by the concentration of iodine in the containment atmosphere at some time after decontamination. The effectiveness of the wall deposition in removing elemental iodine is presumed to end when the maximum elemental iodine DF is reached. DF for the containment atmosphere achieved by the wall deposition is time dependent and is determined based on NUREG-0800, SRP 6.5.2 (Ref. 15A.5-3). The DF cannot exceed 200.

15A.1.2.2 Particulate removal

The first-order removal coefficient for particulates, λp, can be determined by the method described in Reference 15A.5-3, or estimated by:

$$\lambda_p = \frac{3hFE}{2VD}$$

where:

h = spray drop fall height
V = containment building net free volume
F = spray flow
E/D = ratio of a dimensionless collection efficiency E to the average spray drop diameter D. Since the removal of particulate material chiefly depends on the relative sizes of the particles and the spray drops, it is convenient to combine parameters that cannot be known. It is conservative to assume E/D to be 10 per meter initially (i.e., 1% efficiency for spray drops of 1 millimeter in diameter), changing abruptly to 1 spray drop per meter after the aerosol mass has been depleted by a

factor of 50 (i.e., 98% of the suspended mass is 10 times more readily removed than the remaining 2%).

Removal of particulate iodine by natural deposition is determined based on the Powers model (10th percentile), as shown in NUREG/CR-6189 (Ref. 15A.5-4).

15A.1.3 Flash Fraction

In cases where the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne is assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, is determined using a constant enthalpy, h, process based on the maximum time-dependent temperature of the refueling water storage pit water circulating outside the containment:

$$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$$

where h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.

In cases where the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne is assumed to be 10% of the total iodine activity in the leaked fluid.

15A.1.4 Nuclide Parameters

The exposure-to-committed effective dose equivalent (CEDE) dose conversion factors used in the US-APWR analysis are taken from Table 2.1 of "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11 (Ref. 15A.5-5). Table 15A-10 lists the dose conversion factors for calculation of the CEDE doses.

The receptor breathing rates are assumed to be 3.5×10^{-4}, 1.8×10^{-4}, and 2.3×10^{-4} m^3/s for the first 8 hours after the accident, the period from 8 to 24 hours after the accident, and the period from 24 hours until the end of the accident, respectively. Table 15.0.13 lists the receptor breathing rates. These receptor breathing rates are consistent with information in Section 4.1.3 of RG1.183.

The analysis discussed in the sections for Chapter 15 uses Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 15A.5-6), for external effective dose equivalent (EDE) conversion factors. Table 15A-10 lists the dose conversion factors for calculation of the EDE doses.

15A.1.5 Atmospheric Dispersion Factors

The offsite atmospheric dispersion factors (χ/Q values) are determined by representative values at the EAB and LPZ distance selected from the χ/Q values of a reasonable number of the existing sites. The χ/Q values used in determining the radiological consequences of postulated accidents are listed in Table 15A-17.

The MCR χ/Q values for the different required time intervals are listed by design basis accident event in Tables 15A-18 through 15A-23. The locations of the potential release points and their relationship to the MCR air intake and inleak are shown in Figure 15A-1.

15A.2 Dose Computation Model

RADTRAD code can track the transport of radionuclides as they are released from the reactor vessel, travel through the containment and other buildings, and are released to the environment. Leakage path of each accident is modeled as follows.

15A.2.1 LOCA Dose Computation Model

Following a postulated LOCA, the release of radioactivity from the containment to the environment with containment spray and engineered safety features (ESF) systems in full operation. The release due to the containment leakage and the recirculation water leakage from ESF systems is considered. The release from the containment takes into account a two-region spray model within the containment. The release from ESF systems pass filters of annulus emergency exhaust systems. Figure 15A-2 and Table 15A-24 show the leakage path and released activity during LOCA, respectively. For this accident, plant vent and containment releases to the environment are used.

15A.2.2 Steam System Piping Failure or Steam Generator Tube Rupture Dose Computation Model

Radioactivity leakage path to the environment due to steam system piping failure or steam generator tube rupture (SGTR) is direct and unfiltered. The activity release calculations for these accidents are complex involving spiking effects, time dependent flashing fractions, scrubbing of activities. Figure 15A-3 and Tables 15A-25 through 15A-28 show the leakage path and released activity during steam system piping failure or SGTR, respectively. For each accident, ground level release to the environment is used.

15A.2.3 Reactor Coolant Pump Rotor Seizure Dose Computation Model

Radioactivity leakage path to the environment due to reactor coolant pump (RCP) rotor seizure accident is similar to steam system piping failure or SGTR leakage path. The activity release calculations for this accident do not involve spiking effects which are considered in steam system piping failure or SGTR release calculations. Figure 15A-4 and Table 15A-29 show the leakage path and released activity during RCP rotor seizure event, respectively. For this accident, ground level release to the environment is used.

15A.2.4 Rod Ejection Accident Dose Computation Model

Radioactivity leakage paths to the environment due to rod ejection accident consist of the containment leakage and the secondary side leakages. The activity leakage paths for this accident are described in combination with the containment leakage path and the secondary leakage path. Figure 15A-5 and Table 15A-30 show the leakage paths and released activity during rod ejection accident, respectively. For this accident, plant vent and containment releases to the environment are used.

15A.2.5 Fuel Handling Accident Dose Computation Model

Radioactivity leakage path to the environment due to fuel handling accident is comparatively simple. The activity leakage path for this accident is from the fuel handling area or the containment. Figure 15A-6 and Table 15A-31 show the leakage path and released activity during fuel handling accidents, respectively. For this accident, ground level release to the environment is used.

15A.2.6 Failure of Small Lines Carrying Primary Coolant Outside Containment Dose Computation Model

Radioactivity leakage path to the environment due to failure of small lines carrying primary coolant outside containment is similar to fuel handling accident inside the containment. The activity leakage path for this accident is from the auxiliary building. Figure 15A-7 and Table 15A-32 show the leakage path and released activity during failure of small lines carrying primary coolant outside containment, respectively. For this accident, ground level release to the environment is used.

15A.3 Offsite Dose Calculation

15A.3.1 Immersion Dose (EDE)

For the EAB and LPZ, the EDE is determined by summing (for each type of airborne nuclide released from the reactor plant that reaches the postulated receptor) the product of: (1) the activity of the radioactive nuclide that is released; (2) the exposure-to-EDE ratio; and (3) the χ/Q value.

$$D_{im} = \sum_i DCF_i \sum_j R_{ij}(\chi/Q)$$

where:

D_{im}	=	Immersion (EDE) dose (rem)
DCF_i	=	EDE dose conversion factor for isotope i (rem-m^3/Ci-s)
R_{ij}	=	Amount of isotope i released during time period j (Ci)
$(\chi/Q)_j$	=	Atmospheric dispersion factor during time period j (s/m^3)

15A.3.2 Inhalation Dose (CEDE)

For the EAB and LPZ, the CEDE is determined by summing (for each type of airborne nuclide released from the reactor plant that reaches the postulated receptor) the product of: (1) the activity of the radioactive nuclide that is released; (2) the exposure-to-CEDE ratio; (3) the receptor's breathing rate; and (4) the χ/Q value.

$$D_{CEDE} = \sum_i DCF_i \sum_j R_{ij}\,(BR)_j\,(\chi/Q)_j$$

where:

D_{CEDE}	=	CEDE dose (rem)
DCF_i	=	CEDE dose conversion factor (rem per curie inhaled) for isotope i
R_{ij}	=	Amount of isotope i released during time period j (Ci)

$(BR)_j$ = Breathing rate during time period j (m^3/s)
$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

15A.3.3 Total Dose (TEDE)

The total effective dose equivalent (TEDE) doses are the sum of the EDE and the CEDE doses.

15A.4 Main Control Room Dose Calculation

15A.4.1 Immersion Dose (EDE)

Main control room doses are calculated using dose conversion factors identified in Regulatory Position 4.1 of RG1.183. The EDE from photons is corrected for the difference between finite cloud geometry in the MCR and the semi-infinite cloud assumption used in calculating the dose conversion factors.

$$D_{im} = \frac{1}{GF} \sum_i DCF_i \sum_j (IAR)_{ij} O_j$$

where:

D_{im} = Immersion (EDE) dose (rem)
GF = MCR geometry factor
= $1173/V^{0.338}$
V = Volume of the MCR (ft^3)
DCF_i = EDE dose conversion factor for isotope i ($rem\text{-}m^3/Ci\text{-}s$)
$(IAR)_{ij}$ = Integrated activity for isotope i in the MCR during time period j ($Ci\text{-}s/m^3$)
O_j = Fraction of time period j that the operator is assumed to be present

15A.4.2 Inhalation Dose (CEDE)

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j (IAR)_{ij} (BR)_j O_j$$

where:

D_{CEDE} = CEDE dose (rem)
DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i
$(IAR)_{ij}$ = Integrated activity for isotope i in the MCR during time period j ($Ci\text{-}s/m^3$)
$(BR)_j$ = Breathing rate during time period j (m^3/s)
O_j = Fraction of time period j that the operator is assumed to be present

15A.4.3 Total Dose (TEDE)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.5 References

15A.5-1 S.L. Humphreys et al., RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose estimation, NUREG/CR-6604, April 1998.

15A.5-2 ORIGEN 2.2 Isotope Generation and Depletion Code - Matrix Exponential Method, RSIC Computer Code Collection CCC-371, June, 2002.

15A.5-3. Containment Spray as a Fission Product Cleanup System, U.S. Nuclear Regulatory Commission, NUREG-0800 (Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants), Chapter 6.5.2 Revision 4, March 2007.

15A.5-4. D.A Powers and S.B. Burson, A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments, NUREG/CR-6189, July 1996.

15A.5-5. Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, EPA Federal Guidance Report No. 11, EPA-520/1-88-020, September 1988.

15A.5-6. External Exposure to Radionuclides in Air, Water, and Soil, EPA Federal Guidance Report No. 12, EPA 402-R-93-081, September 1993.

