APPENDIX B

AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

General Non-LOCA Change Summary

The following items are general change items that are applicable to the Non-LOCA Safety Analyses provided in the Core Reference Report.

- All of the Non-LOCA safety analyses of Chapter 15 have been revised for the Advanced First Core design changes. These changes included any impacts on core reactivity feedback parameters, core thermal limits and core peaking factors.

- The revisions to Chapter 15 also include numerous editorial changes. The editorial changes included further clarification on how a loss of AC power is addressed for each event and the application of the protection system delay between reactor trip and turbine trip.

- The system transient analyses are performed using the LOFTRAN computer code. The original and previous system analyses used a modified version of LOFTRAN referred to as LOFTRAN-AP (see WCAP-15644-P). LOFTRAN was modified to include the passive safety system of the AP1000 design to create LOFTRAN-AP. The passive plant changes have been combined with the latest version of LOFTRAN to create a single version of the code for use in AP1000 analyses and operating plant analyses. This latest version of LOFTRAN has been used for the Advanced First Core analyses.

Change No.	Chapter 15 Section 15.0	Change Summary Description
[15.0-1]	15.0.1.2 first paragraph	Editorial change for consistency with subsection 15.5.2.
[15.0-2]	15.0.3.2 second paragraph	Core power uncertainty and average reactor coolant temperature uncertainty revised to reflect values used in the non-LOCA analyses for the Advanced First Core (AFC) design.
[15.0-3]	15.0.4 second paragraph	Clarification added as a result of markups from the Chapter 15 analyses for the AFC design.
[15.0-4]	15.0.5 first and second paragraph	The times for RCCA insertion to the dashpot with all RCPs coasting down and when some or all RCPs are running have been revised to 2.3 and 2.7 seconds, respectively. This is consistent with the revised rod insertion curve modeled in the Chapter 15 analyses for the AFC design.
[15.0-5]	15.0.10.1 first paragraph	WCAP-12945-P-A revised to reflect the current status of the WCAP and WCAP-16009-P-A added to reflect the ASTRUM methodology that was used in the Chapter 15 analyses used for the AFC design.
[15.0-6]	15.0.11.1 first paragraph	"… such as rod ejection accidents …" deleted because FACTRAN is no longer used for the Rod Ejection analyses (see subsection 15.4.8) performed in accordance with the requirements of SRP 4.2 Revision 3.
[15.0-7]	Entire Section 15.0.11.6	Section added to describe the ANC code that is used in the analyses of the Rod Ejection event (see subsection 15.4.8) per the requirements of SRP 4.2 Rev 3.
[15.0-8]	15.0.13 first paragraph	Paragraph added to include a discussion of operator actions which are necessary as demonstrated in the Chapter 15 analyses for the AFC design. This discussion is consistent with a response to an NRC RAI (RAI-SRP15.0-SRSB-03, Revision 2).
[15.0-9]	15.0.16 Reference 15	Added new reference WCAP-16009 – consistent with the change to Section 15.0.10.1.
[15.0-10]	Table 15.0-2 (Sheet 1 of 5)	Updated computer codes used in "Feedwater system malfunctions that result in an increase in feedwater flow" entry to be consistent with the loss of offsite power assumption.
[15.0-11]	Table 15.0-2 (Sheet 1 of 5)	Updated computer codes used in "Excessive increase in secondary steam flow" entry to be consistent with the loss of offsite power assumption.
[15.0-12]	Table 15.0-2 (Sheet 1 of 5)	Revised the "Steam system piping failure" entry to include information for Hot Full Power and Zero Power cases from the Advanced First Core non LOCA analyses.
[15.0-13]	Table 15.0-2 (Sheet 2 of 5)	The Loss of Load/Turbine Trip entry was revised to reflect the use of a moderator density coefficient (MDC) which is a function of density for the DNB case, and the use of 101% rated thermal power (RTP) for the non-DNB case.
[15.0-14] through [15.0-16]	Table 15.0-2 (Sheet 2 of 5)	The Loss of AC Power (LOAC), Loss of Normal Feedwater (LONF) and Feedline Break (FLB) entries were revised to reflect the use of 101% RTP

Change No.	Chapter 15 Section 15.0	Change Summary Description
[15.0-17]	Table 15.0-2 (Sheet 3 of 5)	The Loss of Flow entry was revised to reflect the use of a moderator density coefficient (MDC) which is a function of density.
[15.0-18]	Table 15.0-2 (Sheet 3 of 5)	The Locked Rotor entry was revised to reflect the use of a moderator density coefficient (MDC) which is a function of density, as well as the use of 100% and 101% RTP for the appropriate cases.
[15.0-19]	Table 15.0-2 (Sheet 3 of 5)	The Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition event was revised to reflect the appropriate Doppler defect used in the non LOCA analyses for the AFC design.
[15.0-20]	Table 15.0-2 (Sheet 3 of 5)	Revised the computer codes used in the "Uncontrolled RCCA bank withdrawal at power" to be consistent with the loss of offsite power assumption.
[15.0-21]	Table 15.0-2 (Sheet 4 of 5)	Revised Spectrum of RCCA Ejection Accidents to reflect revised methodology for analyzing rod ejection accidents in compliance with SRP 4.2 Revision 3.
[15.0-22]	Table 15.0-2 (Sheet 4 of 5)	Revised Inadvertent Operation of CMT during Power Operation to reflect the Chapter 15 analyses of the AFC design.
[15.0-23]	Table 15.0-2 (Sheet 4 of 5)	Revised CVS Malfunction entry to reflect the use of 101% RTP in the Chapter 15 analyses of the AFC design.
[15.0-24]	Table 15.0-2 (Sheet 5 of 5)	Updated computer codes and Doppler input entries used in the analysis of the "Inadvertent opening of a pressurizer safety valve and inadvertent operation of ADS" event consistent with the Chapter 15 analyses for the Advanced First Core design.
[15.0-25]	Table 15.0-2 (Sheet 5 of 5)	Revised "Steam Generator tube failure" to be consistent with the inputs for the Advanced First Core non LOCA analyses.
[15.0-26]	Table 15.0-2 (Sheet 5 of 5)	Clarify the initial thermal power output assumed for the LOCA analyses.
[15.0-27]	Table 15.0-2 footnote a	Footnote clarified to reflect inputs used in Chapter 15 analyses of the AFC design.
[15.0-28]	Table 15.0-2 footnote b	Footnote deleted to be consistent with the methodology change for the Rod Ejection analyses.
[15.0-29]	Table 15.0-4a (Sheet 1 of 2)	The time delay for the Overpower ΔT trip was changed from 2.0 to 1.0 seconds. This is consistent with the most limiting assumption for the Chapter 15 analyses for the AFC design, which was in the hot full power steamline break analysis.
[15.0-30]	Table 15.0-4a (Sheet 1 of 2)	The time delay for the reactor trip on reactor coolant pump underspeed was increased to 0.8 second. This provided additional margin between the calculated reactor trip delay and the safety analysis reactor trip delay.

Change No.	Chapter 15 Section 15.0	Change Summary Description
[15.0-31]	Table 15.0-4a (Sheet 1 of 2)	The setpoint for the reactor trip on low steam generator narrow range level was changed to 0% span from 95,000 lbm. The events which trip on this function (LONF/ LOAC and FLB) model the trip occurring at the steam generator mass corresponding to the 0% level for the conditions associated with that event. However, the mass associated with 0% level is different for the different events. Therefore, it is more appropriate to report this trip in terms of % span.
[15.0-32]	Table 15.0-4a (Sheet 1 of 2)	The setpoint for the PRHR actuation on low steam generator wide range level was changed to 22.3% span from 55,000 lbm. A low steam generator wide rage level setpoint of 22.3% span corresponds to 54,634 lbm during a loss of normal feedwater event The FLB conservatively models this setpoint at the steam generator mass corresponding to 5% wide range level span for the conditions associated with that event (19,000 lbm).
[15.0-33]	Table 15.0-4a (Sheet 1 of 2)	The upper bound limiting setpoint for the low Tcold was added, since both directions are modeled in the LONF/LOAC analysis.
[15.0-34]	Table 15.0-4a (Sheet 2 of 2)	Deletion of 2.2 (LBLOCA) for the Time Delay is consistent with the BELOCA analysis performed for the AFC design.
[15.0-35]	Table 15.0-4a (Sheet 2 of 2)	Revisions to Time Delay for RCP trip following S signal reflect the input values used for the LOCA and non-LOCA analyses performed for the AFC design.
[15.0-36]	Table 15.0-4a (Sheet 2 of 2)	Revised "of" to "on" as an editorial change in PRHR actuation on high-3 pressurizer water level.
[15.0-37]	Table 15.0-4a (Sheet 2 of 2)	The limiting setpoint for the CVS isolation on High-2 pressurizer level was revised to reflect the input modeled in the limiting LONF case
[15.0-38]	Table 15.0-4a (Sheet 2 of 2)	The limiting setpoint for the CVS isolation on High-1 pressurizer level was revised to reflect the input modeled in the limiting LONF case
[15.0-39]	Table 15.0-4a footnote	Footnote deleted because they are no longer applicable.
[15.0-40]	Table 15.0-4b footnote	Footnote deleted because they are no longer applicable.
[15.0-41]	Table 15.0-5	Assumed calorimetric error updated To be consistent with Change Number 15.0-2 idenftied above.
[15.0-42]	Table 15.0-6 (Sheet 2 of 5)	Revised the Loss of nonemergency ac power to the station auxiliaries Reactor trip functions to be consistent with the input to the Advanced First Core non LOCA analyses.
[15.0-43]	Table 15.0-6 (Sheet 2 of 5)	Added reactor vessel head vent to be consistent with operator action assumed in the analysis.
[15.0-44]	Table 15.0-6 (Sheet 2 of 5)	Editorial change to be consistent with the text in Section 15.2.8.1.
[15.0-45]	Table 15.0-6 (Sheet 3 of 5)	Low pressurizer pressure added to the RCCA misalignment Reactor trip functions to be consistent with the assumptions in the Advanced First Core non LOCA analyses.
[15.0-46]	Table 15.0-6 (Sheet 4 of 5)	Added reactor vessel head vent to be consistent with operator action assumed in the analysis.

Change No.	Chapter 15 Section 15.0	Change Summary Description
[15.0-47]	Table 15.0-6 (Sheet 4 of 5)	Added reactor vessel head vent to be consistent with operator action assumed in the analysis.
[15.0-48]	Table 15.0-6 (Sheet 4 of 5)	Deleted ADS from the ESF and Other Equipment entry for the Inadvertent opening of a pressurizer safety valve or ADS path to be consistent with the Advanced First Core non LOCA analyses.
[15.0-49]	Table 15.0-7 (Sheet 1 of 2)	Consistent with the Chapter 15 analyses for the AFC design, the failure of one protection division for the excessive steam flow event was deleted and Note (a) was associated with this event since no protection action is required for this event.
[15.0-50]	Table 15.0-7 (Sheet 1 of 2)	For the steam system piping failure, the failure of one core makeup tank discharge valve is assumed for the zero power case. For the full power case, the single failure assumed is that of one protection system. These assumptions are consistent with the Chapter 15 analyses for the AFC design.
[15.0-51]	Table 15.0-7 (Sheet 2 of 2)	Revised the steam Generator Tube Rupture failure entry to be consistent with the steam generator tube rupture analysis for the AFC design.
[15.0-52]	Table 15.0-8	Corrected to be consistent with section number for the event.
[15.0-53]	Figure 15.0.3-1	This figure was replaced with one that incorporates the digital OPΔT and OTΔT protection functions modeled in the Chapter 15 analyses for the AFC design. The new limit lines are based on the core limit lines used in the analyses and the limiting OPΔT setpoint modeled in the hot full power steamline break.
[15.0-54]	Figure 15.0.3-2	For multi-purpose use, DCD Rev 19 figure replaced with one that included metric dimensions. Nothing else was changed.
[15.0-55]	Figure 15.0.5-1	This figure reflects the revised rod insertion curve, updated for the new rod drop times, modeled in the Chapter 15 analyses for the AFC design.
[15.0-56]	Figure 15.0.5-3	This figure reflects the revised rod insertion curve, updated for the new rod drop times, modeled in the Chapter 15 analyses for the AFC design.
[15.0-57]	Table 15.0-2 (Sheet 5 of 5)	This table has been updated consistent with the response provided to CRR-008.

CHAPTER 15

ACCIDENT ANALYSES

15.0.1 Classification of Plant Conditions

The ANSI 18.2 (Reference 1) classification divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

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

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk, and those extreme situations having the potential for the greatest risk should be those least likely to occur. Where applicable, reactor trip and engineered safeguards functioning are assumed to the extent allowed by considerations such as the single failure criterion in fulfilling this principle.

15.0.1.1 Condition I: Normal Operation and Operational Transients

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

Because Condition I events occur frequently, they must be considered from the point of view of their effect on the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions that can occur during Condition I operation.

A typical list of Condition I events follows.

Steady-state and Shutdown Operations

See Table 1.1-1 of Chapter 16.

Operation with Permissible Deviations

Various deviations that occur during continued operation as permitted by the plant Technical Specifications are considered in conjunction with other operational modes. These deviations include the following:

- Operation with components or systems out of service (such as an inoperable rod cluster control assembly [RCCA])
- Leakage from fuel with limited cladding defects
- Excessive radioactivity in the reactor coolant:
  - Fission products
  - Corrosion products
  - Tritium
- Operation with steam generator tube leaks
- Testing

Operational Transients

- Plant heatup and cooldown
- Step load changes (up to $\pm$10 percent)
- Ramp load changes (up to 5 percent/minute)
- Load rejection up to and including design full-load rejection transient

15.0.1.2 Condition II: Faults of Moderate Frequency

These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault (Condition III or IV events). In addition, Condition II events are not expected to result in fuel rod failures, reactor coolant system failures, or secondary system overpressurization. The following faults are included in this category:

- Feedwater system malfunctions that result in a decrease in feedwater temperature (see subsection 15.1.1)
- Feedwater system malfunctions that result in an increase in feedwater flow (see subsection 15.1.2)
- Excessive increase in secondary steam flow (see subsection 15.1.3)
- Inadvertent opening of a steam generator relief or safety valve (see subsection 15.1.4)

- Inadvertent operation of the passive residual heat removal heat exchanger (see subsection 15.1.6)
- Loss of external electrical load (see subsection 15.2.2)
- Turbine trip (see subsection 15.2.3)
- Inadvertent closure of main steam isolation valves (see subsection 15.2.4)
- Loss of condenser vacuum and other events resulting in turbine trip (see subsection 15.2.5)
- Loss of ac power to the station auxiliaries (see subsection 15.2.6)
- Loss of normal feedwater flow (see subsection 15.2.7)
- Partial loss of forced reactor coolant flow (see subsection 15.3.1)
- Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition (see subsection 15.4.1)
- Uncontrolled RCCA bank withdrawal at power (see subsection 15.4.2)
- RCCA misalignment (dropped full-length assembly, dropped full-length assembly bank, or statically misaligned assembly) (see subsection 15.4.3)
- Startup of an inactive reactor coolant pump at an incorrect temperature (see subsection 15.4.4)
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (see subsection 15.4.6)
- Inadvertent operation of the passive core cooling system during power operation (see subsection 15.5.1)
- Chemical and volume control system malfunction that increases reactor coolant inventory (see subsection 15.5.2)
- Inadvertent opening of a pressurizer safety valve (see subsection 15.6.1)
- Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate containment (see subsection 15.6.2)

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15.0.1.3 Condition III: Infrequent Faults

Condition III events are faults that may occur infrequently during the life of the plant. They may result in the failure of only a small fraction of the fuel rods. The release of radioactivity is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary, in accordance with the guidelines of 10 CFR 50.34. By definition, a Condition III event alone does not generate a Condition IV event or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults are included in this category:

- Steam system piping failure (minor) (see subsection 15.1.5)
- Complete loss of forced reactor coolant flow (see subsection 15.3.2)
- RCCA misalignment (single RCCA withdrawal at full power) (see subsection 15.4.3)
- Inadvertent loading and operation of a fuel assembly in an improper position (see subsection 15.4.7)
- Inadvertent operation of automatic depressurization system (see subsection 15.6.1)
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (see subsection 15.6.5)
- Gas waste management system leak or failure (see subsection 15.7.1)
- Liquid waste management system leak or failure (see subsection 15.7.2)
- Release of radioactivity to the environment due to a liquid tank failure (see subsection 15.7.3)
- Spent fuel cask drop accidents (see subsection 15.7.5)

15.0.1.4 Condition IV: Limiting Faults

Condition IV events are faults that are not expected to take place, but are postulated because their consequences include the potential of the release of significant amounts of radioactive material. They are the faults that must be designed against, and they represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of the guideline values of 10 CFR 50.34. A single Condition IV event is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults are classified in this category:

- Steam system piping failure (major) (see subsection 15.1.5)
- Feedwater system pipe break (see subsection 15.2.8)
- Reactor coolant pump shaft seizure (locked rotor) (see subsection 15.3.3)
- Reactor coolant pump shaft break (see subsection 15.3.4)
- Spectrum of RCCA ejection accidents (see subsection 15.4.8)
- Steam generator tube rupture (see subsection 15.6.3)
- LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (see subsection 15.6.5)
- Design basis fuel handling accidents (see subsection 15.7.4)

15.0.2 Optimization of Control Systems

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

The plant control system setpoint study is comprised of analyses of the following control systems: plant control, axial offset control, rapid power reduction, steam dump (turbine bypass), steam generator level, pressurizer pressure, and pressurizer level.