Table 15A-1
Reactor Coolant Iodine Concentrations
for 1.0 µCi/g DE I-131

Nuclide	µCi/g
I-131	7.37×10^{-1}
I-132	3.95×10^{-1}
I-133	1.27×10^{0}
I-134	2.71×10^{-1}
I-135	8.34×10^{-1}

Table 15A-2
Reactor Coolant Noble Gases Concentrations
for 300 µCi/g DE Xe-133

Nuclide	µCi/g
Kr-85	4.22×10^{1}
Kr-85m	8.15×10^{-1}
Kr-87	5.30×10^{-1}
Kr-88	1.52×10^{0}
Xe-133	1.43×10^{2}
Xe-135	4.69×10^{0}

Table 15A-3
Reactor Coolant Source Term (Sheet 1 of 2)

Nuclide	Half Life	Inventory (Ci)
Noble Gases		
Kr-85	10.72y	1.24×10^{4}
Kr-85 m	4.48h	2.39×10^{2}
Kr-87	76.3m	1.55×10^{2}
Kr-88	2.84h	4.47×10^{2}
Xe-133	5.245d	4.20×10^{4}
Xe-135	9.09h	1.38×10^{3}
Iodines		
I-131	8.04d	2.16×10^{2}
I-132	2.30h	1.16×10^{2}
I-133	20.8h	3.73×10^{2}
I-134	52.6m	7.95×10^{1}
I-135	6.61h	2.44×10^{2}
Alkali Metals		
Rb-86	18.66d	2.25×10^{0}
Cs-134	2.062y	2.29×10^{2}
Cs-136	13.1d	6.06×10^{1}
Cs-137	30.0y	1.31×10^{2}
Tellurium Group		
Sb-129	4.32h	8.84×10^{-3}
Te-127	9.35h	2.72×10^{0}
Te-127m	109d	5.18×10^{-1}
Te-129	69.6m	2.20×10^{0}
Te-129m	33.6d	1.77×10^{0}
Te-131m	30h	4.68×10^{0}
Te-132	78.2h	5.11×10^{1}
Strontium and Barium		
Sr-89	50.5d	5.68×10^{-1}
Sr-90	29.12y	3.69×10^{-2}
Sr-91	9.5h	3.82×10^{-1}
Sr-92	2.71h	2.13×10^{-1}
Ba-140	12.74d.	6.96×10^{-1}

Table 15A-3
Reactor Coolant Source Term (Sheet 2 of 2)

Nuclide	Half Life	Inventory (Ci)
Noble Metals		
Co-58	70.80d	1.82×10^{0}
Co-60	5.271y	2.66×10^{-1}
Mo-99	66.0h	1.34×10^{2}
Tc-99m	6.02h	5.40×10^{1}
Ru-103	39.28d	9.12×10^{-2}
Ru-106	368.2d	3.20×10^{-2}
Lanthanides		
Y-90	64.0h	8.42×10^{-3}
Y-91	58.51d	9.00×10^{-2}
Y-92	3.54h	1.66×10^{-1}
Y-93	10.1h	7.25×10^{-2}
Zr-95	63.98d	1.10×10^{-1}
Nb-95	35.15d	1.11×10^{-1}
La-140	40.272h	1.81×10^{-1}
La-141	3.93h	4.69×10^{-2}
Pr-143	13.56d	9.79×10^{-2}
Cerium Group		
Ce-141	32.501d	1.07×10^{-1}
Ce-143	33.0h	9.03×10^{-2}
Ce-144	284.3d	8.08×10^{-2}

Table 15A-4
Reactor Coolant Iodine Concentrations for Maximum Iodine Spike of 60 μCi/g DE I-131

Nuclide	μCi/g
I-131	4.42×10^{1}
I-132	2.37×10^{1}
I-133	7.63×10^{1}
I-134	1.63×10^{1}
I-135	5.00×10^{1}

Table 15A-5
Iodine Appearance Rates in the Reactor Coolant
(Steam System Piping Failure)

Nuclide	500 x Equilibrium Appearance Rate (Ci/min)
I-131	2.67×10^2
I-132	1.43×10^2
I-133	4.60×10^2
I-134	9.80×10^1
I-135	3.01×10^2

Table 15A-6
Iodine Appearance Rates in the Reactor Coolant
(Failure of Small Lines Carrying Primary Coolant Outside Containment)

Nuclide	500 x Equilibrium Appearance Rate (Ci/min)
I-131	2.79×10^2
I-132	1.50×10^2
I-133	4.82×10^2
I-134	1.03×10^2
I-135	3.16×10^2

Table 15A-7
Iodine Appearance Rates in the Reactor Coolant
(SGTR)

Nuclide	335 x Equilibrium Appearance Rate (Ci/min)
I-131	2.22×10^2
I-132	1.19×10^2
I-133	3.82×10^2
I-134	8.15×10^1
I-135	2.51×10^2

Table 15A-8
Secondary Coolant Source Term
(SGTR, Rod Ejection Accident and RCP Rotor Seizure)

Nuclide	Inventory (Ci)
Iodines	
I-131	1.52×10^{1}
I-132	8.18×10^{0}
I-133	2.63×10^{1}
I-134	5.61×10^{0}
I-135	1.72×10^{1}
Alkali Metals	
Rb-86	1.59×10^{-1}
Cs-134	1.62×10^{1}
Cs-136	4.28×10^{0}
Cs-137	9.22×10^{0}

Table 15A-9
Secondary Coolant Source Term
(Steam System Piping Failure)

Nuclide	Inventory (Ci)
Iodines	
I-131	3.24×10^{1}
I-132	1.74×10^{1}
I-133	5.58×10^{1}
I-134	1.19×10^{1}
I-135	3.66×10^{1}
Alkali Metals	
Rb-86	3.38×10^{-1}
Cs-134	3.44×10^{1}
Cs-136	9.08×10^{0}
Cs-137	1.96×10^{1}

Table 15A-10
Reactor Fission Product Nuclide Inventory and Related Parameters
(Sheet 1 of 4)

Nuclide	Half Life	Inventory (Ci) *1	CEDE Dose Conversion Factor *2 (Sv/Bq)	EDE Dose Conversion Factor *2 (Sv - m^3/ Bq - s)
Noble Gases				
Kr-85	10.72y	$1.73x10^{6}$	—	$1.19x10^{-16}$
Kr-85 m	4.48h	$4.83x10^{7}$	—	$7.48x10^{-15}$
Kr-87	76.3m	$9.59x10^{7}$	—	$4.12x10^{-14}$
Kr-88	2.84h	$1.35x10^{8}$	—	$1.02x10^{-13}$
Xe-133	5.245d	$2.99x10^{8}$	—	$1.56x10^{-15}$
Xe-135	9.09h	$9.14x10^{7}$	—	$1.19x10^{-14}$
Iodines				
I-131	8.04d	$1.44x10^{8}$	$8.89x10^{-9}$	$1.82x10^{-14}$
I-132	2.30h	$2.08x10^{8}$	$1.03x10^{-10}$	$1.12x10^{-13}$
I-133	20.8h	$3.00x10^{8}$	$1.58x10^{-9}$	$2.94x10^{-14}$
I-134	52.6m	$3.35x10^{8}$	$3.55x10^{-11}$	$1.30x10^{-13}$
I-135	6.61h	$2.80x10^{8}$	$3.32x10^{-10}$	$8.29x10^{-14}$

Notes:

*1 These inventories are assumed to be 32.1 MW/MTU of specific power, 28 months as irradiation time for a cycle (these conditions are equivalent to about 55 GWD/MTU.), and 102% of the design core thermal power.

*2 These conversion factors are listed in NUREG/CR-6604 (Ref. 15A.5-1). NUREG/CR-6604 is based on Federal Guidance Report 11 (Ref.15A.5-5) and 12 (Ref.15A.5-6).

Table 15A-10
Reactor Fission Product Nuclide Inventory and Related Parameters
(Sheet 2 of 4)

Nuclide	Half Life	Inventory (Ci) *1	CEDE Dose Conversion Factor *2 (Sv/Bq)	EDE Dose Conversion Factor *2 (Sv - m^3/ Bq - s)
Alkali Metals				
Rb-86	18.66d	$3.40x10^{5}$	$1.79x10^{-9}$	$4.81x10^{-15}$
Cs-134	2.062y	$3.39x10^{7}$	$1.25x10^{-8}$	$7.57x10^{-14}$
Cs-136	13.1d	$9.23x10^{6}$	$1.98x10^{-9}$	$1.06x10^{-13}$
Cs-137	30.0y	$1.93x10^{7}$	$8.63x10^{-9}$	$2.73x10^{-14}$
Tellurium Group				
Sb-127	3.85d	$1.50x10^{7}$	$1.63x10^{-9}$	$3.33x10^{-14}$
Sb-129	4.32h	$4.54x10^{7}$	$1.74x10^{-10}$	$7.14x10^{-14}$
Te-127	9.35h	$1.48x10^{7}$	$8.60x10^{-11}$	$2.42x10^{-16}$
Te-127m	109d	$1.95x10^{6}$	$5.81x10^{-9}$	$1.47x10^{-16}$
Te-129	69.6m	$4.47x10^{7}$	$2.09x10^{-11}$	$2.75x10^{-15}$
Te-129m	33.6d	$6.69x10^{6}$	$6.48x10^{-9}$	$3.34x10^{-15}$
Te-131m	30h	$2.06x10^{7}$	$1.76x10^{-9}$	$7.46x10^{-14}$
Te-132	78.2h	$2.05x10^{8}$	$2.55x10^{-9}$	$1.03x10^{-14}$
Strontium and Barium				
Sr-89	50.5d	$1.67x10^{8}$	$1.12x10^{-8}$	$7.73x10^{-17}$
Sr-90	29.12y	$1.39x10^{7}$	$3.51x10^{-7}$	$7.53x10^{-18}$
Sr-91	9.5h	$2.21x10^{8}$	$4.55x10^{-10}$	$4.92x10^{-14}$
Sr-92	2.71h	$2.32x10^{8}$	$2.18x10^{-10}$	$6.79x10^{-14}$
Ba-139	82.7m	$2.78x10^{8}$	$4.64x10^{-11}$	$2.17x10^{-15}$
Ba-140	12.74d.	$2.66x10^{8}$	$1.01x10^{-9}$	$8.58x10^{-15}$

Notes:

*1 These inventories are assumed to be 32.1 MW/MTU of specific power, 28 months as irradiation time for a cycle (these conditions are equivalent to about 55 GWD/MTU.), and 102% of the design core thermal power.

*2 These conversion factors are listed in NUREG/CR-6604 (Ref. 15A.5-1). NUREG/CR-6604 is based on Federal Guidance Report 11 (Ref.15A.5-5) and 12 (Ref.15A.5-6).

Table 15A-10
Reactor Fission Product Nuclide Inventory and Related Parameters
(Sheet 3 of 4)

Nuclide	Half Life	Inventory (Ci) *1	CEDE Dose Conversion Factor *2 (Sv/Bq)	EDE Dose Conversion Factor *2 (Sv - m^3/ Bq - s)
Noble Metals				
Co-58	70.80d	$0.00x10^{0}$	$2.94x10^{-9}$	$4.76x10^{-14}$
Co-60	5.271y	$4.35x10^{5}$	$5.91x10^{-8}$	$1.26x10^{-13}$
Mo-99	66.0h	$2.72x10^{8}$	$1.07x10^{-9}$	$7.28x10^{-15}$
Tc-99m	6.02h	$2.38x10^{8}$	$8.80x10^{-12}$	$5.89x10^{-15}$
Ru-103	39.28d	$2.15x10^{8}$	$2.42x10^{-9}$	$2.25x10^{-14}$
Ru-105	4.44h	$1.41x10^{8}$	$1.23x10^{-10}$	$3.81x10^{-14}$
Ru-106	368.2d	$7.53x10^{7}$	$1.29x10^{-7}$	$1.04x10^{-14}$
Rh-105	35.36h	$1.31x10^{8}$	$2.58x10^{-10}$	$3.72x10^{-15}$
Lanthanides				
Y-90	64.0h	$1.46x10^{7}$	$2.28x10^{-9}$	$1.90x10^{-16}$
Y-91	58.51d	$2.09x10^{8}$	$1.32x10^{-8}$	$2.60x10^{-16}$
Y-92	3.54h	$2.32x10^{8}$	$2.11x10^{-10}$	$1.30x10^{-14}$
Y-93	10.1h	$2.57x10^{8}$	$5.82x10^{-10}$	$4.80x10^{-15}$
Zr-95	63.98d	$2.66x10^{8}$	$6.39x10^{-9}$	$3.60x10^{-14}$
Zr-97	16.90h	$2.66x10^{8}$	$1.17x10^{-9}$	$4.43x10^{-14}$
Nb-95	35.15d	$2.68x10^{8}$	$1.57x10^{-9}$	$3.74x10^{-14}$
La-140	40.272h	$2.70x10^{8}$	$1.31x10^{-9}$	$1.17x10^{-13}$
La-141	3.93h	$2.54x10^{8}$	$1.57x10^{-10}$	$2.39x10^{-15}$
La-142	92.5m	$2.50x10^{8}$	$6.84x10^{-11}$	$1.44x10^{-13}$
Pr-143	13.56d	$2.37x10^{8}$	$2.19x10^{-9}$	$2.10x10^{-17}$
Nd-147	10.98d	$9.94x10^{7}$	$1.85x10^{-9}$	$6.19x10^{-15}$
Am-241	432.2y	$2.64x10^{4}$	$1.20x10^{-4}$	$8.18x10^{-16}$
Cm-242	162.8d	$6.55x10^{6}$	$4.67x10^{-6}$	$5.69x10^{-18}$
Cm-244	18.11y	$7.96x10^{5}$	$6.70x10^{-5}$	$4.91x10^{-18}$

Notes:

*1 These inventories are assumed to be 32.1 MW/MTU of specific power, 28 months as irradiation time for a cycle (these conditions are equivalent to about 55 GWD/MTU.), and 102% of the design core thermal power.

*2 These conversion factors are listed in NUREG/CR-6604 (Ref. 15A.5-1). NUREG/CR-6604 is based on Federal Guidance Report 11 (Ref.15A.5-5) and 12 (Ref.15A.5-6).

Table 15A-10
Reactor Fission Product Nuclide Inventory and Related Parameters
(Sheet 4 of 4)

Nuclide	Half Life	Inventory (Ci) *1	CEDE Dose Conversion Factor *2 (Sv/Bq)	EDE Dose Conversion Factor *2 ($Sv - m^3/ Bq - s$)
Cerium Group				
Ce-141	32.501d	$2.51x10^{8}$	$2.42x10^{-9}$	$3.43x10^{-15}$
Ce-143	33.0h	$2.45x10^{8}$	$9.16x10^{-10}$	$1.29x10^{-14}$
Ce-144	284.3d	$1.90x10^{8}$	$1.01x10^{-7}$	$2.77x10^{-15}$
Np-239	2.355d	$2.71x10^{9}$	$6.78x10^{-10}$	$7.69x10^{-15}$
Pu-238	87.74y	$7.47x10^{5}$	$7.79x10^{-5}$	$4.88x10^{-18}$
Pu-239	24065y	$5.64x10^{4}$	$8.33x10^{-5}$	$4.24x10^{-18}$
Pu-240	6537y	$8.85x10^{4}$	$8.33x10^{-5}$	$4.75x10^{-18}$
Pu-241	14.4y	$1.96x10^{7}$	$1.34x10^{-6}$	$7.25x10^{-20}$

Notes:

*1 These inventories are assumed to be 32.1 MW/MTU of specific power, 28 months as irradiation time for a cycle (these conditions are equivalent to about 55 GWD/MTU.), and 102% of the design core thermal power.

*2 These conversion factors are listed in NUREG/CR-6604 (Ref. 15A.5-1). NUREG/CR-6604 is based on Federal Guidance Report 11 (Ref.15A.5-5) and 12 (Ref.15A.5-6).

Table 15A-11
Fraction of Fission Product Inventory in Fuel Rod Gap Used in Non-LOCA Radiological Consequence Evaluations

Nuclide or Nuclide Group	Fraction of Inventory in Gap	
	Non-LOCA	Rod ejection accident
I-131	0.08	0.10
Kr-85	0.10	0.10
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.12
All Other Nuclides	0.00	0.00

Table 15A-12
Radionuclide Groups

Radionuclide Group	Elements
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se, Ba, Sr
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium Group	Ce, Pu, Np

Table 15A-13
Fraction of Fission Product Inventory for Fuel Rod Gap and Early In-Vessel Release Phases Used in LOCA Radiological Consequence Evaluations

Nuclide Group	Inventory Fraction in Each Release Phase	
	Gap Release	Early In-Vessel Release
Noble Gases	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble Metals	0.00	0.0025
Cerium Group	0.00	0.0005
Lanthanides	0.00	0.0002

Table 15A-14
Time of Onset and Duration for Fuel Rod Gap and Early In-Vessel Release Phases Used in LOCA Radiological Consequence Evaluations

Phase	Onset (time after initiation of accident)	Phase Duration
Gap Release	30 sec	0.5 hr
Early In-Vessel Release	0.5 hr	1.3 hr

Table 15A-15
The Peak Concentration [*1] in Containment During LOCA (Sheet 1 of 2)

Nuclide	Half Life	Activity (Ci)
Noble Gases		
Kr-85	10.72y	$1.73x10^{6}$
Kr-85m	4.48h	$3.08x10^{7}$
Kr-87	76.3m	$2.00x10^{7}$
Kr-88	2.84h	$6.67x10^{7}$
Xe-133	5.245d	$2.98x10^{8}$
Xe-135	9.09h	$1.06x10^{8}$
Iodines		
I-131	8.04d	$1.62x10^{7}$
I-132	2.30h	$2.09x10^{7}$
I-133	20.8h	$3.08x10^{7}$
I-134	52.6m	$3.24x10^{6}$
I-135	6.61h	$2.29x10^{7}$
Alkali Metals		
Rb-86	18.66d	$2.54x10^{4}$
Cs-134	2.062y	$2.54x10^{6}$
Cs-136	13.1d	$6.87x10^{5}$
Cs-137	30.0y	$1.44x10^{6}$
Tellurium Group		
Sb-127	3.85d	$2.12x10^{5}$
Sb-129	4.32h	$3.93x10^{5}$
Te-127	9.35h	$2.12x10^{5}$
Te-127m	109d	$2.83x10^{4}$
Te-129	69.6m	$4.61x10^{5}$
Te-129m	33.6d	$9.69x10^{4}$
Te-131m	30h	$2.77x10^{5}$
Te-132	78.2h	$2.88x10^{6}$
Strontium and Barium		
Sr-89	50.5d	$9.68x10^{5}$
Sr-90	29.12y	$8.03x10^{4}$
Sr-91	9.5h	$1.01x10^{6}$
Sr-92	2.71h	$5.92x10^{5}$
Ba-139	82.7m	$3.24x10^{5}$
Ba-140	12.74d.	$1.53x10^{6}$

Note:
*1 The peak concentration is at 1.8 hour after accident.

Table 15A-15
The Peak Concentration [*1] in Containment During LOCA (Sheet 2 of 2)

Nuclide	Half Life	Activity (Ci)
Noble Metals		
Co-58	70.80d	0.00×10^{0}
Co-60	5.271y	3.15×10^{2}
Mo-99	66.0h	1.90×10^{5}
Tc-99m	6.02h	1.72×10^{5}
Ru-103	39.28d	1.56×10^{5}
Ru-105	4.44h	6.21×10^{4}
Ru-106	368.2d	5.45×10^{4}
Rh-105	35.36h	9.43×10^{4}
Lanthanides		
Y-90	64.0h	1.19×10^{3}
Y-91	58.51d	1.22×10^{4}
Y-92	3.54h	6.15×10^{4}
Y-93	10.1h	1.19×10^{4}
Zr-95	63.98d	1.54×10^{4}
Zr-97	16.90h	1.35×10^{4}
Nb-95	35.15d	1.55×10^{4}
La-140	40.272h	2.60×10^{4}
La-141	3.93h	8.35×10^{3}
La-142	92.5m	3.44×10^{3}
Pr-143	13.56d	1.37×10^{4}
Nd-147	10.98d	5.71×10^{3}
Am-241	432.2y	1.53×10^{0}
Cm-242	162.8d	3.79×10^{2}
Cm-244	18.11y	4.61×10^{1}
Cerium Group		
Ce-141	32.501d	3.64×10^{4}
Ce-143	33.0h	3.32×10^{4}
Ce-144	284.3d	2.75×10^{4}
Np-239	2.355d	3.77×10^{5}
Pu-238	87.74y	1.08×10^{2}
Pu-239	24065y	8.17×10^{0}
Pu-240	6537y	1.28×10^{1}
Pu-241	14.1y	2.83×10^{3}

Note:
*1 The peak concentration is at 1.8 hour after accident.