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values assumed in the analyses performed. The thermal power output includes the effective thermal power generated by the reactor coolant pumps. Selected AP1000 loop layout elevations are shown in Figure 15.0.3-2 to aid in interpreting plots shown in other Chapter 15 subsections.

The values of other pertinent plant parameters used in the accident analyses are given in Table 15.0-3.

15.0.3.2 Initial Conditions

For most accidents that are departure from nucleate boiling (DNB) limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the departure from nucleate boiling ratio (DNBR) design limit values (see subsection 4.4), as described in WCAP-11397-P-A (Reference 2). This procedure is known as the Revised Thermal Design Procedure (RTDP) and is discussed more fully in Section 4.4.

For most accidents that are not DNB limited, or for which the revised thermal design procedure is not used, the initial conditions are obtained by adding the maximum steady-state errors to rated values. The following conservative steady-state errors are assumed in the analysis:

Core power	± 1 percent allowance for calorimetric error
Average reactor coolant system temperature	±8.0°F allowance for controller deadband and measurement errors
Pressurizer pressure	± 50 psi allowance for steady-state fluctuations and measurement errors

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Initial values for core power, average reactor coolant system temperature, and pressurizer pressure are selected to minimize the initial DNBR unless otherwise stated in the sections describing the specific accidents. Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies and control rods. Power distribution may be characterized by the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) and the total peaking factor (F_q). Unless specifically noted otherwise, the peaking factors used in the accident analyses are those presented in Chapter 4.

For transients that may be DNB limited, the radial peaking factor is important. The radial peaking factor increases with decreasing power level due to control rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.0.3-1. Transients that may be departure from nucleate boiling limited are assumed to begin with an $F_{\Delta H}$, consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is a chopped cosine, as discussed in subsection 4.4, for transients analyzed at full power and the most limiting power shape calculated or allowed for accidents initiated at nonfull power or asymmetric RCCA conditions.

The radial and axial power distributions just described are input to the VIPRE-01 code as described in subsection 4.4.

For transients that may be overpower-limited, the total peaking factor (F_q) is important. Transients that may be overpower-limited are assumed to begin with plant conditions, including power distributions, which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients that are slow with respect to the fuel rod thermal time constant (for example, the chemical and volume control system malfunction that results in a slow decrease in the boron concentration in the reactor coolant system as well as an excessive increase in secondary steam flow) and that may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in subsection 4.4.

For overpower transients that are fast with respect to the fuel rod thermal time constant (for example, the uncontrolled RCCA bank withdrawal from subcritical or lower power startup and RCCA ejection incident, both of which result in a large power rise over a few seconds), a detailed fuel transient heat transfer calculation is performed.

15.0.4 Reactivity Coefficients Assumed in the Accident Analysis

The transient response of the reactor system is dependent on reactivity feedback effects, in particular, the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients are discussed in subsection 4.3.2.3.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values; while for other events, the use of small reactivity coefficient values is conservative. The values used are given in Figure 15.0.4-1, which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis. In some cases, conservative combinations of parameters are used to bound the effects of core life, although these combinations may not represent possible realistic situations.

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15.0.5 Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs as a function of time and the variation in rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry,

or approximately 85 percent of the rod cluster travel. In analyses where all of the reactor coolant pumps are coasting down prior to, or simultaneous, with RCCA insertion, a time of 2.3 seconds is used for insertion time to dashpot entry.

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In Figure 15.0.5-1, the curve labeled "complete loss of flow transients" shows the RCCA position versus time normalized to 2.3 seconds assumed in accident analyses where all reactor coolant pumps are coasting down. In analyses where some or all of the reactor coolant pumps are running, the RCCA insertion time to dashpot is conservatively taken as 2.7 seconds. The RCCA position versus time normalized to 2.7 seconds is also shown in Figure 15.0.5-1.

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The use of such a long insertion time provides conservative results for accidents and is intended to apply to all types of RCCAs, which may be used throughout plant life. Drop time testing requirements are specified in the Technical Specifications.

Figure 15.0.5-2 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core. An axial distribution skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, which is input to the point kinetics core models used in transient analyses. The bottom-skewed power distribution itself is not an input into the point kinetics core model.

There is inherent conservatism in the use of Figure 15.0.5-2 in that it is based on a skewed flux distribution, which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significantly more negative reactivity is inserted than that shown in the curve, due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in Figure 15.0.5-3. The curves shown in this figure were obtained from Figures 15.0.5-1 and 15.0.5-2. A total negative reactivity insertion following a trip of 4 percent Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3-3.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.0.5-3) is used in those transient analyses for which a point kinetics core model is used. Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.0.5-1 is used as code input.

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

A reactor trip signal acts to open two trip breaker sets connected in series, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.0-4a. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure 15.0.3-1.

Table 15.0-4a also summarizes the setpoints and the instrumentation delay for engineered safety features (ESF) functions used in accident analyses. Time delays associated with equipment actuated (such as valve stroke times) by ESF functions are summarized in Table 15.0-4b.

The difference between the limiting setpoint assumed for the analysis and the nominal setpoint represents an allowance for instrumentation channel error and setpoint error. Nominal setpoints are specified in the plant Technical Specifications. During plant startup tests, it is demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrument response times are determined periodically in accordance with the plant Technical Specifications.

15.0.7 Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux

Examples of the instrumentation uncertainties and calorimetric uncertainties used in establishing the power range high neutron flux setpoint are presented in Table 15.0-5.

The calorimetric uncertainty is the uncertainty assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a daily basis.

The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. Installed plant instrumentation is used for these measurements.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

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

Chapter 17 discusses the quality assurance program that is implemented to provide confidence that the plant systems satisfactorily perform their assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, provides confidence that the normally operating systems and components listed in Table 15.0-6 are available for mitigation of the events discussed in Chapter 15.

In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI N18.2-1973 (Reference 1) is used. The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-2000 and Regulatory Guide 1.53 in the application of the single-failure criterion. Conformance to Regulatory Guide 1.53 is summarized in subsection 1.9.1.

Table 15.0-8 summarizes the nonsafety-related systems assumed in the analyses to mitigate the consequences of events. Except for the cases listed in Table 15.0-8, control system action is not used for mitigation of accidents.

15.0.9 Fission Product Inventories

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

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

Residual heat in a subcritical core is calculated for the LOCA according to the requirements of 10 CFR 50.46, as described in WCAP-10054-P-A and WCAP-12945-P-A and WCAP-16009-P-A (References 3, 4, and 15). The large-break LOCA methodology considers uncertainty in the decay power level. The small-break LOCA events and post-LOCA long-term cooling analyses use 10 CFR 50, Appendix K, decay heat, which assumes infinite irradiation time before the core goes subcritical to determine fission product decay energy. For all other accidents, the same models are used, except that fission product decay energy is based on core average exposure at the end of an equilibrium cycle.

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15.0.10.2 Distribution of Decay Heat Following a Loss-of-Coolant Accident

During a LOCA, the core is rapidly shut down by void formation, RCCA insertion, or both, and a large fraction of the heat generation considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady-state fission power. Local peaking effects, which are important for the neutron-dependent part of the heat generation, do not apply to the gamma ray contribution. The steady-state factor, which represents the fraction of heat generated within the cladding and pellet, drops to 95 percent or less for the hot rod in a LOCA.

For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; one-half second after the rupture, about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect on the hot rod is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total heat. Because the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods; the remaining 2 percent is absorbed by water, thimbles, sleeves, and grids. Combining the 3 percent total heat reduction from gamma redistribution with this 2 percent absorption produce as the net effect a factor of 0.95, which exceeds the actual heat production in the hot rod. The actual hot rod heat generation is computed during the AP1000 large-break LOCA transient as a function of core fluid conditions.

15.0.11 Computer Codes Used

Summaries of some of the principal computer codes used in transient analyses are given as follows. Other codes – in particular, specialized codes in which the modeling has been developed to simulate one given accident, such as those used in the analysis of the reactor coolant system pipe rupture (see subsection 15.6.5) – are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.0-2. WCAP-15644 (Reference 11) provides the basis for use of analysis codes.

15.0.11.1 FACTRAN Computer Code

FACTRAN (Reference 5) calculates the transient temperature distribution in a cross section of a metal-clad UO_2 fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

- A sufficiently large number of radial space increments to handle fast transients.

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- Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation
- The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials

FACTRAN is further discussed in WCAP-7908-A (Reference 5).

15.0.11.2 LOFTRAN Computer Code

The LOFTRAN (Reference 6) program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides), and pressurizer. The pressurizer heaters, spray, and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The protection and safety monitoring system is simulated to include reactor trips on high neutron flux, overtemperature ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated, including rod control, steam dump, feedwater control, and pressurizer level and pressure control. The emergency core cooling system, including the accumulators, is also modeled.

LOFTRAN is a versatile program suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated in Figure 15.0.3-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

The LOFTRAN code is modified to allow the simulation of the passive residual heat removal (PRHR) heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. A discussion of these models and additional validation is presented in WCAP-14234 (Reference 10).

LOFTTR2 (Reference 8) is a modified version of LOFTRAN with a more realistic break flow model, a two-region steam generator secondary side, and an improved capability to simulate operator actions during a steam generator tube rupture (SGTR) event.

The LOFTTR2 code is modified to allow the simulation of the PRHR heat exchanger, core makeup tanks, and associated protection system actuation logic. The modifications are identical to those made to the LOFTRAN code. A discussion of these models is presented in WCAP-14234 (Reference 10).

15.0.11.3 TWINKLE Computer Code

The TWINKLE (Reference 7) program is a multidimensional spatial neutron kinetics code, which is patterned after steady-state codes currently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions, such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided (for example, channelwise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures).

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

15.0.11.4 VIPRE-01 Computer Code

The VIPRE-01 code is described in subsection 4.4.4.5.2.

15.0.11.5 COAST Computer Program

The COAST computer program is used to calculate the reactor coolant flow coastdown transient for any combination of active and inactive pumps and forward or reverse flow in the hot or cold legs. The program is described in Reference 13 and was referenced in Reference 12. The program was approved in Reference 14.

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

15.0.11.6 ANC Computer Code

The ANC computer code is used to solve the two-group neutron diffusion equation in three spatial dimensions. ANC can also solve the three-dimensional kinetics equations for six delayed neutron groups. The ANC code is described in subsection 4.3.3.3.

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15.0.12 Component Failures

15.0.12.1 Active Failures

SECY-77-439 (Reference 9) provides a description of active failures. An active failure results in the inability of a component to perform its intended function.

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

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

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

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

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

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

15.0.12.2 Passive Failures

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

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

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

15.0.12.3 Limiting Single Failures

The most limiting single active failure (where one exists), as described in Section 3.1, of safety-related equipment, is identified in each analysis description. The consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure that could adversely affect the consequences of the transient is identified. The failure assumed in each analysis is listed in Table 15.0-7.

15.0.13 Operator Actions

There are several events analyzed in the following sections which require operator action to terminate or mitigate the event. The loss of normal feedwater (Section 15.2.7), the inadvertent actuation of a core makeup tank (Section 15.5.1), and the chemical and volume control system malfunction (Section 15.5.2) assume operator action, after the high-2 pressurizer water level setpoint is reached, to open the safety grade reactor vessel head vent. This action prevents filling the pressurizer and allowing water to escape through the pressurizer safety valves. The analysis of the boron dilution for Mode 1 operation with automatic rod control (Section 15.4.6) relies on the operator to terminate the dilution source, after the rod insertion limit alarm, before the required shutdown margin is lost. The small line break outside containment event (Section 15.6.2) assumes the operator will isolate the break. In all cases where operator actions are credited, no operator actions are required within the first 30 minutes of the transient. For these events, before operator action is required numerous alarms and indications would be available to the operator to diagnose the transient and ensure that the proper action is taken.

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

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

15.0.14 Loss of Offsite ac Power

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

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

For those events where offsite ac power is lost, an appropriate time delay between turbine trip and the postulated loss of offsite ac power is assumed in the analyses. A time delay of 3 seconds is used. This time delay is based on the inherent stability of the offsite power grid as discussed in Section 8.2. Following the time delay, the effect of the loss of offsite ac power on plant auxiliary equipment – such as reactor coolant pumps, main feedwater pumps, condenser, startup feedwater pumps, and RCCAs – is considered in the analyses. Turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay is part of the AP1000 reactor trip system.

Design basis LOCA analyses are governed by the GDC-17 requirement to consider the loss of offsite power. For the AP1000 design, in which all the safety-related systems are passive, the availability of offsite power is significant only regarding reactor coolant pump operation for LOCA events. A sensitivity study for AP1000 has shown that for large-break LOCAs, assuming the loss of offsite power coincident with the inception of the LOCA event is nonlimiting relative to assuming continued reactor coolant pump operation until the automatic reactor coolant pump trip occurs following an "S" signal less than 10 seconds into the transient. For small-break LOCA events, the AP1000 automatic reactor coolant pump trip feature prevents continued operation of the reactor coolant pumps from mixing the liquid and vapor present within a two-phase reactor coolant system inventory to increase the liquid break flow and deplete the reactor coolant system mass inventory rapidly. The automatic reactor coolant pump trip occurs early enough during AP1000 small-break LOCA transients that emergency core cooling system performance is not affected by the loss of offsite power assumption because the total break flow is approximately equivalent for reactor coolant pump trip occurring either at time zero or as a result of the "S" signal. Whether a loss of offsite power is postulated at the inception of the LOCA event or occurs automatically later on is unimportant in the subsection 15.6.5.4C long-term cooling analyses because with either

assumption, the reactor coolant pumps are tripped long before the long-term cooling timeframe.

The AP1000 protection and safety monitoring system and passive safeguards systems are not dependent on offsite power or on any backup diesel generators. Following a loss of ac power, the protection and safety monitoring system and passive safeguards are able to perform the safety functions and there are no additional time delays for these functions to be completed.

15.0.15 Combined License Information

15.0.15.1 Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters prior to fuel load, the Combined License holder will calculate the primary power calorimetric uncertainty. The calculations will be completed using an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values.

15.0.16 References

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

2. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-Proprietary), April 1989.

3. Lee, N., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081 (Non-Proprietary), August 1985.

4. Bajorek, S. M., et al., "Code Qualification Document for Best-Estimate LOCA Analysis," WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, (Proprietary) and WCAP-14747 (Non-Proprietary), 1998.

5. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

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7. Risher, D. H., Jr., and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

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

10. Bachrach, U., Carlin, E. L., "LOFTRAN and LOFTTR2 AP600 Code Applicability Document," WCAP-14234, Revision 1 (Proprietary) and WCAP-14235, Revision 1 (Non-Proprietary), August 1997.