Table 15A-16
The Peak Concentration [*1] in Containment During Rod Ejection Accident

Nuclide	Half Life	Activity (Ci)
Noble Gases		
Kr-85	10.72y	8.49×10^4
Kr-85m	4.48h	2.02×10^6
Kr-87	76.3m	4.01×10^6
Kr-88	2.84h	5.66×10^6
Xe-133	5.245d	1.25×10^7
Xe-135	9.09h	3.83×10^6
Iodines		
I-131	8.04d	5.53×10^6
I-132	2.30h	8.00×10^6
I-133	20.8h	1.16×10^7
I-134	52.6m	1.29×10^7
I-135	6.61h	1.08×10^7
Alkali Metals		
Rb-86	18.66d	7.65×10^3
Cs-134	2.062y	7.62×10^5
Cs-136	13.1d	2.07×10^5
Cs-137	30.0y	4.33×10^5

Note:
*1 The peak concentration is at 0 hour after accident.

Table 15A-17
Offsite χ/Q for Accident Dose Analysis

EAB χ/Q (s/m^3)		
χ/Q	0-2 hr *1	5.0×10^{-4}
LPZ outer boundary χ/Q (s/m^3)		
χ/Q	0-8 hr	2.1×10^{-4}
	8-24 hr	1.3×10^{-4}
	24-96 hr	6.9×10^{-5}
	96-720 hr	2.8×10^{-5}

Note:
*1 Nominally defined as the 0- to 2-hour interval but is applied to the 2-hour interval having the highest activity releases in order to address 10 CFR 50.34 requirements.

Table 15A-18
Main Control Room χ/Q for Steam System Piping Failure Analysis

Accidents		Steam system piping failure			
Sources		Steam line break releases		Main steam relief valve and safety valve releases	
Receptors		Intake	Inleak	Intake	Inleak
		MCR heating, ventilation, and air conditioning (HVAC) intake	MCR HVAC intake	Class 1E electrical room HVAC intake	Class 1E electrical room HVAC intake
Horizontal Distance (m)		17	17	24	24
Vertical Distance (m)		0	0	22	22
χ/Q (s/m^3)	0-8 hr	1.9×10^{-2}	1.9×10^{-2}	5.3×10^{-3}	5.3×10^{-3}
	8-24 hr	1.1×10^{-2}	1.1×10^{-2}	3.1×10^{-3}	3.1×10^{-3}
	24-96 hr	7.1×10^{-3}	7.1×10^{-3}	2.0×10^{-3}	2.0×10^{-3}
	96-720 hr	3.1×10^{-3}	3.1×10^{-3}	8.7×10^{-4}	8.7×10^{-4}

Table 15A-19
Main Control Room χ/Q for RCP Rotor Seizure Analysis

Accidents		RCP rotor seizure accident	
Sources		Main steam relief valve and safety valve releases	
Receptors		Intake	Inleak
		Class 1E electrical room HVAC intake	Class 1E electrical room HVAC intake
Horizontal Distance (m)		24	24
Vertical Distance (m)		22	22
χ/Q (s/m^3)	0-8 hr	5.3×10^{-3}	5.3×10^{-3}
	8-24 hr	3.1×10^{-3}	3.1×10^{-3}
	24-96 hr	2.0×10^{-3}	2.0×10^{-3}
	96-720 hr	8.7×10^{-4}	8.7×10^{-4}

Table 15A-20
Main Control Room χ/Q for Rod Ejection Accident Analysis (Sheet 1 of 2)

Accidents		Rod ejection accident			
Sources		Plant vent		Ground level containment release point	
Receptors		Intake	Inleak	Intake	Inleak
		MCR HVAC intake	Auxiliary building HVAC intake	MCRHVAC intake	Class 1E electrical room HVAC intake
Horizontal Distance (m)		56	55	32	27
Vertical Distance (m)		52	43	32	33
χ/Q (s/m^3)	0-8 hr	1.1×10^{-3}	1.4×10^{-3}	2.2×10^{-3}	2.4×10^{-3}
	8-24 hr	6.6×10^{-4}	8.0×10^{-4}	1.3×10^{-3}	1.4×10^{-3}
	24-96 hr	4.2×10^{-4}	5.1×10^{-4}	8.3×10^{-4}	9.1×10^{-4}
	96-720 hr	1.9×10^{-4}	2.2×10^{-4}	3.6×10^{-4}	4.0×10^{-4}

Table 15A-20
Main Control Room χ/Q for Rod Ejection Accident Analysis (Sheet 2 of 2)

Accidents		Rod ejection accident	
Sources		Main steam relief valve and safety valve releases	
Receptors		Intake	Inleak
		Class 1E electrical room HVAC intake	Class 1E electrical room HVAC intake
Horizontal Distance (m)		24	24
Vertical Distance (m)		22	22
χ/Q (s/m^3)	0-8 hr	5.3×10^{-3}	5.3×10^{-3}
	8-24 hr	3.1×10^{-3}	3.1×10^{-3}
	24-96 hr	2.0×10^{-3}	2.0×10^{-3}
	96-720 hr	8.7×10^{-4}	8.7×10^{-4}

Table 15A-21
Main Control Room χ/Q for Failure of Small Lines Carrying Primary Coolant Outside Containment and SGTR Analyses

Accidents		Failure of small lines carrying primary coolant outside containment		SGTR	
Sources		Auxiliary building (RCS sample line)		Main steam relief valve and safety valve releases	
Receptors		Intake	Inleak	Intake	Inleak
		MCR HVAC intake	Reactor building door	Class 1E electrical room HVAC intake	Class 1E electrical room HVAC intake
Horizontal Distance (m)		52	34	24	24
Vertical Distance (m)		7	0	22	22
χ/Q (s/m^3)	0-8 hr	2.2×10^{-3}	4.9×10^{-3}	5.3×10^{-3}	5.3×10^{-3}
	8-24 hr	1.3×10^{-3}	2.9×10^{-3}	3.1×10^{-3}	3.1×10^{-3}
	24-96 hr	8.4×10^{-4}	1.8×10^{-3}	2.0×10^{-3}	2.0×10^{-3}
	96-720 hr	3.7×10^{-4}	8.1×10^{-4}	8.7×10^{-4}	8.7×10^{-4}

Table 15A-22
Main Control Room χ/Q for LOCA Analysis

Accidents		LOCA			
Sources		Plant vent		Ground level containment release point	
Receptors		Intake	Inleak	Intake	Inleak
		MCR HVAC intake	Reactor building door	MCR HVAC intake	Class 1E electrical room HVAC intake
Horizontal Distance (m)		56	37	32	27
Vertical Distance (m)		52	60	32	33
χ/Q (s/m^3)	0-8 hr	1.1×10^{-3}	1.3×10^{-3}	2.2×10^{-3}	2.4×10^{-3}
	8-24 hr	6.6×10^{-4}	7.7×10^{-4}	1.3×10^{-3}	1.4×10^{-3}
	24-96 hr	4.2×10^{-4}	4.9×10^{-4}	8.3×10^{-4}	9.1×10^{-4}
	96-720 hr	1.9×10^{-4}	2.2×10^{-4}	3.6×10^{-4}	4.0×10^{-4}

Table 15A-23
Main Control Room χ/Q for Fuel Handling Accident Analysis

Accidents		Fuel handling accident in the containment		Fuel handling accident in the fuel handling area	
Sources		Air lock of containment		Fuel handling area	
Receptors		Intake MCR HVAC intake	Inleak Class 1E electrical room HVAC intake	Intake MCR HVAC intake	Inleak Class 1E electrical room HVAC intake
Horizontal Distance (m)		35	29	82	76
Vertical Distance (m)		5.2	5.8	8.5	8.5
χ/Q (s/m^3)	0-8 hr	4.7×10^{-3}	6.4×10^{-3}	9.9×10^{-4}	1.1×10^{-3}
	8-24 hr	2.8×10^{-3}	3.8×10^{-3}	5.9×10^{-4}	6.7×10^{-4}
	24-96 hr	1.8×10^{-3}	2.4×10^{-3}	3.7×10^{-4}	4.3×10^{-4}
	96-720 hr	7.7×10^{-4}	1.1×10^{-3}	1.6×10^{-4}	1.9×10^{-4}

Table 15A-24
Time Dependent Released Activity during LOCA (Ci) (Sheet 1 of 2)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	$7.75x10^{2}$	$1.74x10^{3}$	$3.92x10^{3}$	$3.35x10^{4}$	$3.99x10^{4}$
Kr-85m	$9.16x10^{3}$	$4.37x10^{3}$	$1.99x10^{2}$	$0.00x10^{0}$	$1.37x10^{4}$
Kr-87	$3.54x10^{3}$	$7.83x10^{1}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.62x10^{3}$
Kr-88	$1.68x10^{4}$	$3.68x10^{3}$	$3.70x10^{1}$	$0.00x10^{0}$	$2.05x10^{4}$
Xe-133	$1.26x10^{5}$	$2.76x10^{5}$	$4.93x10^{5}$	$9.77x10^{5}$	$1.87x10^{6}$
Xe-135	$3.79x10^{4}$	$4.05x10^{4}$	$9.60x10^{3}$	$4.41x10^{1}$	$8.80x10^{4}$
Iodines					
I-131	$1.42x10^{3}$	$5.61x10^{2}$	$1.85x10^{3}$	$5.60x10^{3}$	$9.43x10^{3}$
I-132	$1.50x10^{3}$	$1.01x10^{2}$	$2.22x10^{2}$	$2.48x10^{2}$	$2.07x10^{3}$
I-133	$2.67x10^{3}$	$7.37x10^{2}$	$8.09x10^{2}$	$8.07x10^{1}$	$4.30x10^{3}$
I-134	$4.22x10^{2}$	$1.84x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.22x10^{2}$
I-135	$1.95x10^{3}$	$2.44x10^{2}$	$4.67x10^{1}$	$1.20x10^{-1}$	$2.24x10^{3}$
Alkali Metals					
Rb-86	$1.44x10^{0}$	$1.60x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.45x10^{0}$
Cs-134	$1.44x10^{2}$	$1.62x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.46x10^{2}$
Cs-136	$3.90x10^{1}$	$4.31x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.94x10^{1}$
Cs-137	$8.19x10^{1}$	$9.21x10^{-1}$	$1.00x10^{-3}$	$0.00x10^{0}$	$8.28x10^{1}$
Tellurium Group					
Sb-127	$1.04x10^{1}$	$1.26x10^{-1}$	$1.00x10^{-5}$	$0.00x10^{0}$	$1.05x10^{1}$
Sb-129	$1.99x10^{1}$	$6.87x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.00x10^{1}$
Te-127	$1.04x10^{1}$	$1.30x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.05x10^{1}$
Te-127m	$1.39x10^{0}$	$1.80x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.40x10^{0}$
Te-129	$2.30x10^{1}$	$1.12x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.31x10^{1}$
Te-129m	$4.75x10^{0}$	$6.13x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.81x10^{0}$
Te-131m	$1.36x10^{1}$	$1.44x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.37x10^{1}$
Te-132	$1.41x10^{2}$	$1.71x10^{0}$	$1.00x10^{-4}$	$0.00x10^{0}$	$1.43x10^{2}$
Strontium and Barium					
Sr-89	$4.74x10^{1}$	$6.12x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.80x10^{1}$
Sr-90	$3.93x10^{0}$	$5.10x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.98x10^{0}$
Sr-91	$5.01x10^{1}$	$3.54x10^{-1}$	$1.00x10^{-3}$	$0.00x10^{0}$	$5.05x10^{1}$
Sr-92	$3.11x10^{1}$	$4.95x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.11x10^{1}$
Ba-139	$1.96x10^{1}$	$5.04x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.96x10^{1}$
Ba-140	$7.49x10^{1}$	$9.53x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$7.59x10^{1}$

Table 15A-24
Time Dependent Released Activity during LOCA (Ci) (Sheet 2 of 2)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Metals					
Co-58	$3.36x10^{-3}$	$4.50x10^{-8}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.36x10^{-3}$
Co-60	$1.59x10^{-2}$	$2.00x10^{-4}$	$1.01x10^{-6}$	$0.00x10^{0}$	$1.61x10^{-2}$
Mo-99	$9.57x10^{0}$	$1.11x10^{-1}$	$1.00x10^{-4}$	$0.00x10^{0}$	$9.68x10^{0}$
Tc-99m	$8.50x10^{0}$	$1.04x10^{-1}$	$1.00x10^{-4}$	$0.00x10^{0}$	$8.60x10^{0}$
Ru-103	$7.62x10^{0}$	$9.83x10^{-2}$	$1.01x10^{-4}$	$0.00x10^{0}$	$7.72x10^{0}$
Ru-105	$3.14x10^{0}$	$1.12x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.15x10^{0}$
Ru-106	$2.67x10^{0}$	$3.46x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.70x10^{0}$
Rh-105	$4.61x10^{0}$	$5.41x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.67x10^{0}$
Lanthanides					
Y-90	$7.44x10^{-2}$	$5.12x10^{-3}$	$6.06x10^{-6}$	$0.00x10^{0}$	$7.96x10^{-2}$
Y-91	$6.00x10^{-1}$	$8.54x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$6.09x10^{-1}$
Y-92	$4.13x10^{0}$	$1.04x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.24x10^{0}$
Y-93	$5.90x10^{-1}$	$4.32x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.94x10^{-1}$
Zr-95	$7.55x10^{-1}$	$9.76x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$7.65x10^{-1}$
Zr-97	$6.65x10^{-1}$	$6.12x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$6.71x10^{-1}$
Nb-95	$7.60x10^{-1}$	$9.85x10^{-3}$	$1.01x10^{-5}$	$0.00x10^{0}$	$7.69x10^{-1}$
La-140	$1.76x10^{0}$	$1.43x10^{-1}$	$2.02x10^{-4}$	$0.00x10^{0}$	$1.90x10^{0}$
La-141	$4.25x10^{-1}$	$1.29x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.27x10^{-1}$
La-142	$2.01x10^{-1}$	$7.07x10^{-5}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.01x10^{-1}$
Pr-143	$6.74x10^{-1}$	$8.91x10^{-3}$	$1.00x10^{-5}$	$0.00x10^{0}$	$6.83x10^{-1}$
Nd-147	$2.80x10^{-1}$	$3.55x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.83x10^{-1}$
Am-241	$7.51x10^{-5}$	$9.77x10^{-7}$	$0.00x10^{0}$	$0.00x10^{0}$	$7.60x10^{-5}$
Cm-242	$1.86x10^{-2}$	$2.41x10^{-4}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.88x10^{-2}$
Cm-244	$2.26x10^{-3}$	$2.93x10^{-5}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.29x10^{-3}$
Cerium Group					
Ce-141	$1.78x10^{0}$	$2.29x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.80x10^{0}$
Ce-143	$1.63x10^{0}$	$1.78x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.65x10^{0}$
Ce-144	$1.35x10^{0}$	$1.75x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.36x10^{0}$
Np-239	$1.85x10^{1}$	$2.16x10^{-1}$	$1.00x10^{-5}$	$0.00x10^{0}$	$1.87x10^{1}$
Pu-238	$5.30x10^{-3}$	$6.88x10^{-5}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.37x10^{-3}$
Pu-239	$4.00x10^{-4}$	$5.19x10^{-6}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.05x10^{-4}$
Pu-240	$6.28x10^{-4}$	$8.14x10^{-6}$	$1.01x10^{-8}$	$0.00x10^{0}$	$6.36x10^{-4}$
Pu-241	$1.39x10^{-1}$	$1.81x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.41x10^{-1}$
TOTAL	$2.03x10^{5}$	$3.28x10^{5}$	$5.09x10^{5}$	$1.02x10^{6}$	$2.06x10^{6}$

Table 15A-25
Time Dependent Released Activity during Steam System Piping failure (Ci)
(Transient-initiated Iodine Spike)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	3.21×10^{1}	2.40×10^{1}	0.00×10^{0}	0.00×10^{0}	5.61×10^{1}
Kr-85m	3.56×10^{-1}	8.77×10^{-2}	0.00×10^{0}	0.00×10^{0}	4.43×10^{-1}
Kr-87	9.12×10^{-2}	1.13×10^{-3}	0.00×10^{0}	0.00×10^{0}	9.23×10^{-2}
Kr-88	5.10×10^{-1}	6.46×10^{-2}	0.00×10^{0}	0.00×10^{0}	5.74×10^{-1}
Xe-133	1.08×10^{2}	8.03×10^{1}	0.00×10^{0}	0.00×10^{0}	1.88×10^{2}
Xe-135	7.61×10^{0}	1.33×10^{1}	0.00×10^{0}	0.00×10^{0}	2.09×10^{1}
Iodines					
I-131	5.05×10^{1}	6.50×10^{1}	0.00×10^{0}	0.00×10^{0}	1.16×10^{2}
I-132	9.89×10^{0}	1.49×10^{0}	0.00×10^{0}	0.00×10^{0}	1.14×10^{1}
I-133	7.65×10^{1}	8.09×10^{1}	0.00×10^{0}	0.00×10^{0}	1.57×10^{2}
I-134	3.77×10^{0}	9.11×10^{-3}	0.00×10^{0}	0.00×10^{0}	3.78×10^{0}
I-135	3.77×10^{1}	2.45×10^{1}	0.00×10^{0}	0.00×10^{0}	6.21×10^{1}
Alkali Metals					
Rb-86	8.64×10^{-2}	1.62×10^{-3}	0.00×10^{0}	0.00×10^{0}	8.80×10^{-2}
Cs-134	8.80×10^{0}	1.68×10^{-1}	0.00×10^{0}	0.00×10^{0}	8.97×10^{0}
Cs-136	2.32×10^{0}	4.33×10^{-2}	0.00×10^{0}	0.00×10^{0}	2.37×10^{0}
Cs-137	5.01×10^{0}	9.56×10^{-2}	0.00×10^{0}	0.00×10^{0}	5.11×10^{0}
TOTAL	3.43×10^{2}	2.90×10^{2}	0.00×10^{0}	0.00×10^{0}	6.33×10^{2}

Table 15A-26
Time Dependent Released Activity during Steam System Piping failure (Ci)
(Pre-transient Iodine Spike)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	$3.21x10^{1}$	$2.40x10^{1}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.61x10^{1}$
Kr-85m	$3.56x10^{-1}$	$8.77x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.43x10^{-1}$
Kr-87	$9.12x10^{-2}$	$1.13x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$9.23x10^{-2}$
Kr-88	$5.10x10^{-1}$	$6.46x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.74x10^{-1}$
Xe-133	$1.07x10^{2}$	$7.75x10^{1}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.85x10^{2}$
Xe-135	$4.38x10^{0}$	$3.39x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$7.78x10^{0}$
Iodines					
I-131	$1.72x10^{1}$	$7.25x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.44x10^{1}$
I-132	$6.18x10^{0}$	$1.66x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$6.35x10^{0}$
I-133	$2.79x10^{1}$	$9.03x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.69x10^{1}$
I-134	$3.49x10^{0}$	$1.01x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.49x10^{0}$
I-135	$1.62x10^{1}$	$2.73x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.89x10^{1}$
Alkali Metals					
Rb-86	$8.64x10^{-2}$	$1.62x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$8.80x10^{-2}$
Cs-134	$8.80x10^{0}$	$1.68x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$8.97x10^{0}$
Cs-136	$2.32x10^{0}$	$4.33x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.37x10^{0}$
Cs-137	$5.01x10^{0}$	$9.56x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.11x10^{0}$
TOTAL	$2.32x10^{2}$	$1.25x10^{2}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.56x10^{2}$