11. "AP1000 Code Applicability Report," WCAP-15644-P (Proprietary) and WCAP-15644-NP (Non-Proprietary), Revision 2, March 2004.

12. "Combustion Engineering Standard Safety Analysis Report," CESSAR Docket No. STN-50-470, December 1975.

13. "COAST Code Description," CENPD-98-A, April 1973, Proprietary Information.

14. CENPD-98-A, "COAST Code Description," April 1973 (NRC Approval Letter dated December 4, 1974).

15. Nissley, M. E., et al., 2005, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A and WCAP-16009-NP-A (Non-proprietary).

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Table 15.0-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Thermal power output (MWt)	3415
Effective thermal power generated by the reactor coolant pumps (MWt)	15
Core thermal power (MWt)	3400

Table 15.0-2 (Sheet 1 of 5)							
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED							
Section	**Faults**	**Computer Codes Used**	**Reactivity Coefficients Assumed**			**Initial Thermal Power Output Assumed (MWt)**
			Moderator Density ($\Delta k/gm/cm^3$)	**Moderator Temperature (pcm/°F)**	**Doppler**	
15.1	Increase in heat removal from the primary system					
	Feedwater system malfunctions causing a reduction in feedwater temperature	Bounded by excessive increase in secondary steam flow	–	–	–	–
	Feedwater system malfunctions that result in an increase in feedwater flow	LOFTRAN	0.470	–	Upper curve of Figure 15.0.4-1	0 and 3415
	Excessive increase in secondary steam flow	LOFTRAN	0.0 and 0.470	–	Upper and lower curves of Figure 15.0.4-1	3415
	Inadvertent opening of a steam generator relief or safety valve	LOFTRAN, VIPRE-01	Function of moderator density (see Figure 15.1.4-1)	–	See subsection 15.1.4.	0 (subcritical)
	Steam system piping failure	LOFTRAN, VIPRE-01	Function of moderator density (see Figure 15.1.4-1) for zero power case 0.470 for full power case	–	See subsection 15.1.5 for zero power case Upper curve of Figure 15.0.4-1 for full power case	0 (subcritical) and 3415
	Inadvertent operation of the PRHR heat exchanger	N/A	N/A	–	N/A	3415

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Table 15.0-2 (Sheet 2 of 5) **SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED**						
			Reactivity Coefficients Assumed			
Section	**Faults**	**Computer Codes Used**	**Moderator Density ($\Delta k/gm/cm^3$)**	**Moderator Temperature (pcm/°F)**	**Doppler**	**Initial Thermal Power Output Assumed (MWt)**
15.2	Decrease in heat removal by the secondary system					
	Loss of external electrical load and/or turbine trip	LOFTRAN, FACTRAN, VIPRE-01	0.470 and function of moderator density	–	Lower and upper curves of Figure 15.0.4-1	3415 and 3449.15 (a)
	Inadvertent closure of main steam isolation valves	Bounded by turbine trip event	–	–	–	–
	Loss of condenser vacuum and other events resulting in turbine trip	Bounded by turbine trip event	–	–	–	–
	Loss of nonemergency ac power to the plant auxiliaries	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
	Loss of normal feedwater flow	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
	Feedwater system pipe break	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
15.3	Decrease in reactor coolant system flow rate					

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Table 15.0-2 (Sheet 3 of 5)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed			Initial Thermal Power Output Assumed (MWt)
			Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	
15.3	Partial and complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, COAST, VIPRE-01	0.0 and function of moderator density	–	Lower curve of Figure 15.0.4-1	3415
	Reactor coolant pump shaft seizure (locked rotor) and reactor coolant pump shaft break	LOFTRAN, FACTRAN, COAST, VIPRE-01	0.0 and function of moderator density	–	Lower curve of Figure 15.0.4-1	3415 and 3449.15 (a)
15.4	Reactivity and power distribution anomalies					
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, VIPRE-01	–	0.0	Coefficient is consistent with a Doppler defect of -0.90%Δk	0
	Uncontrolled RCCA bank withdrawal at power	LOFTRAN	0.0 and 0.470	–	Upper and lower curves of Figure 15.0.4-1	10%, 60%, and 100% of 3415
	RCCA misalignment	LOFTRAN, VIPRE-01	NA	–	NA	3415
	Startup of an inactive reactor coolant pump at an incorrect temperature	NA	NA	–	NA	NA

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Table 15.0-2 (Sheet 4 of 5)						
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED						
			Reactivity Coefficients Assumed			
Section	**Faults**	**Computer Codes Used**	**Moderator Density ($\Delta k/gm/cm^3$)**	**Moderator Temperature (pcm/°F)**	**Doppler**	**Initial Thermal Power Output Assumed (MWt)**
15.4	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	–	NA	0 and 3415
	Inadvertent loading and operation of a fuel assembly in an improper position	ANC	NA	–	NA	3415
	Spectrum of RCCA ejection accidents	ANC, VIPRE	Refer to subsection 15.4.8	Refer to subsection 15.4.8	Refer to subsection 15.4.8	Refer to subsection 15.4.8
15.5	Increase in reactor coolant inventory					
	Inadvertent operation of the core makeup tanks during power operation	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3449.15 (a)
	Chemical and volume control system malfunction that increases reactor coolant inventory	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3449.15 (a)

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Table 15.0-2 (Sheet 5 of 5)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.6	Decrease in reactor coolant inventory					
	Inadvertent opening of a pressurizer safety valve and inadvertent operation of ADS	LOFTRAN	0.0	–	Upper curve of Figure 15.0.4-1	3415
	Steam generator tube failure	LOFTTR2	0.0	–	Lower curve of Figure 15.0.4-1	3449.15 (a)
	A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate containment	NA	NA	–	NA	NA
	LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	NOTRUMP WCOBRA/ TRAC	See subsection 15.6.5 references	–	See subsection 15.6.5 references	3434.0 (a) (b)

Notes:

a. The Non LOCA analyses assume an initial power of 101% of the NSSS Power (NSSS Power = rated thermal power (RTP) plus 15 MWt for pump heat) and the LOCA analyses assume an initial power of 101% of RTP.

b. Section 15.6.5.4A describes the large-break LOCA analysis methodology, which includes treatment of the initial thermal power output uncertainty.

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EOC – End of core cycle

Table 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS USED IN ACCIDENT ANALYSES

		Without RTDP[a]	
	RTDP With 10% Steam Generator Tube Plugging	**Without Steam Generator Tube Plugging**	**With 10% Steam Generator Tube Plugging**
Thermal output of NSSS (MWt)	3415	3415	3415
Core inlet temperature (°F)	535.8	535.5	535.0
Vessel average temperature (°F)	573.6	573.6	573.6
Reactor coolant system pressure (psia)	2250.0	2250.0	2250.0
Reactor coolant flow per loop (gpm)	15.08 E+04	14.99 E+04	14.8 E+04
Steam flow from NSSS (lbm/hr)	14.96 E+06	14.96 E+06	14.95 E+06
Steam pressure at steam generator outlet (psia)	802.2	814.0	796.0
Assumed feedwater temperature at steam generator inlet (°F)	440.0	440.0	440.0
Average core heat flux (Btu/-hr-ft^2)	1.99 E+05	1.99 E+05	1.99 E+05

Note:

a. Steady-state errors discussed in subsection 15.0.3 are added to these values to obtain initial conditions for most transient analyses.

Table 15.0-4a (Sheet 1 of 2) **PROTECTION AND SAFETY MONITORING SYSTEM SETPOINTS AND TIME DELAY ASSUMED IN ACCIDENT ANALYSES**		
Function	**Limiting Setpoint Assumed in Analyses**	**Time Delays (seconds)**
Reactor trip on power range high neutron flux, high setting	118%	0.9
Reactor trip on power range high neutron flux, low setting	35%	0.9
Reactor trip on source range neutron flux reactor trip	Not applicable	0.9
Overtemperature ΔT	Variable (see Figure 15.0.3-1)	2.0
Overpower ΔT	Variable (see Figure 15.0.3-1)	1.0
Reactor trip on high pressurizer pressure	2460 psia	2.0
Reactor trip on low pressurizer pressure	1800 psia	2.0
Reactor trip on low reactor coolant flow in either hot leg	87% loop flow	1.45
Reactor trip on reactor coolant pump under speed	90%	0.8
Reactor trip on low steam generator narrow range level	0% of span	2.0
High steam generator narrow range level coincident with reactor trip (P-4)	85% of narrow range level span	2.0 (startup feedwater isolation) 2.0 (chemical and volume control system makeup isolation)
High-2 steam generator level	95% of narrow range level span	2.0 (reactor trip) 0.0 (turbine trip) 2.0 (feedwater isolation)
Reactor trip on high-3 pressurizer water level	76% of span	2.0
PRHR actuation on low steam generator wide range level	22.3% of span	2.0
"S" signal and steam line isolation on low T_{cold}	500°F lower bound 510°F upper bound	2.0

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Table 15.0-4a (Sheet 2 of 2) **PROTECTION AND SAFETY MONITORING SYSTEM SETPOINTS AND TIME DELAY ASSUMED IN ACCIDENT ANALYSES**		
Function	**Limiting Setpoint Assumed in Analyses**	**Time Delays (seconds)**
"S" signal and steam line isolation on low steam line pressure	405 psia (with an adverse environment assumed) 535 psia (without an adverse environment assumed)	2.0
"S" signal on low pressurizer pressure	1700 psia	2.0
Reactor trip on PRHR discharge valves not closed	Valve not closed	1.25
"S" signal on high-2 containment pressure	8 psig	2.0
Reactor coolant pump trip following "S"	–	5.0 5.3 (LBLOCA)
PRHR actuation on high-3 pressurizer water level	76% of span	2.0 (plus 15.0-second timer delay)
Chemical and volume control system isolation on high-2 pressurizer water level	69% of span	2.0
Chemical and volume control system isolation on high-1 pressurizer water level coincident with "S" signal	33% of span	2.0
Boron dilution block on source range flux doubling	3 over 50 minutes	80.0
ADS Stage 1 actuation on core makeup tank low level signal	67.5% of tank volume	32.0 seconds for control valve to begin to open)
ADS Stage 4 actuation on core makeup tank low-low level signal	20% of tank volume	2.0 seconds for squib valve to begin to open)
CMT actuation on pressurizer low-2 water level	0% of span	2.0

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1. The delay times reflect the design basis of the AP1000. The applicable DCD Chapter 15 accidents were evaluated for the design basis delay times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remain valid. The output provided for the analyses is representative of the transient phenomenon.

Table 15.0-4b **LIMITING DELAY TIMES FOR EQUIPMENT ASSUMED IN ACCIDENT ANALYSES**	
Component	**Time Delays (seconds)**
Feedwater isolation valve closure, feedwater control valve closure, or feedwater pump trip	10 (maximum value for non-LOCA) 5 (maximum value for mass/energy)
Steam line isolation valve closure	5
Core makeup tank discharge valve opening time	15 (maximum) 10 (nominal value for best-estimate LOCA)
Chemical and volume control system isolation valve closure	30
PRHR discharge valve opening time	15 (maximum) 10 (nominal value for best-estimate LOCA) 1.0 second (small-break LOCA value: follows a 15-second interval of no valve movement)
Demineralized water transfer and storage system isolation valve closure time	20
Steam generator power-operated relief valve block valve closure	44
Automatic depressurization system (ADS) valve opening times	See Table 15.6.5-10

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1. The valve stroke times reflect the design basis of the AP1000. The applicable DCD Chapter 15 accidents were evaluated for the design basis valve stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remain valid. The output provided for the analyses is representative of the transient phenomenon.

Table 15.0-5 **DETERMINATION OF MAXIMUM POWER RANGE NEUTRON FLUX CHANNEL TRIP SETPOINT, BASED ON NOMINAL SETPOINT AND INHERENT TYPICAL INSTRUMENTATION UNCERTAINTIES**		
Nominal setpoint (% of rated power)		109
Calorimetric errors in the measurement of secondary system thermal power:		
Variable	**Accuracy of Measurement of Variable**	**Effect on Thermal Power Determination (% of Rated Power)**
Feedwater temperature	±3°F	
Steam pressure (small correction on enthalpy)	±6 psi	
Feedwater flow	±0.5% ΔP instrument span (two channels per steam generator)	
Assumed calorimetric error		1.0
Radial power distribution effects on total ion chamber current		7.8 (b)*
Allowed mismatch between power range neutron flux channel and calorimetric measurement		2.0 (c)*
Instrumentation channel drift and setpoint reproducibility	0.4% of instrument span (120% power span)	0.84(d)*
Instrumentation channel temperature effects		0.48(e)*
*Total assumed error in setpoint (% of rated power): $[(a)^2 + (b)^2 + (c)^2 + (d)^2 + (e)^2]^{1/2}$		±8.4
Maximum power range neutron flux trip setpoint assuming a statistical combination of individual uncertainties (% of rated power)		118

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The main feedwater flow measurement supports a 1% power uncertainty; use of a 2% power uncertainty is conservative.

Table 15.0-6 (Sheet 1 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.1			
Increase in heat removal from the primary system			
Feedwater system malfunctions that result in an increase in feedwater flow	High-2 Steam Generator Level, Power range high flux, overtemperature	High-2 steam generator level produced feedwater isolation and turbine trip	Feedwater isolation valves
Excessive increase in secondary steam flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	–	–
Inadvertent opening of a steam generator safety valve	Power range high flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, "S", manual	Low pressurizer pressure, low compensated steam line pressure, low T_{cold}, low-2 pressurizer level	Core makeup tank, feedwater isolation valves, main steam isolation valves (MSIVs), startup feedwater isolation, accumulators
Steam system piping failure	Power range high flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, "S", manual	Low pressurizer pressure, low compensated steam line pressure, high-2 containment pressure, low T_{cold}, manual	Core makeup tank, feedwater isolation valves, main steam line isolation valves (MSIVs), accumulators, startup feedwater isolation
Inadvertent operation of the PRHR	PRHR discharge valve position	Low pressurizer pressure, low T_{cold}, low-2 pressurizer level	Core makeup tank

Table 15.0-6 (Sheet 2 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.2			
Decrease in heat removal by the secondary system			
Loss of external load/turbine trip	High pressurizer pressure, high pressurizer water level, overtemperature ΔT, overpower ΔT, Steam generator low narrow range level, low RCP speed, manual	–	Pressurizer safety valves, steam generator safety valves
Loss of nonemergency ac power to the station auxiliaries	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, low RCP speed, manual	Steam generator low narrow range level coincident with low startup water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves
Loss of normal feedwater flow	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low startup water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves, reactor vessel head vent
Feedwater system pipe break	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, overtemperature ΔT, manual	Steam generator low narrow range level coincident with low startup feedwater flow, Steam generator low wide range level, low steam line pressure, high-2 containment pressure	PRHR, core makeup tank, MSIVs, feedline isolation, pressurizer safety valves, steam generator safety valves

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Table 15.0-6 (Sheet 3 of 5) **PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS**			
Incident	**Reactor Trip Functions**	**ESF Actuation Functions**	**ESF and Other Equipment**
Section 15.3			
Decrease in reactor coolant system flow rate			
Partial and complete loss of forced reactor coolant flow	Low flow, underspeed, manual	–	Steam generator safety valves, pressurizer safety valves
Reactor coolant pump shaft seizure (locked rotor)	Low flow, high pressurizer pressure, manual	–	Pressurizer safety valves, steam generator safety valves
Section 15.4			
Reactivity and power distribution anomalies			
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Source range high neutron flux, intermediate range high neutron flux, power range high neutron flux (low setting), power range high neutron flux (high setting), high nuclear flux rate, manual	–	–
Uncontrolled RCCA bank withdrawal at power	Power range high neutron flux, high power range positive neutron flux rate, overtemperature ΔT, over-power ΔT, high pressurizer pressure, high pressurizer water level, manual	–	Pressurizer safety valves, steam generator safety valves
RCCA misalignment	Overtemperature ΔT, low pressurizer pressure, manual	–	–
Startup of an inactive reactor coolant pump at an incorrect temperature	Power range high flux, low flow (P-10 interlock), manual	–	–