Table 15A-27
Time Dependent Released Activity during SGTR (Ci)
(Transient-initiated Iodine Spike)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	$3.43x10^{3}$	$4.64x10^{1}$	$2.06x10^{2}$	$1.59x10^{3}$	$5.27x10^{3}$
Kr-85m	$6.17x10^{1}$	$9.70x10^{-2}$	$8.00x10^{-3}$	$0.00x10^{0}$	$6.18x10^{1}$
Kr-87	$3.40x10^{1}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.40x10^{1}$
Kr-88	$1.11x10^{2}$	$6.00x10^{-2}$	$1.00x10^{-2}$	$0.00x10^{0}$	$1.11x10^{2}$
Xe-133	$1.16x10^{4}$	$1.45x10^{2}$	$5.06x10^{2}$	$9.44x10^{2}$	$1.32x10^{4}$
Xe-135	$3.70x10^{2}$	$3.82x10^{0}$	$6.70x10^{-1}$	$0.00x10^{0}$	$3.74x10^{2}$
Iodines					
I-131	$1.10x10^{2}$	$1.03x10^{1}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.20x10^{2}$
I-132	$5.24x10^{1}$	$2.12x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.26x10^{1}$
I-133	$1.87x10^{2}$	$1.27x10^{1}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.00x10^{2}$
I-134	$3.05x10^{1}$	$1.06x10^{-3}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.05x10^{1}$
I-135	$1.19x10^{2}$	$3.74x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.23x10^{2}$
Alkali Metals					
Rb-86	$4.54x10^{-3}$	$5.44x10^{-4}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.09x10^{-3}$
Cs-134	$4.63x10^{-1}$	$5.63x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.19x10^{-1}$
Cs-136	$1.22x10^{-1}$	$1.45x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.37x10^{-1}$
Cs-137	$2.64x10^{-1}$	$3.21x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.96x10^{-1}$
TOTAL	$1.61x10^{4}$	$2.22x10^{2}$	$7.12 x10^{2}$	$2.53x10^{3}$	$1.96x10^{4}$

Table 15A-28
Time Dependent Released Activity during SGTR (Ci)
(Pre-transient Iodine Spike)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	$3.43x10^{3}$	$4.64x10^{1}$	$2.06x10^{2}$	$1.59x10^{3}$	$5.27x10^{3}$
Kr-85m	$6.17x10^{1}$	$9.70x10^{-2}$	$8.00x10^{-3}$	$0.00x10^{0}$	$6.18x10^{1}$
Kr-87	$3.40x10^{1}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.40x10^{1}$
Kr-88	$1.11x10^{2}$	$6.00x10^{-2}$	$1.00x10^{-2}$	$0.00x10^{0}$	$1.11x10^{2}$
Xe-133	$1.16x10^{4}$	$1.44x10^{2}$	$5.06x10^{2}$	$9.44x10^{2}$	$1.32x10^{4}$
Xe-135	$3.75x10^{2}$	$2.18x10^{0}$	$6.70x10^{-1}$	$0.00x10^{0}$	$3.78x10^{2}$
Iodines					
I-131	$4.18x10^{2}$	$1.81x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.20x10^{2}$
I-132	$2.09x10^{2}$	$3.92x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.09x10^{2}$
I-133	$7.16x10^{2}$	$2.24x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$7.18x10^{2}$
I-134	$1.28x10^{2}$	$6.00x10^{-5}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.28x10^{2}$
I-135	$4.61x10^{2}$	$6.70x10^{-1}$	$0.00x10^{0}$	$0.00x10^{0}$	$4.62x10^{2}$
Alkali Metals					
Rb-86	$4.54x10^{-3}$	$5.44x10^{-4}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.09x10^{-3}$
Cs-134	$4.63x10^{-1}$	$5.63x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.19x10^{-1}$
Cs-136	$1.22x10^{-1}$	$1.45x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.37x10^{-1}$
Cs-137	$2.64x10^{-1}$	$3.21x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.96x10^{-1}$
TOTAL	$1.76x10^{4}$	$1.98x10^{2}$	$7.12x10^{2}$	$2.53x10^{3}$	$2.10x10^{4}$

Table 15A-29
Time Dependent Released Activity during RCP Rotor Seizure (Ci)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	1.12×10^{2}	8.40×10^{1}	0.00×10^{0}	0.00×10^{0}	1.96×10^{2}
Kr-85m	6.40×10^{2}	1.58×10^{2}	0.00×10^{0}	0.00×10^{0}	7.98×10^{2}
Kr-87	5.02×10^{2}	6.21×10^{0}	0.00×10^{0}	0.00×10^{0}	5.08×10^{2}
Kr-88	1.37×10^{3}	1.74×10^{2}	0.00×10^{0}	0.00×10^{0}	1.55×10^{3}
Xe-133	6.87×10^{3}	4.96×10^{3}	0.00×10^{0}	0.00×10^{0}	1.18×10^{4}
Xe-135	1.61×10^{3}	7.67×10^{2}	0.00×10^{0}	0.00×10^{0}	2.37×10^{3}
Iodines					
I-131	8.81×10^{1}	2.32×10^{2}	0.00×10^{0}	0.00×10^{0}	3.20×10^{2}
I-132	1.94×10^{1}	8.35×10^{0}	0.00×10^{0}	0.00×10^{0}	2.77×10^{1}
I-133	9.85×10^{1}	2.17×10^{2}	0.00×10^{0}	0.00×10^{0}	3.15×10^{2}
I-134	6.46×10^{0}	1.10×10^{-1}	0.00×10^{0}	0.00×10^{0}	6.57×10^{0}
I-135	6.38×10^{1}	9.16×10^{1}	0.00×10^{0}	0.00×10^{0}	1.55×10^{2}
Alkali Metals					
Rb-86	3.23×10^{-2}	8.66×10^{-2}	0.00×10^{0}	0.00×10^{0}	1.19×10^{-1}
Cs-134	3.24×10^{0}	8.78×10^{0}	0.00×10^{0}	0.00×10^{0}	1.20×10^{1}
Cs-136	8.72×10^{-1}	2.33×10^{0}	0.00×10^{0}	0.00×10^{0}	3.21×10^{0}
Cs-137	1.84×10^{0}	5.00×10^{0}	0.00×10^{0}	0.00×10^{0}	6.84×10^{0}
TOTAL	1.14×10^{4}	6.71×10^{3}	0.00×10^{0}	0.00×10^{0}	1.81×10^{4}

Table 15A-30
Time Dependent Released Activity during Rod Ejection Accident (Ci)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	2.63×10^{2}	2.50×10^{2}	1.90×10^{2}	1.63×10^{3}	2.33×10^{3}
Kr-85m	3.59×10^{3}	9.58×10^{2}	9.86×10^{0}	0.00×10^{0}	4.56×10^{3}
Kr-87	2.81×10^{3}	3.50×10^{1}	0.00×10^{0}	0.00×10^{0}	2.85×10^{3}
Kr-88	7.70×10^{3}	1.02×10^{3}	2.05×10^{0}	0.00×10^{0}	8.72×10^{3}
Xe-133	3.81×10^{4}	3.46×10^{4}	2.11×10^{4}	4.22×10^{4}	1.36×10^{5}
Xe-135	9.31×10^{3}	5.32×10^{3}	5.40×10^{2}	2.81×10^{0}	1.52×10^{4}
Iodines					
I-131	5.82×10^{2}	7.17×10^{2}	2.58×10^{2}	7.79×10^{2}	2.34×10^{3}
I-132	4.62×10^{2}	3.93×10^{1}	1.40×10^{-2}	0.00×10^{0}	5.01×10^{2}
I-133	1.12×10^{3}	1.06×10^{3}	1.13×10^{2}	1.13×10^{1}	2.30×10^{3}
I-134	4.95×10^{2}	5.15×10^{-1}	0.00×10^{0}	0.00×10^{0}	4.95×10^{2}
I-135	8.75×10^{2}	4.39×10^{2}	6.60×10^{0}	4.00×10^{-3}	1.32×10^{3}
Alkali Metals					
Rb-86	4.16×10^{-1}	9.65×10^{-2}	0.00×10^{0}	0.00×10^{0}	5.13×10^{-1}
Cs-134	4.15×10^{1}	9.79×10^{0}	1.01×10^{-3}	0.00×10^{0}	5.13×10^{1}
Cs-136	1.13×10^{1}	2.60×10^{0}	1.00×10^{-6}	0.00×10^{0}	1.39×10^{1}
Cs-137	2.36×10^{1}	5.57×10^{0}	0.00×10^{0}	0.00×10^{0}	2.92×10^{1}
TOTAL	6.53×10^{4}	4.45×10^{4}	2.22×10^{4}	4.46×10^{4}	1.77×10^{5}

Table 15A-31
Time Dependent Released Activity during Fuel Handling Accident (Ci)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	$1.20x10^{3}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.20x10^{3}$
Kr-85m	$3.90x10^{2}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.90x10^{2}$
Kr-87	$5.98x10^{-2}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$5.98x10^{-2}$
Kr-88	$1.25x10^{2}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.25x10^{2}$
Xe-133	$9.90x10^{4}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$9.90x10^{4}$
Xe-135	$2.21x10^{4}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.21x10^{4}$
Iodines					
I-131	$3.67x10^{2}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.67x10^{2}$
I-132	$2.75x10^{2}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.75x10^{2}$
I-133	$2.31x10^{2}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.31x10^{2}$
I-134	$2.71x10^{-6}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$2.71x10^{-6}$
I-135	$3.80x10^{1}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$3.80x10^{1}$
TOTAL	$1.24x10^{5}$	$0.00x10^{0}$	$0.00x10^{0}$	$0.00x10^{0}$	$1.24x10^{5}$

Table 15A-32
Time Dependent Released Activity during Failure of Small Lines Carrying Primary Coolant Outside Containment (Ci)

Nuclide	0-8hr	8-24hr	24-96hr	96-720hr	TOTAL
Noble Gases					
Kr-85	6.84×10^{2}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	6.84×10^{2}
Kr-85m	1.25×10^{1}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.25×10^{1}
Kr-87	7.05×10^{0}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	7.05×10^{0}
Kr-88	2.26×10^{1}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	2.26×10^{1}
Xe-133	2.32×10^{3}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	2.32×10^{3}
Xe-135	7.70×10^{1}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	7.70×10^{1}
Iodines					
I-131	1.72×10^{2}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.72×10^{2}
I-132	7.98×10^{1}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	7.98×10^{1}
I-133	2.93×10^{2}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	2.93×10^{2}
I-134	4.33×10^{1}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	4.33×10^{1}
I-135	1.85×10^{2}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	1.85×10^{2}
TOTAL	3.90×10^{3}	0.00×10^{0}	0.00×10^{0}	0.00×10^{0}	3.90×10^{3}