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Table 15.0-6 (Sheet 4 of 5) **PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS**			
Incident	**Reactor Trip Functions**	**ESF Actuation Functions**	**ESF and Other Equipment**
Section 15.4 (Cont.)			
Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, power range high flux, overtemperature ΔT, manual	Source range flux doubling	CVS to RCS isolation valves, makeup pump suction isolation valves, from the demineralized water transfer and storage system
Spectrum of RCCA ejection accidents	Power range high flux, high positive flux rate, manual	–	Pressurizer safety valves
Section 15.5			
Increase in reactor coolant inventory			
Inadvertent operation of the CMT during power operation	High pressurizer pressure, manual, "safeguards" trip, high pressurizer level	High pressurizer level, low T_{cold}	Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR, steam generator safety valves, reactor vessel head vent
Chemical and volume control system malfunction that increases reactor coolant inventory	High pressurizer pressure, "safeguards" trip, high pressurizer level, manual	High pressurizer level, low T_{cold}, low steam line pressure	Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR, reactor vessel head vent
Section 15.6			
Decrease in reactor coolant inventory			
Inadvertent opening of a pressurizer safety valve or ADS path	Low pressurizer pressure, overtemperature ΔT, manual	Low pressurizer pressure	Core makeup tank, accumulator

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Table 15.0-6 (Sheet 5 of 5)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.6 (Cont.)			
Failure of small lines carrying primary coolant outside containment	–	Manual isolation of the Sample System or CVS discharge lines	Sample System isolation valves, Chemical and volume control system discharge line isolation valves
Steam generator tube rupture	Low pressurizer pressure, overtemperature ΔT, safeguards ("S"), manual	Low pressurizer pressure, high-2 steam generator water level, high steam generator level coincident with reactor trip (P-4), low steam line pressure, low pressurizer level	Core makeup tank, PRHR, steam generator safety and/or relief valves, MSIVs, radiation monitors (air removal, steam line, and steam generator blowdown), startup feedwater isolation, chemical and volume control system pump isolation, pressurizer heater isolation, steam generator power-operated relief valve isolation
LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure, safeguards ("S"), manual	High-2 containment pressure, low pressurizer pressure	Core makeup tank, accumulator, ADS, steam generator safety and/or relief valves, PRHR, in-containment water storage tank (IRWST)

Table 15.0-7 (Sheet 1 of 2) **SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES**	
Event Description	**Failure**
Feedwater temperature reduction[(a)]	–
Excessive feedwater flow	One protection division
Excessive steam flow[(a)]	
Inadvertent secondary depressurization	One core makeup tank discharge valve
Steam system piping failure	One core makeup tank discharge valve (zero power case) One protection division (full power case)
Inadvertent operation of the PRHR	One protection division
Steam pressure regulator malfunction[(b)]	–
Loss of external load	One protection division
Turbine trip	One protection division
Inadvertent closure of main steam isolation valve	One protection division
Loss of condenser vacuum	One protection division
Loss of ac power	One PRHR discharge valve
Loss of normal feedwater	One PRHR discharge valve
Feedwater system pipe break	One PRHR discharge valve
Partial loss of forced reactor coolant flow	One protection division
Complete loss of forced reactor coolant flow	One protection division
Reactor coolant pump locked rotor	One protection division
Reactor coolant pump shaft break	One protection division
RCCA bank withdrawal from subcritical	One protection division
RCCA bank withdrawal at power	One protection division
Dropped RCCA, dropped RCCA bank	One protection division
Statically misaligned RCCA[(c)]	–
Single RCCA withdrawal	One protection division

Notes:

a. No protection action required

b. Not applicable to AP1000

c. No transient analysis

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Table 15.0-7 (Sheet 2 of 2) **SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES**	
Event Description	**Failure**
Flow controller malfunction(b)	–
Uncontrolled boron dilution	One protection division
Improper fuel loading(c)	–
RCCA ejection	One protection division
Inadvertent CMT operation at power	One PRHR discharge valve
Increase in reactor coolant system inventory	One PRHR discharge valve
Inadvertent reactor coolant system depressurization	One protection division
Failure of small lines carrying primary coolant outside containment(c)	–
Steam generator tube rupture	Ruptured steam generator power-operated relief valve fails open
Spectrum of LOCA Small breaks Large breaks	 One ADS Stage 4 valve One CMT valve
Long-term cooling	One ADS Stage 4 valve

Notes:

a. No protection action required

b. Not applicable to AP1000

c. No transient analysis

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Table 15.0-8 **NONSAFETY-RELATED SYSTEM AND EQUIPMENT USED FOR MITIGATION OF ACCIDENTS**	
Event	**Nonsafety-related System and Equipment**
15.1.2 Feedwater system malfunctions that result in an increase in feedwater flow	Main feedwater pump trip
15.1.4 Inadvertent opening of a steam generator relief or safety valve	MSIV backup valves[1] Main steam branch isolation valves
15.1.5 Steam system piping failure	MSIV backup valves[1] Main steam branch isolation valves
15.2.7 Loss of normal feedwater	Pressurizer heater block
15.5.1 Inadvertent operation of the core makeup tanks during power operation	Pressurizer heater block
15.5.2 Chemical and volume control system malfunction that increases reactor coolant inventory	Pressurizer heater block
15.6.2 Failure of small lines carrying primary coolant outside containment	Sample line isolation valves
15.6.3 Steam generator tube rupture	Pressurizer heater block MSIV backup valves(1) Main steam branch isolation valves
15.6.5 Small-break LOCA	Pressurizer heater block

Note:

1. These include the turbine stop or control valves, the turbine bypass valves, and the moisture separator reheater 2nd stage steam isolation valves.

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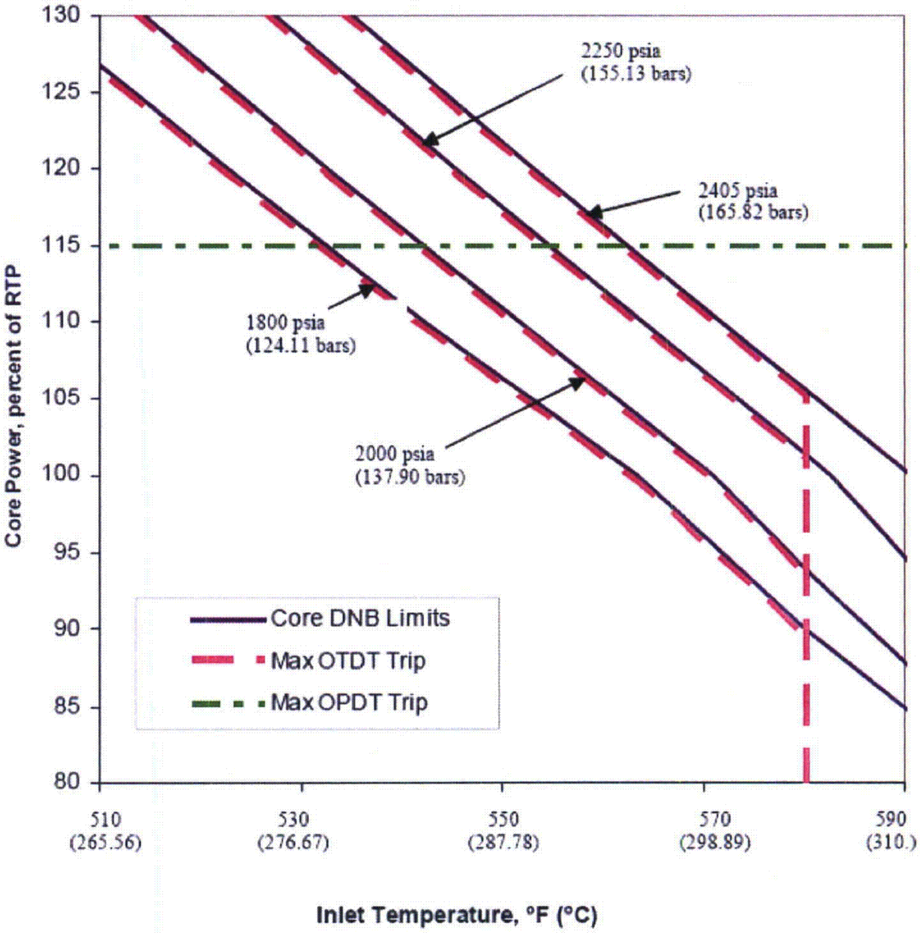

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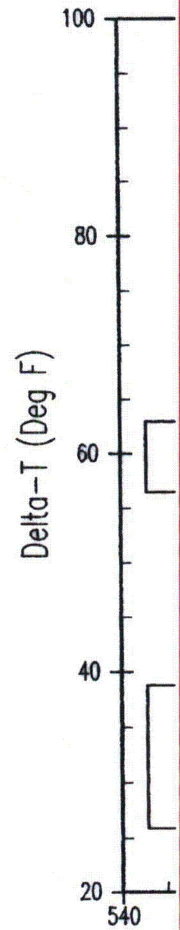

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Figure 15.0.3-1

Overpower and Overtemperature ΔT Protection

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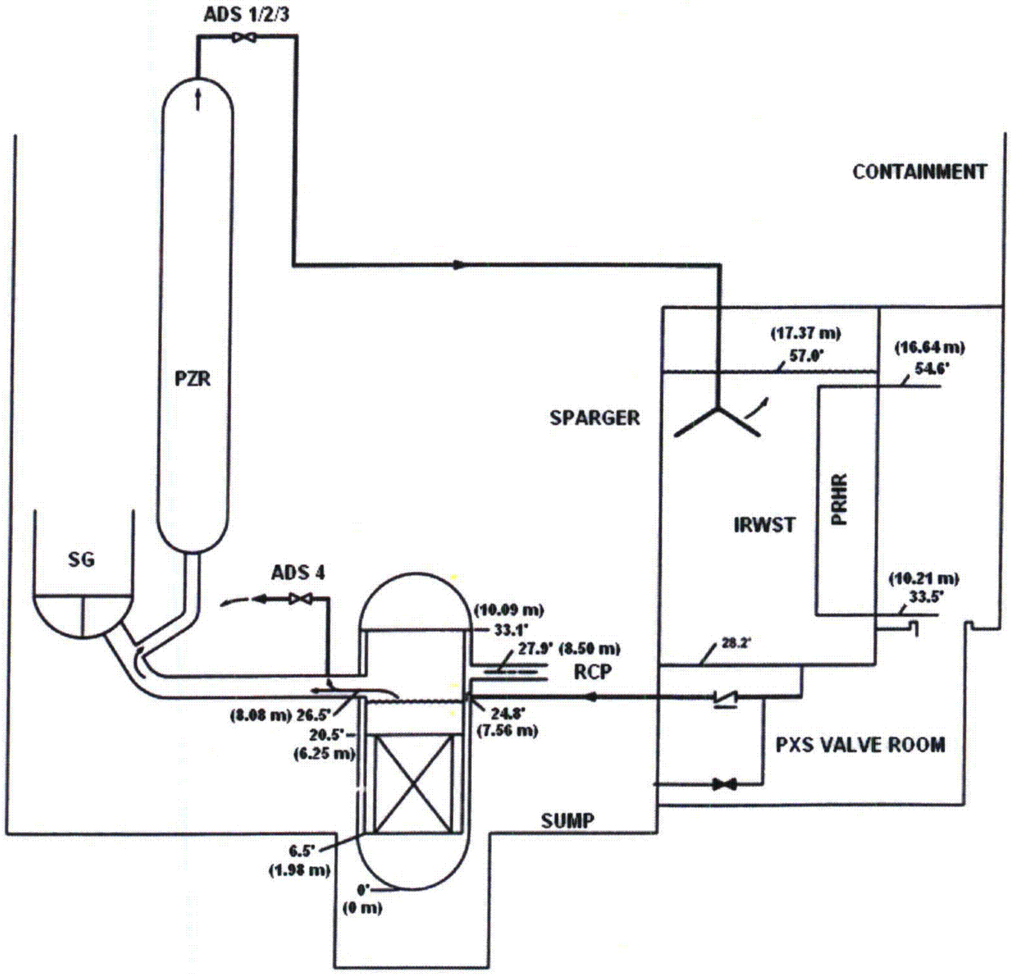

Note: All elevations are relative to the bottom inside surface of the Reactor Vessel

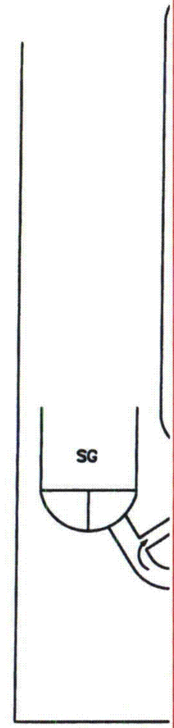

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Figure 15.0.3-2

AP1000 Loop Layout

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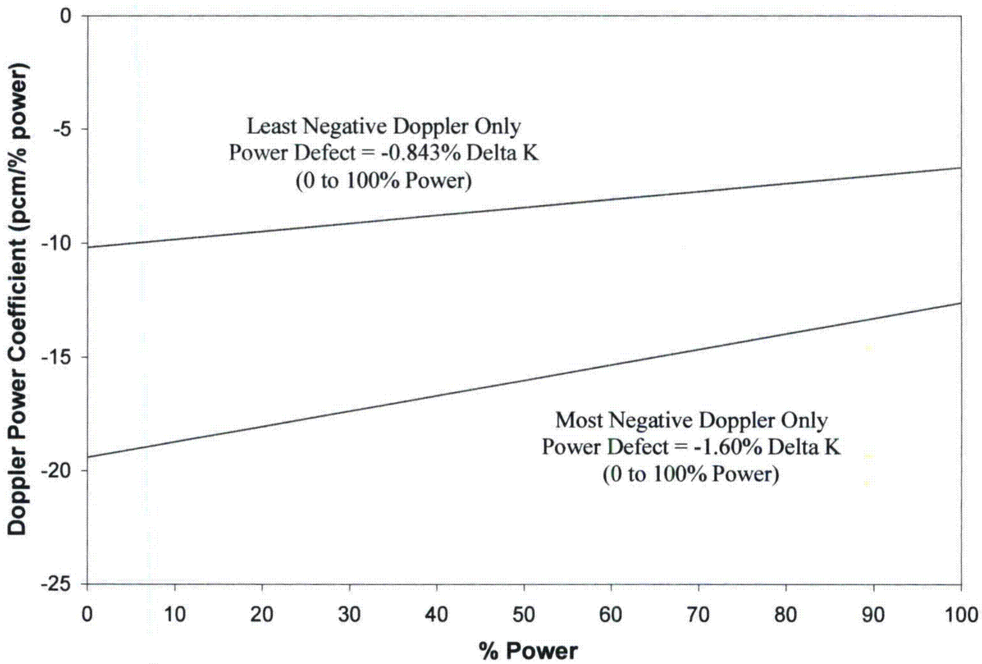
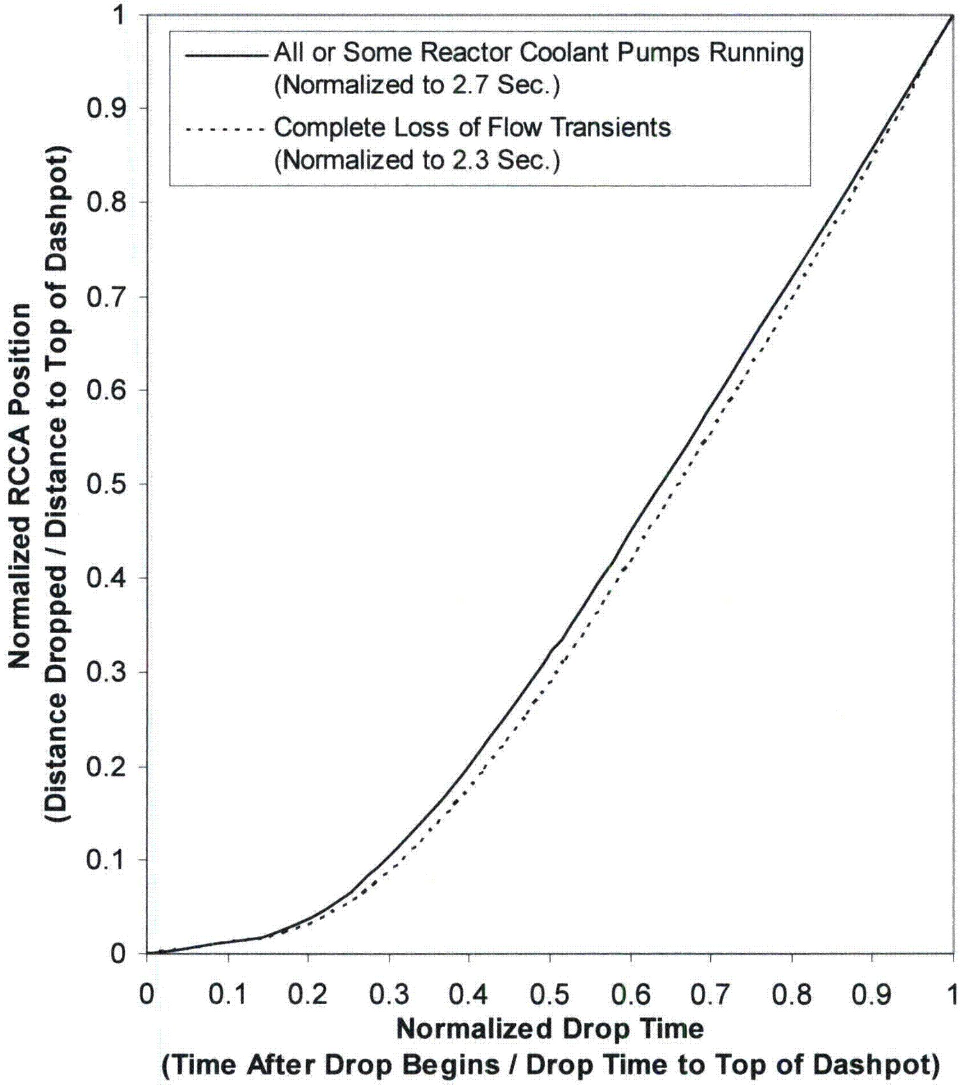