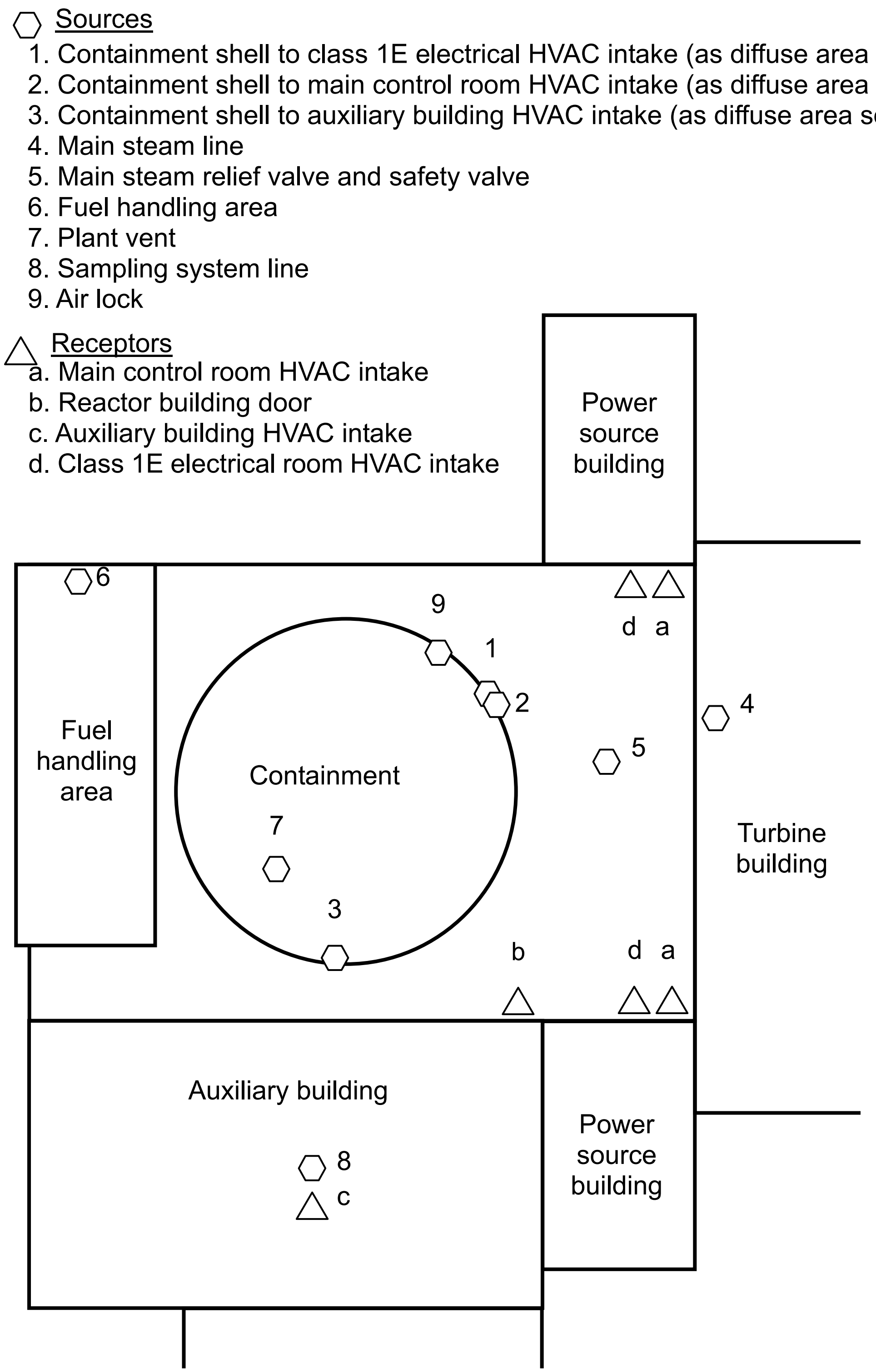


Figure 15A-1 Site Plan with Release and Intake Locations

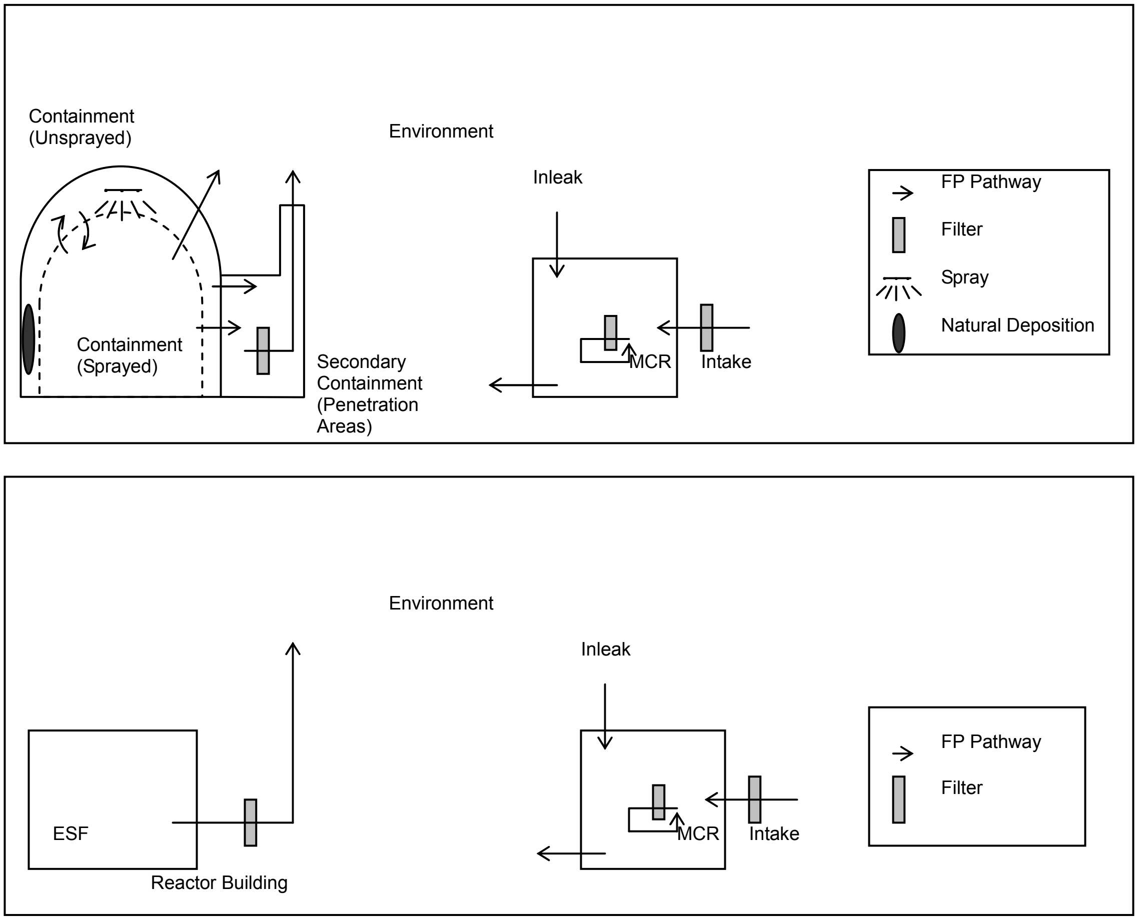


Figure 15A-2 Leakage Dose Model for LOCA

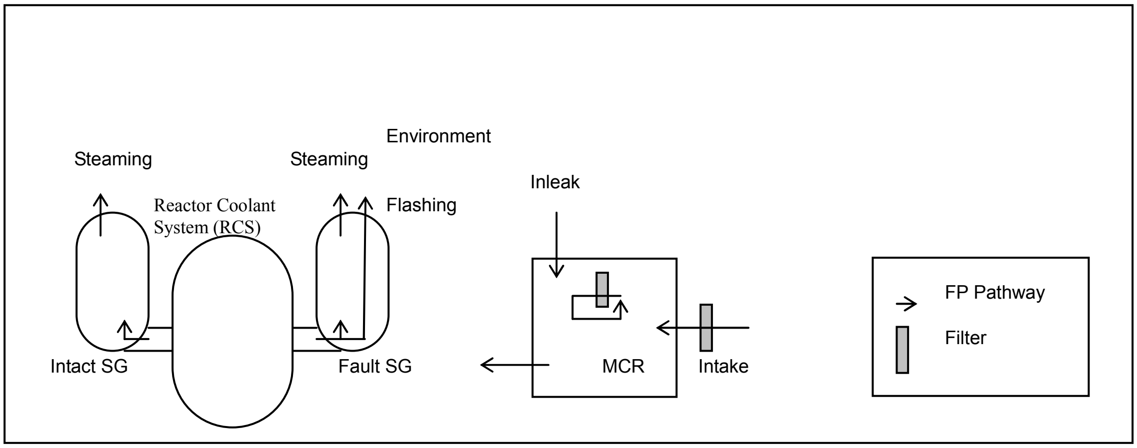


Figure 15A-3 Leakage Dose Model for Steam System Piping Failure or SGTR

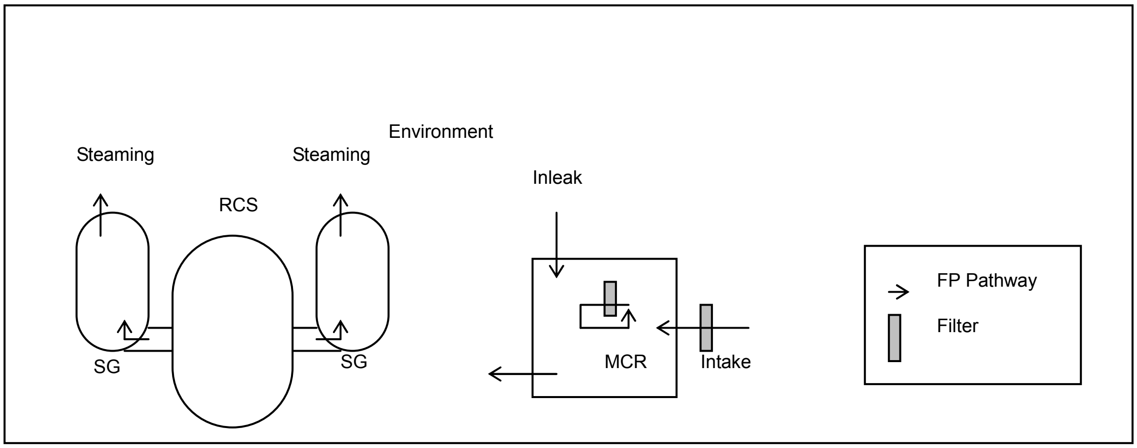


Figure 15A-4 Leakage Dose Model for RCP Rotor Seizure