Figure 15.0.4-1

Doppler Power Coefficient used in Accident Analysis



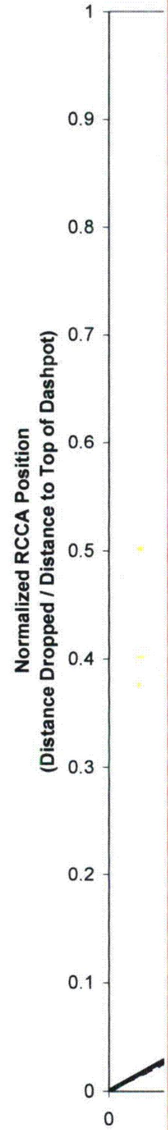


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Figure 15.0.5-1

RCCA Position Versus Time to Dashpot

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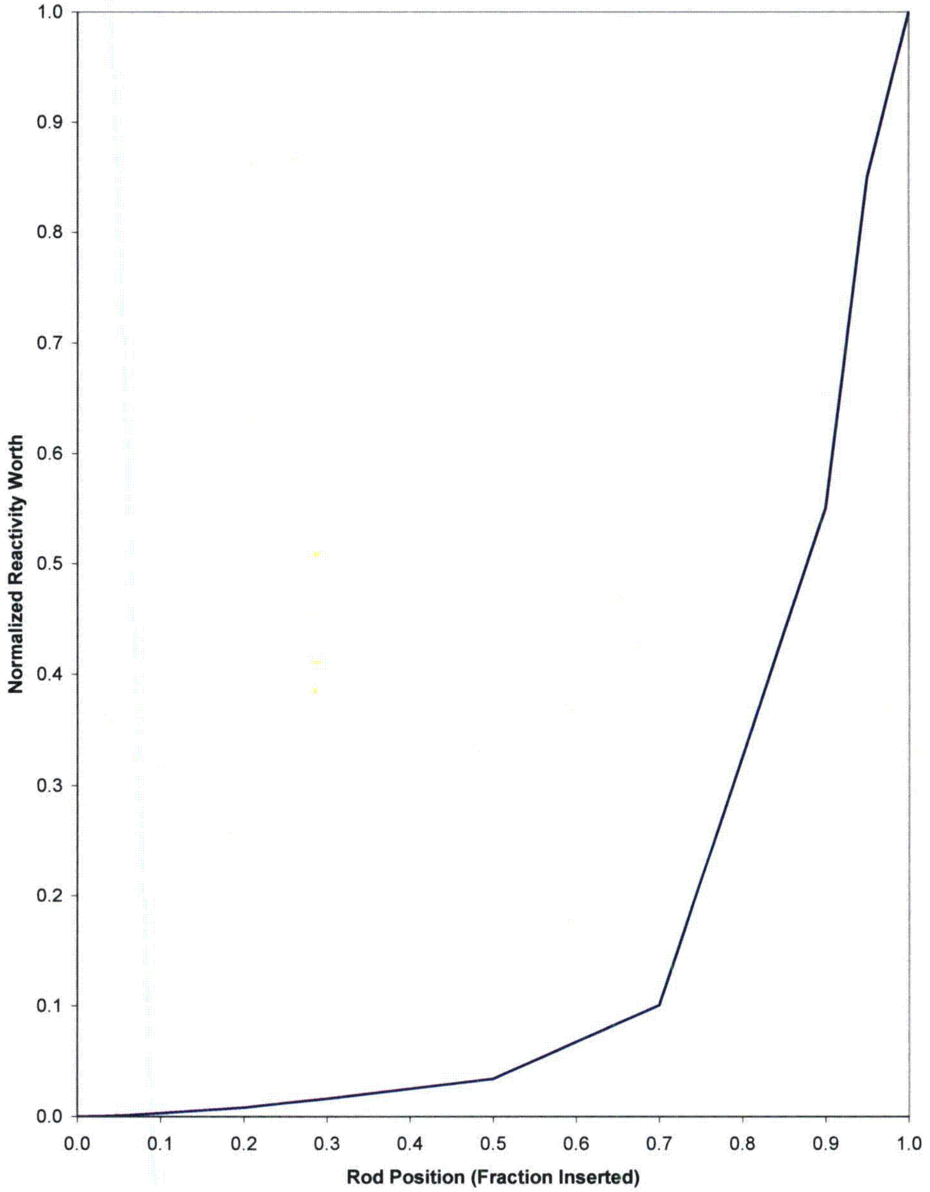
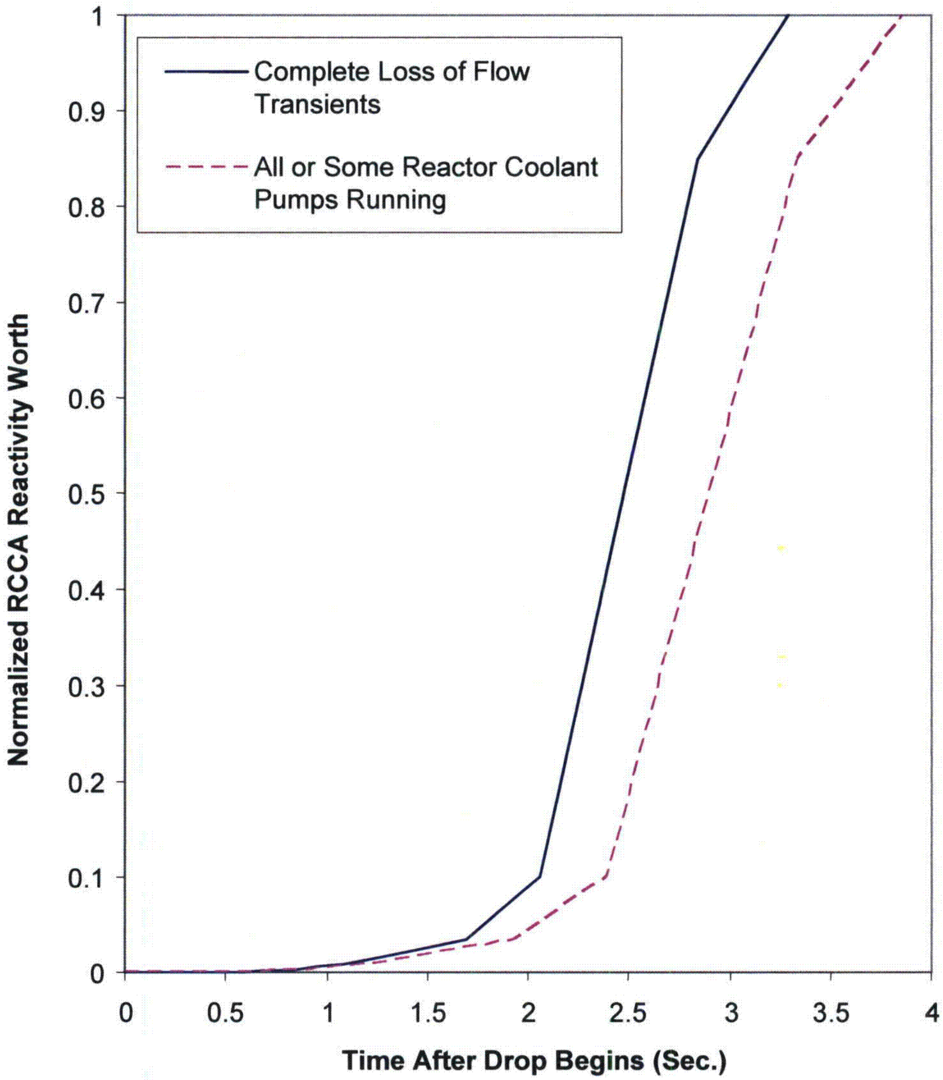


Figure 15.0.5-2

Normalized Rod Worth Versus Position



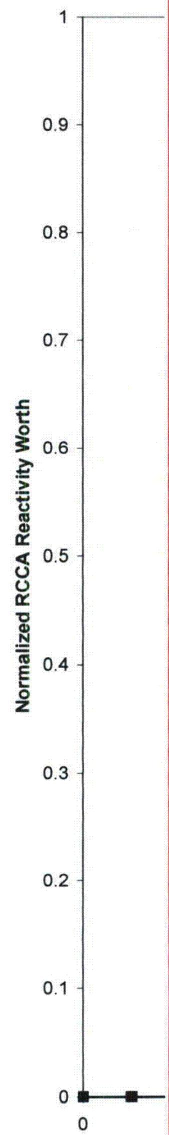


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Figure 15.0.5-3

Normalized RCCA Bank
Reactivity Worth Versus Drop Time

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AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 15 Section 15.1	Change Summary Description
[15.1-1]	15.1.1, Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.1-2]	15.1.2, Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients. The analysis was also updated and expanded to include flows increases to both steam generators.
[15.1-3]	15.1.3, Excessive Increase in Secondary Steam Flow	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increase MSSV inlet piping diameter (increased 1.2 inches), increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.1-4]	15.1.4, Inadvertent Opening of a Steam Generator Relief or Safety Valve	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increase MSSV inlet piping diameter (increased 1.2 inches), increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.1-5]	15.1.5, Steam System Piping Failure	The following changes were incorporated in the updated analysis: increased $F_{\Delta}H$ limit (1.65 to 1.72), addition of the flow skirt, increased lower core support plate flow hole size, increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the safety analysis and the updated valve, nozzle and piping pressure loss coefficients.
[15.1-6]	15.1.5.4, Steam System Piping Failure – Radiological Consequences	Editorial Changes. It is more accurate to describe the initial iodine and noble gas primary coolant concentrations as based on their respective technical specifications (i.e. equilibrium operating limits) because the technical specification limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. The doses were revised based on updated analysis.
[15.1-7]	15.1.5.5 , Steam System Piping Failure at Full Power	Although considered, earlier licensing submittals did not present transient analyses for steam system piping failures initiated from at power conditions. The results of analyses for steam system piping failures initiated from full power have been added.

Change No.	Chapter 15 Section 15.1	Change Summary Description
[15.1-8]	15.1.6, Inadvertent Operation of the PRHR Heat Exchanger	Editorial changes incorporated.
[15.1-9]	15.1.8 References	Updated WCAP-9226 - consistent with the current version used in the Chapter 15 analysis for the Advanced First Core design
[15.1-10]	15.1.8 References	Updated WCAP-15644 consistent with the current version used in the Chapter 15 analysis for the Advanced First Core design
[15.1-11]	15.1.8 References	Added new references WCAP-14565 and WCAP-15306 – consistent with the change to Section 15.1.4.2.1
[15.1-12]	Table 15.1.5-1	Spike duration recalculated based on revised source terms. Faulted and intact SG mass release data updated based on updated values modeled in the analysis.

15.1 Increase in Heat Removal From the Primary System

A number of events that could result in an increase in heat removal from the reactor coolant system are postulated. Detailed analyses are presented for the events that have been identified as limiting cases.

Discussions of the following reactor coolant system cooldown events are presented in this section:

- Feedwater system malfunctions causing a reduction in feedwater temperature
- Feedwater system malfunctions causing an increase in feedwater flow
- Excessive increase in secondary steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure
- Inadvertent operation of the passive residual heat removal (PRHR) heat exchanger

The preceding events are Condition II events, with the exception of small steam system piping failures, which are considered to be Condition III, and large steam system piping failure Condition IV events. Subsection 15.0.1 contains a discussion of classifications and applicable criteria.

The accidents in this section are analyzed. The most severe radiological consequences result from the main steam line break accident discussed in subsection 15.1.5. The radiological consequences are reported only for that limiting case.

15.1.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

Comment [B1]: [15.1.-1]

15.1.1.1 Identification of Causes and Accident Description

Reductions in feedwater temperature cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents a power increase that could lead to a departure from nucleate boiling ratio (DNBR) that is less than the design limit values.

A reduction in feedwater temperature may be caused by a low-pressure heater train or a high-pressure heater train out of service or bypassed. At power, this increased subcooling creates an increased load demand on the reactor coolant system.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in reactor coolant system temperature and a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. However, the rate of energy change is reduced as load and

feedwater flows decrease, so the no-load transient is less severe than the full-power case. The net effect on the reactor coolant system due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow; that is, the reactor reaches a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

A decrease in normal feedwater temperature is classified as a Condition II event, an incident of moderate frequency.

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in subsection 15.0.8 and listed in Table 15.0-6.

15.1.1.2 Analysis of Effects and Consequences

15.1.1.2.1 Method of Analysis

This transient is analyzed by calculating conditions at the feedwater pump inlet following the removal of a low-pressure feedwater heater train from service. These feedwater conditions are then used to recalculate a heat balance through the high-pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

- Initial plant power level corresponding to 100-percent nuclear steam supply system thermal output.
- The worst single failure in the pre-heating section of the Main Feedwater System, resulting in the maximum reduction in feedwater temperature, occurs.

Plant characteristics and initial conditions are further discussed in subsection 15.0.3.

15.1.1.2.2 Results

A fault in the feedwater heaters section of the Feedwater System causes a reduction in feedwater temperature that increases the thermal load on the primary system. The maximum reduction in feedwater enthalpy, due to a single failure in the feedwater system, is 49.98 Btu/lbm. This value is bounded by the enthalpy reduction associated with the Excessive Increase in Secondary Steam Flow event described in Section 15.1.3.

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15.1.1.3 Conclusions

The decrease in feedwater temperature transient is bounded by the Excessive Increase in Secondary Steam Flow event. Based on the results presented in subsection 15.1.3, the applicable Standard Review Plan subsection 15.1.1 evaluation criteria for the decrease in feedwater temperature event are met.

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15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

Addition of excessive feedwater causes an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and the reactor coolant system. The overpower/overtemperature protection (neutron overpower, overtemperature, and overpower ΔT trips) prevents a power increase that leads to a DNBR less than the safety analysis limit value.

An example of excessive feedwater flow is a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes an increased load demand on the reactor coolant system due to increased subcooling in the steam generator.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in reactor coolant system temperature and a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

Continuous addition of excessive feedwater is prevented by the steam generator high-2 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps, and reactor.

An increase in normal feedwater flow is classified as a Condition II event, fault of moderate frequency.

Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is assumed to occur as a consequence of the turbine trip for the excessive feedwater flow case initiated from full-power conditions. As discussed in subsection 15.0.14, an excessive feedwater flow transient initiated with the plant at no-load conditions need not consider a consequential loss of offsite power. With the plant initially at zero-load, the turbine would not have been connected

to the grid, so any subsequent reactor or turbine trip would not disrupt the grid and produce a consequential loss of offsite ac power.

15.1.2.2 Analysis of Effects and Consequences

15.1.2.2.1 Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient primarily is analyzed by using the LOFTRAN computer code (Reference 1). LOFTRAN simulates a multiloop system, neutron kinetics, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

The transient is analyzed to demonstrate plant behavior if excessive feedwater addition occurs because of system malfunction or operator error that allows a feedwater control valve to open fully. The following four cases are analyzed assuming a conservatively large negative moderator temperature coefficient:

- Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions.
- Accidental opening of both feedwater control valves with the reactor just critical at zero load conditions.
- Accidental opening of one feedwater control valve with the reactor in manual and automatic rod control at full power.
- Accidental opening of both feedwater control valves with the reactor in manual and automatic rod control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

- For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 120 percent of nominal feedwater flow to one steam generator.
- For the feedwater control valve accident at zero-load condition, a feedwater control valve malfunction occurs, which results in a step increase in flow to one steam generator from 0 to 120 percent of the nominal full-load value for one steam generator.
- For the zero-load condition, feedwater temperature is at a conservatively low value of 248°F.

Deleted: For that portion of the feedwater malfunction transient that includes a primary coolant flow coastdown caused by the consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analysis. First the LOFTRAN code is used to predict the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient. The FACTRAN code (Reference 5) is then used to calculate the heat flux based on the nuclear power and flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

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

- The feedwater flow resulting from a fully open control valve is terminated by a steam generator high-2 level trip signal, which closes feedwater control and isolation valves and trips the main feedwater pumps, the turbine, and the reactor.

Plant characteristics and initial conditions are further discussed in subsection 15.0.3.

Normal reactor control systems are not required to function. The protection and safety monitoring system may function to trip the reactor because of overpower or high-2 steam generator water level conditions. No single active failure prevents operation of the protection and safety monitoring system. A discussion of anticipated transients without trip considerations is presented in Section 15.8.

The analysis assumes that the turbine trip during the case initiated from full power results in a consequential loss of offsite power that produces the coastdown of the reactor coolant pumps. As described in subsection 15.0.14, the loss of offsite power is modeled to occur 3.0 seconds after the turbine trip. The excessive feedwater flow analysis conservatively delays the start of rod insertion until 2.0 seconds after the reactor trip signal is generated. Turbine trip occurs 5.0 seconds following a reactor trip condition being reached. This delay is part of the AP1000 reactor trip system. Complete rod insertion occurs in less than 5 seconds such that the loss of offsite power has no impact on the feedwater malfunction analysis.

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15.1.2.2.2 Results

In the case of an accidental full opening of both feedwater control valves with the reactor at zero power and the preceding assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in subsection 15.4.1 for an uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition. Therefore, the results of the analysis are not presented here. If the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25-percent nominal full power.

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The full-power case (maximum reactivity feedback coefficients, automatic rod control, multi-loop malfunction) results in the greatest power increase. Assuming the rod control system to be in the manual control mode results in a slightly less severe transient.

When the steam generator water level in the faulted loop reaches the high-2 level setpoint, the feedwater control valves and feedwater isolation valves are automatically closed and the main

feedwater pumps are tripped. This prevents continuous addition of the feedwater. In addition, a turbine trip and a reactor trip are initiated.

Transient results show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor (see Figures 15.1.2-1 and 15.1.2-2). A new equilibrium condition is reached and all the plant parameters, except for the SG water level, remain almost constant. Following the turbine trip, the consequential loss of offsite power produces the reactor coolant system flow coastdown shown in Figure 15.1.2-3. The minimum DNBR is predicted to occur before the reactor trip and the reactor coolant pump coastdown caused by the loss of offsite power. The minimum DNBR predicted is 1.97, which is well above the design limit described in Section 4.4. Following the reactor trip, the plant approaches a stabilized and safe condition; standard plant shutdown procedures may then be followed to further cool down the plant.

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Because the power level rises by a maximum of about 8 percent above nominal during the excessive feedwater flow incident, the fuel temperature also rises until after reactor trip occurs. The core heat flux lags behind the neutron flux response because of the fuel rod thermal time constant. Therefore, the peak value does not exceed 118 percent of its nominal value (the assumed high neutron flux trip setpoint). The peak fuel temperature thus remains well below the fuel melting temperature.

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The transient results show that departure from nucleate boiling (DNB) does not occur at any time during the excessive feedwater flow incident. Thus, the capability of the primary coolant to remove heat from the fuel rods is not reduced and the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for this accident is shown in Table 15.1.2-1.

15.1.2.3 Conclusions

The results of the analysis show that the minimum DNBR encountered for an excessive feedwater addition at power is above the design limit value. The DNBR design basis is described in Section 4.4.

Additionally, the reactivity insertion rate that occurs at no-load conditions following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from subcritical condition analysis (see subsection 15.4.1).

15.1.3 Excessive Increase in Secondary Steam Flow

Comment [B3]: [15.1-3]

15.1.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) results in a power mismatch between the reactor core power and the steam generator load demand. The plant control system is designed to accommodate a 10-percent step load increase or a 5-percent-per-minute ramp load increase in the range of 25- to 100-percent full power. Any loading rate in excess of these values may cause a reactor trip actuated by the protection and safety monitoring system. Steam flow increases greater than 10 percent are analyzed in subsections 15.1.4 and 15.1.5.

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

During power operation, turbine bypass to the condenser is controlled by reactor coolant condition signals. A high reactor coolant temperature indicates a need for turbine bypass. A single controller malfunction does not cause turbine bypass. An interlock blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following protection and safety monitoring system signals:

- Overpower ΔT
- Overtemperature ΔT
- Power range high neutron flux

The possible consequence of this accident (assuming no protective functions) is a departure from nucleate boiling (DNB) with subsequent fuel damage. Note that the accident is typically characterized by an approach of parameter values to the protection setpoints without the setpoints actually being reached. However, the reactor trip setpoints (high neutron flux, overpower ΔT, and overtemperature ΔT) could be reached during the analysis of the excessive load increase event. These protection functions are defeated in the analysis to preclude reactor trip, ensure the most severe DNB condition is reached, and demonstrate that the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

An excessive load increase incident is considered to be a Condition II event, as described in subsection 15.0.1.

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The requirements of GDC 17 of 10 CFR Part 50, Appendix A, which require determination of the effects produced by a possible consequential loss of offsite power during the excessive load increase event are not applicable. As discussed in subsection 15.0.14, the loss of offsite power

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need be considered only as a direct consequence of a turbine trip occurring while the plant is operating at power. For the four excessive load increase cases presented, reactor and turbine trips are not predicted to occur. However, even if a reactor trip were to occur, a consequential loss of ac power would not adversely impact the analysis results. This conclusion is based on a review of the time sequence of events associated with a consequential loss of ac power in comparison to the reactor shutdown time for the event. The primary effect of the loss of ac power is the coastdown of the Reactor Coolant Pumps (RCPs). The Protection & Safety Monitoring System (PMS) includes a five second minimum delay between the reactor trip and the turbine trip. In addition, a three second delay between the turbine trip and the loss of offsite ac power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of ac power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of ac power does not adversely impact this analysis because the plant will be shut down well before the RCPs begin to coast down.

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15.1.3.2 Analysis of Effects and Consequences

15.1.3.2.1 Method of Analysis

This accident is primarily analyzed using the LOFTRAN computer code (Reference 1). LOFTRAN simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate plant behavior following a 10-percent step load increase from rated load. These cases are as follows:

- Reactor control in manual with minimum moderator reactivity feedback
- Reactor control in manual with maximum moderator reactivity feedback
- Reactor control in automatic with minimum moderator reactivity feedback
- Reactor control in automatic with maximum moderator reactivity feedback

For the minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity; therefore, reductions in coolant temperature have the least impact on core power. For the maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value. This results in the largest amount of reactivity feedback due to changes in coolant temperature. For all the cases analyzed both with and without automatic rod control, no credit is taken for ΔT trips on overtemperature or overpower in order to demonstrate the inherent transient capability of the plant. Under actual operating conditions, such a trip may occur, after which the plant quickly stabilizes.

A 10-percent step increase in steam demand is assumed, and each case is analyzed without credit being taken for pressurizer heaters. At initial reactor power, reactor coolant system pressure and temperature are assumed to be at their full power values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A (Reference 2). Plant characteristics and initial conditions are further discussed in subsection 15.0.3.

Normal reactor control systems and engineered safety systems are not required to function.

15.1.3.2.2 Results

Figures 15.1.3-1 through 15.1.3-10 show the transient with the reactor in the manual control mode and no reactor trip signals occur. At the beginning of the minimum moderator feedback case, there is a slight power increase and the average core temperature shows a large decrease. This results in a DNBR that increases above its initial value. At the beginning of the maximum moderator feedback manually controlled case, there is a much faster increase in reactor power due to the moderator feedback. A reduction in the DNBR occurs, but the DNBR remains above the design limit (see Section 4.4).

Figures 15.1.3-11 through 15.1.3-20 show the transient assuming the reactor is in the automatic control mode. At the beginning of the maximum moderator feedback case, the core power increases and the coolant average temperature and pressurizer pressure decrease slowly. For this case, no reactor trip signal is generated. For the minimum moderator feedback case, a reactor trip signal setpoint is reached but, conservatively, reactor trip is not credited. At the beginning of the minimum moderator feedback case, the core power increases but the coolant average temperature and pressurizer pressure decrease rapidly. For this case, the transients oscillate and eventually stabilize. For both of these cases, the minimum DNBR remains above the design limit (see Section 4.4).

The excessive load increase incident is an overpower transient for which the fuel temperature rises. Reactor trip is not credited in any of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Because DNB does not occur during the excessive load increase transients, the capability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

The calculated sequence of events for the excessive load increase cases with no reactor trip are shown in Table 15.1.2-1.

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15.1.3.3 Conclusions

The analysis presented in this subsection demonstrates that for a 10-percent step load increase, the DNBR remains above the design limit. The design basis for DNB is described in Section 4.4. The plant rapidly reaches a stabilized condition following the load increase.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Comment [B4]: [15.1-4]

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in subsection 15.1.5.

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

The analysis is performed to demonstrate that the following Standard Review Plan subsection 15.1.4 evaluation criterion is satisfied:

- Assuming the most reactive stuck RCCA, with offsite power available, and assuming a single failure in the engineered safety features system, there will be no consequential damage to the fuel or reactor coolant system after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve. This criterion is met by showing the DNB design basis is not exceeded.

Accidental depressurization of the secondary system is classified as a Condition II event as described in Section 15.0.1.2.

The following systems provide the necessary protection against an accidental depressurization of the main steam system (see subsection 7.2.1.1.2):

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- Core makeup tank actuation from one of the following signals:
  - Safeguards ("S") signal from:
    - Two out of four low pressurizer pressure signals
    - Two out of four high-2 containment pressure signals
    - Two out of four low T_{cold} signals in any one loop or
    - Two out of four low steam line pressure signals in any one loop

– Two out of four low-2 pressurizer level signals

- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the "S" signal

- Redundant isolation of the main feedwater lines

  Sustained high feedwater flow causes additional cooldown. Therefore, in addition to the normal control action that closes the main feedwater control valves following reactor trip, an "S" signal rapidly closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.

- Redundant isolation of the startup feedwater system

  Sustained high startup feedwater flow causes additional cooldown. Therefore, the low T_{cold} signal closes the startup feedwater control and isolation valves.

- Trip of the fast-acting main steam line isolation valves (assumed to close in less than 10 seconds) on one of the following signals:

  – Two out of four low steam line pressure signals in any one loop (above permissive P-11)
  – Two out of four high negative steam pressure rates in any one loop (below permissive P-11)
  – Two out of four low T_{cold} signals in any one loop, or
  – Two out of four high-2 containment pressure signals

  Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0.6.

15.1.4.2 Analysis of Effects and Consequences

15.1.4.2.1 Method of Analysis

The analysis of a secondary system steam release is performed to determine the following:

- The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown, due to the steam release. The LOFTRAN code (References 1 and 6) is used to model the system transient.

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A full plant digital computer simulation using the LOFTRAN code (Reference 1)

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- The thermal-hydraulic behavior of the core due to the steam release. A detailed thermal-hydraulic digital computer code, VIPRE-01 (Reference 7), is used to determine if DNB occurs for the core transient conditions computed by the LOFTRAN code.

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The following conditions are assumed to exist at the time of a secondary system steam release:

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- End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive RCCA stuck in its fully withdrawn position. Operation of RCCA mechanical shim and axial offset banks during core burnup is restricted by the insertion limits so that shutdown margin requirements are satisfied.

- A most negative moderator temperature coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature is included. The k_{eff} (considering moderator temperature and density effects) versus temperature corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The core power is calculated as a function of core mass flow, core boron concentration, and core inlet temperature.

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- Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the passive core cooling system. There are no single failures that prevent core makeup tank injection, however, the analysis models the failure of one core makeup tank discharge valve. Low-concentration boric acid must be swept from the core makeup tank lines downstream of isolation valves before delivery of boric acid (3400 ppm) to the reactor coolant loops. This effect has been accounted for in the analysis.

- The case analyzed models a flow area of 0.2 ft^2, which is based on a steam flow of 520 pounds per second at 1200 psia with offsite power available. This conservatively bounds the maximum capacity of any single steam dump, relief, or safety valve.

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- Initial hot shutdown conditions at time zero are assumed because this represents the most conservative initial conditions. Should the reactor be just critical or operating at power at the time of a steam release, the reactor is tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the reactor coolant system contains more stored energy than at no-load. This is because the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. The additional stored energy is removed via the cooldown caused by the steam release before the no-load conditions of the reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy is removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis that assumes no-load condition at time zero. However, because the initial steam generator water inventory is

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greatest at no-load, the magnitude and duration of the reactor coolant system cooldown are less for a steam line release occurring at power:

- In computing the steam flow, the Moody Curve (Reference 3) for f(L/D) = 0 is used.
- Perfect moisture separation occurs in the steam generator.
- Offsite power is available, because this maximizes the cooldown.
- Maximum cold startup feedwater flow is assumed.
- Four reactor coolant pumps are initially operating.
- Manual actuation of the PRHR system at time zero is conservatively assumed to maximize the cooldown.

15.1.4.2.2 Results

The calculated sequence of events for the analyzed case is shown in Table 15.1.2-1. The results presented conservatively indicate the events that would occur assuming a secondary system steam release because it is postulated that the conditions described in subsection 15.1.4.2.1 exist simultaneously.

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Figures 15.1.4-2 through 15.1.4-12 show the transient results for the event. The steam release accounted for in the analysis is bounding compared to the capacity of any single steam dump, relief, or safety valve.

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Core makeup tank injection and the associated tripping of the reactor coolant pumps are initiated automatically by the low T_{cold} "S" signal. Boron solution at 3400 ppm enters the reactor coolant system, providing enough negative reactivity to prevent a significant return to power and core damage. Later in the transient, as the reactor coolant pressure continues to fall, the accumulators actuate and inject boron solution at 2600 ppm.

The transient is conservative with respect to cooldown, because no credit is taken for the energy stored in the system metal other than that of the fuel elements and steam generator tubes, and the PRHR system is assumed to be actuated at time zero. Because the limiting portion of the transient occurs over a period of about 5 minutes, the neglected stored energy would have a significant effect in slowing the cooldown.

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15.1.4.3 Margin to Critical Heat Flux

The analysis demonstrates that the DNB design basis, as described in Section 4.4, is met for the inadvertent opening of a steam generator relief or safety valve. As shown in Figure 15.1.4-2, no significant return to power occurs and, therefore, DNB does not occur. The minimum DNBR is conservatively calculated and is above the 95/95 limit.