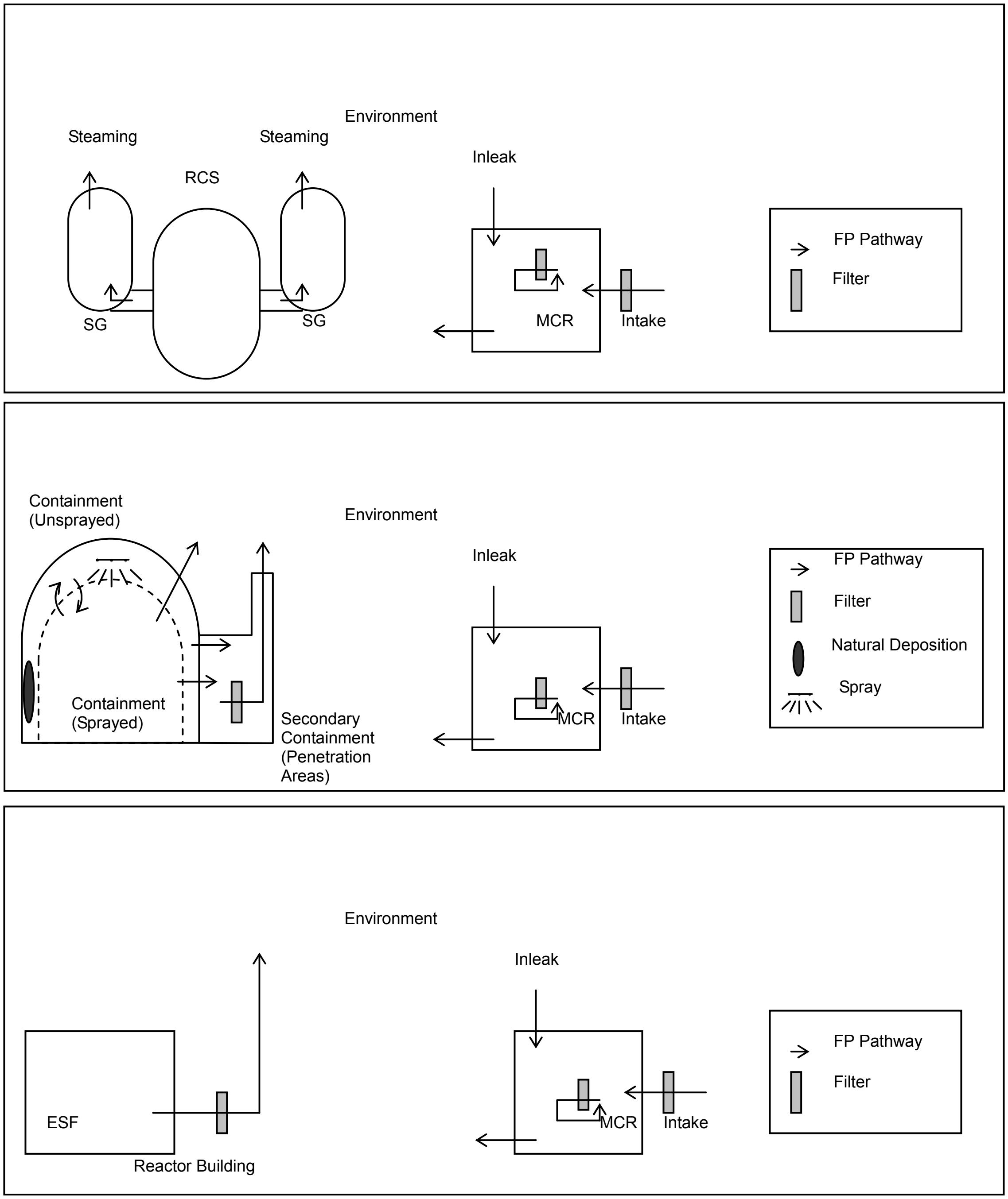
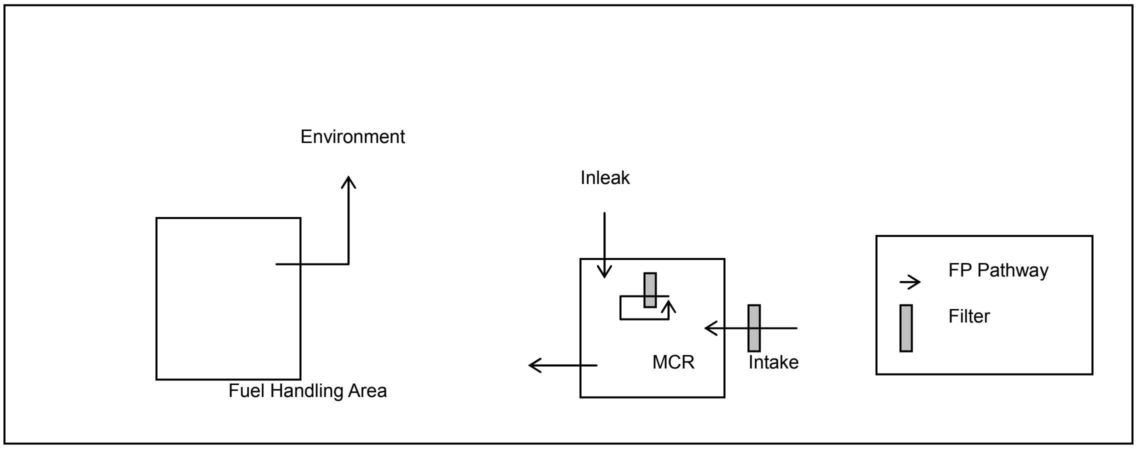


Figure 15A-5 Leakage Dose Model for Rod Ejection Accident



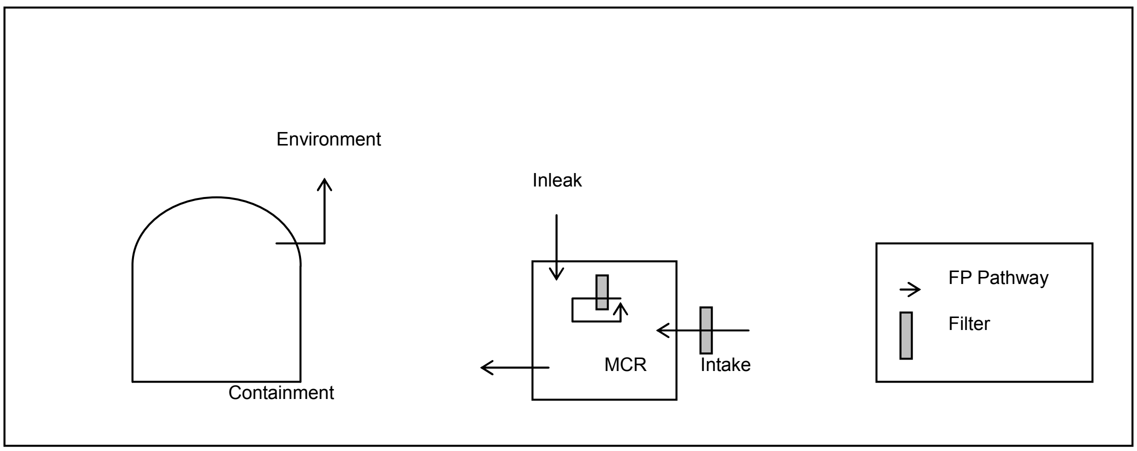


Figure 15A-6 Leakage Dose Model for Fuel Handling Accident

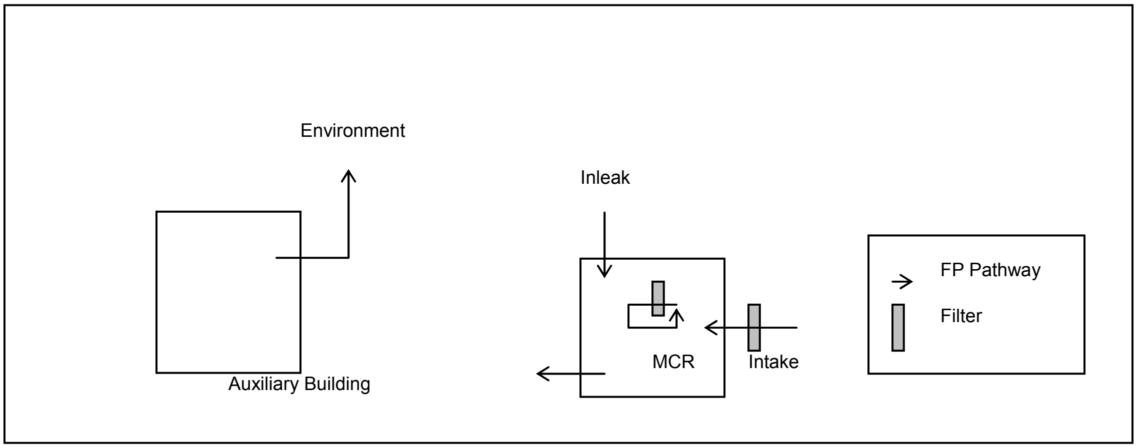


Figure 15A-7 Leakage Dose Model for Failure of Small Lines Carrying Primary Coolant Outside Containment