15.1.4.4 Conclusions

The analysis shows that the criterion stated in this subsection is satisfied. For an inadvertent opening of any single steam dump or a steam generator relief or safety valve, the DNB design basis is met.

15.1.5 Steam System Piping Failure

Comment [B5]: [15.1-5]

15.1.5.1 Identification of Causes and Accident Description

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

If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the existing high-power peaking factors, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the passive core cooling system.

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The analysis of a main steam line rupture is performed to demonstrate that the following Standard Review Plan subsection 15.1.5 evaluation criterion is satisfied.

- Assuming the most reactive stuck RCCA with or without offsite power and assuming a single failure in the engineered safety features system, the core cooling capability is maintained. As shown in subsection 15.1.5.4, radiation doses are within the guidelines.

DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable. The following analysis shows that the DNB design basis is not exceeded for any steamline rupture, assuming the most reactive RCCA is stuck in its fully withdrawn position.

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A major steam line rupture is classified as a Condition IV event.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in subsection 15.0.1.3.

The major rupture of a steam line is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat retards the cooldown and thereby reduces the likelihood that the reactor returns to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented here. Certain assumptions used in this analysis are discussed in WCAP-9226-P-A (Reference 4). WCAP-9226-P-A also contains a discussion of the spectrum of break sizes and power levels analyzed.

The steam line rupture at full power conditions is explicitly analyzed and discussed in Section 15.1.5.5.

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The following functions provide the protection for a steam line rupture (see subsection 7.2.1.1.2):

- Core makeup tank actuation from one of the following:
  - Safeguards ("S") signal from:
    - Two out of four low pressurizer pressure signals
    - Two out of four high-2 containment pressure signals
    - Two out of four low T_{cold} signals in one loop, or
    - Two out of four low steam line pressure signals one loop
  - Two out of four low-2 pressurizer level signals

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- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the "S" signal

- Redundant isolation of the main feedwater lines

Sustained high feedwater flow causes additional cooldown. Therefore, in addition to the normal control action that closes the main feedwater control valves following reactor trip, an "S" signal rapidly closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.

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- Redundant isolation of the startup feedwater system

Sustained high startup feedwater flow causes additional cooldown. Therefore, the low T_{cold} signal closes the startup feedwater control and isolation valves.

- Trip of the fast-acting main steam line isolation valves (assumed to close in less than 10 seconds) on one of the following signals:

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- Two out of four low steam line pressure signals in any one loop (above permissive P-11)
- Two out of four high negative steam pressure rates in any one loop (below permissive P-11)
- Two out of four low T_{cold} signals in any one loop, or
- Two out of four high-2 containment pressure signals.

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A fast-acting main steam isolation valve is provided in each steam line. These valves are assumed to fully close within 10 seconds of actuation following a large break in the steam line. For breaks downstream of the main steam line isolation valves, closure of the isolation valves will terminate the blowdown. For any break in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the main steam line isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

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Flow restrictors are installed in the steam generator outlet nozzle, as an integral part of the steam generator. The effective throat area of the nozzles is 1.4 ft^2, which is considerably less than the main steam pipe area; thus, the flow restrictors serve to limit the maximum steam flow for a break at any location.

Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

15.1.5.2.1 Method of Analysis

The analysis of the steam pipe rupture is performed to determine the following:

- The core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code (References 1 and 6) is used to model the system transient.

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- The thermal-hydraulic behavior of the core following a steam line break. A detailed thermal-hydraulic digital computer code, VIPRE-01 (Reference 7), is used to determine if DNB occurs for the core transient conditions computed by the LOFTRAN code.

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The following conditions are assumed to exist at the time of a main steam line break accident:

- End-of-cycle shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of RCCA mechanical shim and axial offset banks during core burnup is restricted by the insertion limits so that shutdown margin requirements are satisfied.

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- A most negative moderator temperature coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature is included. The k_{eff} (considering moderator temperature and density effects) versus temperature corresponding to the negative moderator temperature coefficient used is shown in Figure 15.1.4-1. The core power is calculated as a function of core mass flow, core boron concentration, and core inlet temperature.

  The moderator properties used in the LOFTRAN code for feedback calculations are generated by combining those in the sector nearest the affected steam generator with those associated with the remaining sector. The resultant properties reflect a combination process that accounts for inlet plenum fluid mixing and a conservative weighting of the fluid properties from the coldest core sector.

  In verifying the conservatism of this method, the power predictions of the LOFTRAN modeling are confirmed by comparison with detailed core analysis for the limiting conditions of the cases considered. This core analysis conservatively models the hypothetical core configuration (that is, stuck RCCA, non-uniform inlet temperatures, pressure, flow, and boron concentration) and directly evaluates the total reactivity feedback including power, boron, and density redistribution in an integral fashion. The effect of void formation is also included.

  Comparison of the results from the detailed core analysis with the LOFTRAN predictions verifies the overall conservatism of the methodology. That is, the specific power, temperature, and flow conditions used to perform the DNB analysis are conservative.

- Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the passive core cooling system. The core makeup tanks and the accumulators are the portions of the passive core cooling system used in mitigating a steam line rupture. There are no single failures that prevent core makeup tank injection, however, the analysis models the failure of one core makeup tank discharge valve. Low-concentration boric acid must be swept from the core makeup tank lines downstream of isolation valves before delivery of boric acid (3400 ppm) to the reactor coolant loops. This effect has been accounted for in the analysis.

- The maximum overall fuel-to-coolant heat transfer coefficient is used to maximize the rate of cooldown.

- Because the steam generators are provided with integral flow restrictors with a 1.4-ft^2 throat area, any rupture in a steam line with a break area greater than 1.4 ft^2, regardless of location, has the same effect on the primary plant as the 1.4-ft^2 double-ended rupture. The limiting case considered in determining the core power and reactor coolant system transient is the

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complete severance of a pipe, with the plant initially at no-load conditions and full reactor coolant flow with offsite power available. The results of this case bound the loss of offsite power case for the following reasons:

- Loss of offsite power results in an immediate reactor coolant pump coastdown at the initiation of the transient. This reduces the severity of the reactor coolant system cooldown by reducing primary-to-secondary heat transfer. The lessening of the cooldown, in turn, reduces the magnitude of the return to power.

- Following its actuation, the core makeup tank provides borated water that injects into the reactor coolant system. Flow from the core makeup tank increases if the reactor coolant pumps have coasted down. Therefore, the analysis performed with offsite power and continued reactor coolant pump operation reduces the rate of boron injection into the core and is conservative.

- The protection system automatically provides a safety-related signal that initiates the coastdown of the reactor coolant pumps in parallel with core makeup tank actuation. Because this reactor coolant pump trip function is actuated early during the steam line break event (right after core makeup tank actuation), there is very little difference in the predicted DNBR between cases with and without offsite power.

- Because of the passive nature of the safety injection system, the loss of offsite power does not delay the actuation of the safety injection system.

- Power peaking factors corresponding to one stuck RCCA are determined at the end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck RCCA during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck RCCA. The power peaking factors depend upon the core power, temperature, pressure, and flow and, therefore, may differ for each case studied.

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- The analysis assumes initial hot standby conditions at time zero in order to present a representative case which will yield limiting post-trip DNBR results for this transient. If the reactor is just critical or operating at power at the time of a steam line break, the reactor is tripped by the overpower protection system when power level reaches a trip point.

  Following a trip at power, the reactor coolant system contains more stored energy than at no-load because the average coolant temperature is higher than at no-load, and there is energy stored in the fuel. The additional stored energy reduces the cooldown caused by the steam line break before the no-load conditions of reactor coolant system temperature and

shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis that assumes a no-load condition at time zero. However, because the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the reactor coolant system cooldown are less for a steam line break occurring at power.

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- In computing the steam flow during a steam line break, the Moody Curve (Reference 3) for f(L/D) = 0 is used.
- Perfect moisture separation occurs in the steam generator.
- Maximum cold startup feedwater flow plus nominal 100 percent main feedwater flow is assumed.
- Four reactor coolant pumps are initially operating.
- Manual actuation of the PRHR system at time zero is conservatively assumed in order to maximize the cooldown.

15.1.5.2.2 Results

The calculated sequence of events for the analyzed case is shown in Table 15.1.2-1. The results presented conservatively indicate the events that would occur assuming a steam line rupture because it is postulated that the conditions described in subsection 15.1.5.2.1 exist simultaneously.

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15.1.5.2.3 Core Power and Reactor Coolant System Transient

Figures 15.1.5-1 through 15.1.5-13 show the transient results following a main steam line rupture (complete severance of a pipe) at initial no-load condition.

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Offsite power is assumed available so that, initially, full reactor coolant flow exists. During the course of the event, the reactor protection system initiates a trip of the reactor coolant pumps in conjunction with actuation of the core makeup tanks. The transient shown assumes an uncontrolled steam release from only one steam generator. Steam release from more than one steam generator is prevented by automatic trip of the main steam isolation valves in the steam lines by low steam line pressure signals. Even with the failure of one valve, release is limited to approximately 10 seconds for the other steam generator while the one generator blows down. The main steam isolation valves fully close in less than 10 seconds from receipt of a closure signal.

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As shown in Figure 15.1.5-1, the core attains criticality with the RCCAs inserted (with the design shutdown assuming the most reactive RCCA stuck) before boron solution at 3400 ppm (from core makeup tanks) or 2600 ppm (from accumulators) enters the reactor coolant system. A peak core power significantly lower than the nominal full-power value is attained.

The calculation assumes that the boric acid is mixed with and diluted by the water flowing in the reactor coolant system before entering the reactor core. The concentration after mixing depends upon the relative flow rates in the reactor coolant system and from the core makeup tanks or accumulators (or both). The variation of mass flow rate in the reactor coolant system due to water density changes is included in the calculation. The variation of flow rate from the core makeup tanks or accumulators (or both) due to changes in the reactor coolant system pressure and temperature and the pressurizer level is also included. The reactor coolant system and passive injection flow calculations include line losses.

At no time during the analyzed steam line break event does the core makeup tank level approach the setpoint for actuation of the automatic depressurization system. During non-LOCA events, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tank is offset by an equal volume of recirculation flow that enters the core makeup tanks via the reactor coolant system cold leg balance lines.

The PRHR system provides a passive, long-term means of removing the core decay and stored heat by transferring the energy via the PRHR heat exchanger to the in-containment refueling water storage tank (IRWST). The PRHR heat exchanger is normally actuated automatically when the steam generator level falls below the low wide-range level. For the main steam line rupture case analyzed, the PRHR exchanger is conservatively actuated at time zero to maximize the cooldown.

15.1.5.2.4 Margin to Critical Heat Flux

The case analyzed conservatively models the expected behavior of the plant during a steam system piping failure. This includes the tripping of the reactor coolant pumps coincident with core makeup tank actuation. A DNB analysis was performed using limiting assumptions that bound those of subsection 15.1.5.2.1.

Under the low flow (natural circulation) conditions present in the transient, the return to power is severely limited by the large negative feedback due to flow and power. The minimum DNBR is conservatively calculated and remains above the 95/95 limit.

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15.1.5.3 Conclusions

DNB and possible cladding perforation are not unacceptable consequences following a steam pipe rupture based on the applicable acceptance criteria. Nevertheless, the preceding analysis shows that no DNB, and therefore no cladding perforation, occurs for the main steam line rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

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15.1.5.4 Radiological Consequences

The evaluation of the radiological consequences of a postulated main steam line break outside containment assumes that the reactor has been operating with a limited number of fuel rods containing cladding defects) and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant. See Section 15.1.5.4.1 and Table 15.1.5-1.

Comment [B6]: [15.1-6]

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Following the rupture, startup feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Any radioiodines carried from the primary coolant into the faulted steam generator via leaking tubes are assumed to be released directly to the environment. It is conservatively assumed that the reactor is cooled by steaming from the intact loop.

15.1.5.4.1 Source Term

The only significant radionuclide releases due to the main steam line break are the iodines and alkali metals that become airborne and are released to the environment as a result of the accident. Noble gases are also released to the environment. Their impact is secondary because any noble gases entering the secondary side during normal operation are rapidly released to the environment.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs prior to the accident and that the maximum resulting reactor coolant iodine concentration exists at the time the accident occurs.

The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The reactor coolant alkali metal concentrations are assumed to be those associated with the design basis fuel defect level

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The secondary coolant is assumed to have an iodine source term of 0.1 μCi/g dose equivalent I-131. This is 10 percent of the maximum primary coolant activity at equilibrium operating

conditions. The secondary coolant alkali metal concentration is also assumed to be 10 percent of the primary concentration.

15.1.5.4.2 Release Pathways

There are three components to the accident releases:

- The secondary coolant in the steam generator of the faulted loop is assumed to be released out the break as steam. Any iodine and alkali metal activity contained in the coolant is assumed to be released.
- The reactor coolant leaking into the steam generator of the faulted loop is assumed to be released to the environment without any credit for partitioning or plateout onto the interior of the steam generator.
- The reactor coolant leaking into the steam generator of the intact loop would mix with the secondary coolant and thus raise the activity concentrations in the secondary water. While the steam release from the intact loop would have partitioning of non-gaseous activity, this analysis conservatively assumes that any activity entering the secondary side is released.

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.1.5.4.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.1.5.4.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.1.5-1.

15.1.5.4.5 Identification of Conservatisms

The assumptions and parameters used in the analysis contain a number of significant conservatisms:

- The reactor coolant activities are based on conservative assumptions (see Table 15.1.5-1). The activities based on the expected fuel defect level are far less than this (see Section 11.1).
- The assumed leakage of 150 gallons of reactor coolant per day into each steam generator is conservative. The leakage is expected to be a small fraction of this during normal operation.

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- The conservatively selected meteorological conditions are present only rarely.

15.1.5.4.6 Doses

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be 0.5 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 1.3 rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be 0.5 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.4 rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

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At the time the main steam line break occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE. When this is added to the dose calculated for the main steam line break, the resulting total dose remains less than the values reported above.

15.1.5.5 Steam System Piping Failure at Full Power

Comment [B7]: [15.1-7]

15.1.5.5.1 Identification of Causes and Accident Description

A rupture in the main steam system piping from an at-power condition creates an increased steam load, which extracts an increased amount of heat from the reactor coolant system via the steam generators. This results in a reduction in reactor coolant system temperature and pressure. In the presence of a strong negative moderator temperature coefficient, typical of end-of-life conditions, the colder core inlet coolant temperature causes the core power to increase from its initial level due to the positive reactivity insertion. The power approaches a level equal to the total steam flow.

Depending upon the break size, the reactor may be tripped on any of the following trip signals to provide the necessary protection against the rupture of a main steam line.

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- Overpower ΔT
- Low pressurizer pressure

- Safeguards ("S') actuation signal
  - low steam line pressure
  - low cold leg temperature

The steam system piping failure accident analysis described in subsection 15.1.5 is performed assuming a hot zero power initial condition with the control rods inserted in the core, except for the most reactive rod in the fully withdrawn position, out of the core. That condition could occur while the reactor is at hot shutdown at the minimum required shutdown margin or after the plant has been tripped manually or by the reactor protection system following a steam line break from an at-power condition. For an at-power break, the analysis of subsection 15.1.5 represents the limiting condition with respect to core protection for the time period following reactor trip. The purpose of this section is to describe the analysis of a steam system piping failure occurring from an at power initial condition, to demonstrate that core protection is maintained prior to and immediately following reactor trip. The analysis initiated from hot full power does not extend into the portion of the transient where the PRHR or CMTs are actuated.

Depending on the size of the break, this event is classified as either an ANS Condition III or IV event.

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15.1.5.5.2 Analysis of Effects and Consequences

15.1.5.5.2.1 Method of Analysis

The analysis of the steam line rupture is performed in the following stages:

1. The LOFTRAN code (References 1 and 6) is used to calculate the nuclear power, core heat flux, and reactor coolant system temperature and pressure transients resulting from the cooldown following the steam line break.

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2. The core radial and axial peaking factors are determined using the thermal hydraulic conditions from LOFTRAN as input to the nuclear core models. A detailed thermal-hydraulic code, VIPRE-01 (Reference 7), is then used to calculate the DNBR for the limiting time during the transient.

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This accident is analyzed with the Revised Thermal Design Procedure (RTDP) as described in WCAP-11397-P-A (Reference 2).

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The following assumptions are made in the transient analysis:

1. Initial Conditions - RTDP DNB methodology was used, therefore the uncertainties in the initial conditions are included in the DNBR limits; thus, nominal full power values are used in LOFTRAN. The RCS Minimum Measured Flow is used.

2. Break Size – A spectrum of break sizes was analyzed. Small breaks do not result in a reactor trip. Intermediate breaks result in a reactor trip on overpower ΔT. Larger break sizes result in a reactor trip on low steam line pressure safeguards actuation.

3. Break flow – In computing the steam flow during a steam line break, the Moody curve (Reference 3) for fL/D = 0 is used.

4. Reactivity Coefficients – The analysis assumes maximum moderator reactivity feedback and minimum Doppler power feedback to maximize the power increase following the break.

5. Protection System – The protection system features that mitigate the effects of a steam line beak are described in subsection 15.1.5. This analysis only considers the initial phase of the transient initiated from an at-power condition. Protection in this phase of the transient is provided by reactor trip, if necessary (specifically overpower ΔT, and low steam line pressure safeguards actuation).

6. Control Systems – Control systems are not credited in the accident analysis unless their function would result in more severe consequences. The only control system that is assumed to function during the hot full power steam line break event is the main feedwater system. For this event, the feedwater flow is assumed to match the steam flow.

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

For those events where offsite ac power is lost, an appropriate time delay between turbine trip and the postulated loss of offsite ac power is assumed in the analyses. A time delay of 3 seconds is used. This time delay is based on the inherent stability of the offsite power grid. Following the time delay, the effect of the loss of offsite ac power on plant auxiliary equipment – such as reactor coolant pumps, main feedwater pumps, condenser, startup feedwater pumps, and RCCAs

– is considered in the analyses. Turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay is part of the reactor trip system and was chosen to allow the reactor to be tripped and have the rods inserted to the bottom of the core before a turbine trip signal. As a result, RCP coastdown would be delayed an additional 5 seconds, the control rods would be fully inserted and there would be no adverse DNB impact from the resulting core flow reduction. Thus, there is no need for an explicit analysis of this event with loss of offsite ac power.

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15.1.5.5.3 Results

A spectrum of steam line break sizes was analyzed from 0.1 ft^2 to 1.4 ft^2. The results show that for small break sizes up to and including 0.35 ft^2, a reactor trip is not generated. In this case, the event is similar to an excessive load increase event; the core reaches a new equilibrium condition at a higher power equivalent to the increased steam release. For break sizes from 0.36 ft^2 up to and including 0.87 ft^2, the reactor trips on overpower ΔT. For break sizes from 0.88 ft^2 to 1.4 ft^2 the reactor trips on the low steam line pressure safeguards actuation signal.

The limiting case for demonstrating DNB and kW/ft protection is the 0.87 ft^2 break, the largest break size that results in a trip on overpower ΔT. The time sequence of events for this case is shown on Table 15.1.5.5-1. Figures 15.1.5.5-1 through 15.1.5.5-7 show the transient response.

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15.1.5.5.4 Conclusions

The analysis shows that the DNB and fuel centerline melt (kW/ft) design bases are met for the limiting case. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily precluded by the criteria, the above analysis, in fact, shows that the minimum DNBR remains above the limit value for any rupture occurring from an at-power condition prior to and immediately following a reactor trip.

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15.1.6 Inadvertent Operation of the PRHR Heat Exchanger

Comment [B8]: [15.1-8]

15.1.6.1 Identification of Causes and Accident Description

The inadvertent actuation of the PRHR heat exchanger causes an injection of relatively cold water into the reactor coolant system. This produces a reactivity insertion in the presence of a negative moderator temperature coefficient. To prevent this reactivity increase from causing reactor power increase, a reactor trip is initiated when either PRHR discharge valve comes off of its fully shut seat.

The inadvertent actuation of the PRHR heat exchanger could be caused by operator error or a false actuation signal, or by malfunction of a discharge valve. Actuation of the PRHR heat

exchanger involves opening one of the isolation valves, which establishes a flow path from one reactor coolant system hot leg, through the PRHR heat exchanger, and back into its associated steam generator cold leg plenum.

The PRHR heat exchanger is located above the core to promote natural circulation flow when the reactor coolant pumps are not operating. With the reactor coolant pumps in operation, flow through the PRHR heat exchanger is enhanced. The heat sink for the PRHR heat exchanger is provided by the IRWST, in which the PRHR heat exchanger is submerged. Because the fluid in the heat exchanger is in thermal equilibrium with water in the tank, the initial flow out of the PRHR heat exchanger is significantly colder than the reactor coolant system fluid. Following this initial insurge, the reduction in cold leg temperature is limited by the cooling capability of the PRHR heat exchanger. Because the PRHR heat exchanger is connected to only one reactor coolant system loop, the cooldown resulting from its actuation is asymmetric with respect to the core.

The response of the plant to an inadvertent PRHR heat exchanger actuation with the plant at no-load conditions is bounded by the analyses performed for the inadvertent opening of a steam generator relief or safety valve event (subsection 15.1.4) and the steam system piping failure event (subsection 15.1.5). Both of these events are conservatively analyzed assuming PRHR heat exchanger actuation coincident with the steam line depressurization. Therefore, only the response of the plant to an inadvertent PRHR initiation with the core at power is considered.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of ac power during an inadvertent PRHR heat exchanger actuation event have been evaluated to not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of ac power in comparison to the reactor shutdown time for an inadvertent PRHR heat exchanger actuation event. The primary effect of the loss of ac power is to cause the Reactor Coolant Pumps (RCPs) to coast down. The PMS system includes a 5-second minimum delay between the reactor trip and the turbine trip. In addition, a 3-second delay between the turbine trip and the loss of offsite ac power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of ac power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of ac power does not adversely impact this inadvertent PRHR heat exchanger actuation analysis because the plant will be shut down well before the RCPs begin to coast down.

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The inadvertent actuation of the PRHR heat exchanger event is a Condition II event, a fault of moderate frequency. Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0.6. The following reactor protection

system functions are available to provide protection in the event of an inadvertent PRHR heat exchanger actuation:

- PRHR discharge valve not closed
- Overpower/overtemperature reactor trips (neutron flux and ΔT)
- Two out of four low pressurizer pressure signals

Due to the potential consequences as a result of the reactivity excursion, a reactor trip has been designed so that upon an inadvertent PRHR actuation, a reactor trip will occur. This reactor trip is generated when either of the discharge valves is not closed. This ensures that the reactor will be tripped prior to a power increase due to the cold water injection.

15.1.6.2 Analysis of Effects and Consequences

Since a reactor trip is initiated as soon as the PRHR discharge valves are not fully closed, this event is essentially a reactor trip from the initial condition and requires no separate transient analysis. Table 15.1.2-1 shows the sequence of events for the inadvertent PRHR heat exchanger actuation.

15.1.6.3 Conclusions

Inadvertent actuation of the PRHR does not result in violation of the core thermal design limits (DNB and linear power generation) or RCS overpressure.

15.1.7 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

15.1.8 References

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3. Moody, F. S., "Transactions of the ASME, Journal of Heat Transfer," Figure 3, page 134, February 1965.

4. **Wood, D. C., and Hollingsworth, S. D., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226-P-A, Revision 1 (Proprietary) and WCAP-9227 Revision 1 (Nonproprietary), Approved February 1998.**

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Comment [B9]: [15.1-9]

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5. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

6. "AP1000 Code Applicability Report," WCAP-15644-P Revision 2 (Proprietary) and WCAP-15644-NP, (Nonproprietary), March 2004.

7. Sung, Y. X., Schueren, P., and Meliksetian, A., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), October 1999.

Comment [B10]: [15.1-10]

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Comment [B11]: [15.1-11]

Table 15.1.2-1 (Sheet 1 of 2)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

Accident	Event	Time (seconds)
Excessive increase in secondary steam flow		
– Manual reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	200.0
– Manual reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	170.0
– Automatic reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	400.0*
– Automatic reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached (approximate time only)	70.0
Feedwater system malfunctions that result in an increase in feedwater flow	Both main feedwater control valves fail fully open	0.0
	Minimum DNBR occurs	103.9
	Turbine trip/feedwater isolation and reactor trip on high steam generator level	230.7
	Rod motion begins	232.7
Inadvertent operation of the PRHR	PRHR discharge valves go fully open	0.0
	Reactor trip setpoint reached	0.0
	Rod motion begins	1.25
	Rods fully inserted	3.95

*Although oscillation in the transients occurs, the nuclear power and DNBR stabilize after 400 seconds

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Loss of offsite power occurs¶
Minimum DNBR occurs

Table 15.1.2-1 (Sheet 2 of 2)

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

Accident	Event	Time (seconds)
Inadvertent opening of a steam generator relief or safety valve	Inadvertent opening of one main steam safety or relief valve	0.0
	"S" actuation signal on safeguards low T_{cold}	119.0
	Core makeup tank actuation	136.0
	Boron reaches core	156.2
Steam system piping failure	Steam line ruptures	0.0
	"S" actuation signal on safeguards low steam line pressure	1.4
	Criticality attained	28.8
	Boron reaches core	37.4
	Pressurizer and surgeline empty	54.6

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Comment [B12]: [15.1-12]

Table 15.1.5-1 **PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK**	
Reactor coolant iodine activity	
– Accident-initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Appendix 15A). Duration of spike is 5 hours.
– Pre-accident spike	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of reactor coolant concentrations at maximum equilibrium conditions
Duration of accident (hr)	72
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Steam generator in faulted loop	
– Initial water mass (lb)	3.02 E+05
– Primary to secondary leak rate (lb/hr)	52.25 [(a)]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.021E+05 3.66 E+03
Steam generator in intact loop	
– Primary to secondary leak rate (lb/hr)	52.25[(a)]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.021 E+05 3.66 E+03
Nuclide data	See Table 15A-4

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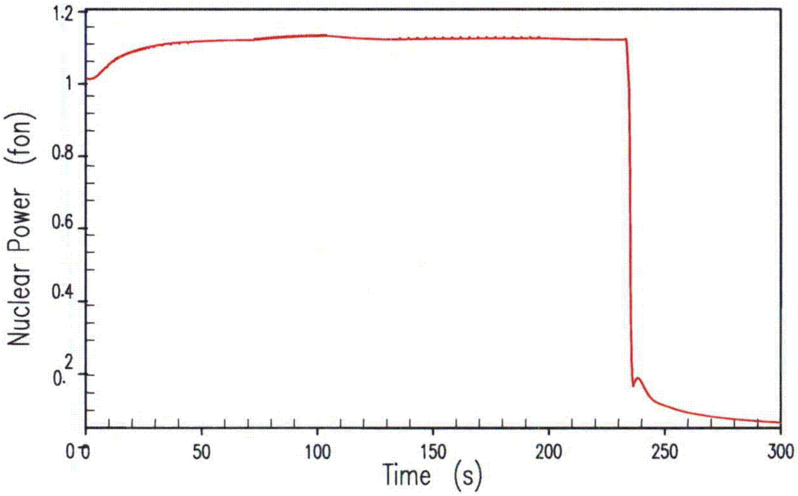
Note:

a. Equivalent to 150 gpd cooled liquid at 62.4 lb/ft^3.

Table 15.1.5.5-1 **TIME SEQUENCE OF EVENTS FOR STEAM SYSTEM PIPING FAILURE AT FULL POWER – 0.87 FT^2 BREAK SIZE**	
Event	**Time** (seconds)
Steam line rupture	0.0
OPΔT reactor trip setpoint reached	12.9
Rods begin to drop	13.9
Minimum DNBR occurs	14.9
Maximum core heat flux occurs	14.9



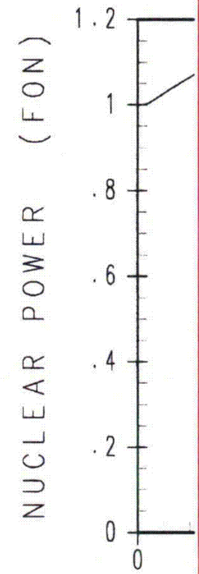


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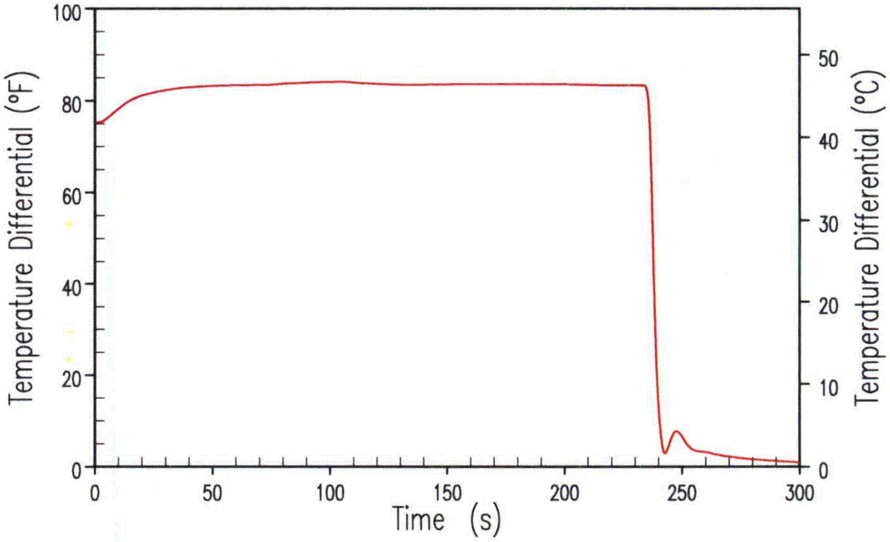
Figure 15.1.2-1

Feedwater Control Valve Malfunction Nuclear Power



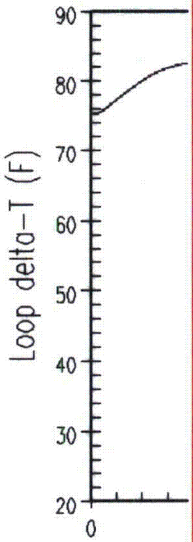


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Figure 15.1.2-2

Feedwater Control Valve Malfunction Loop ΔT

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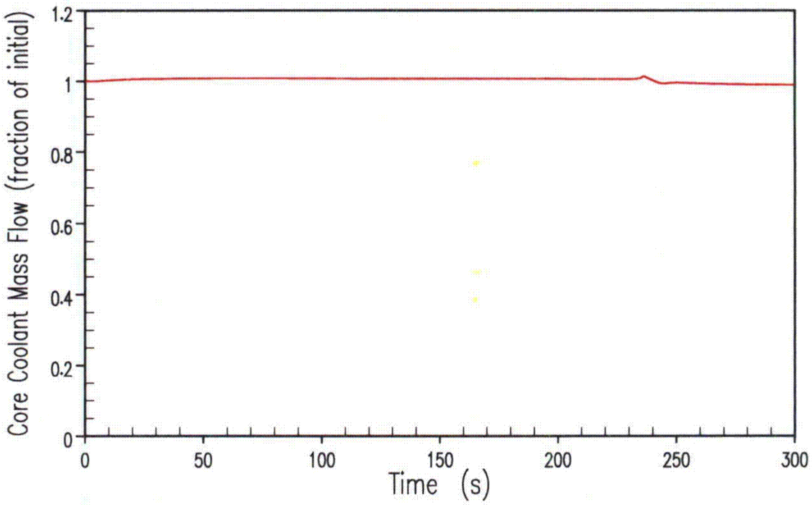

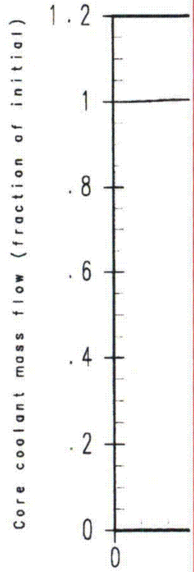

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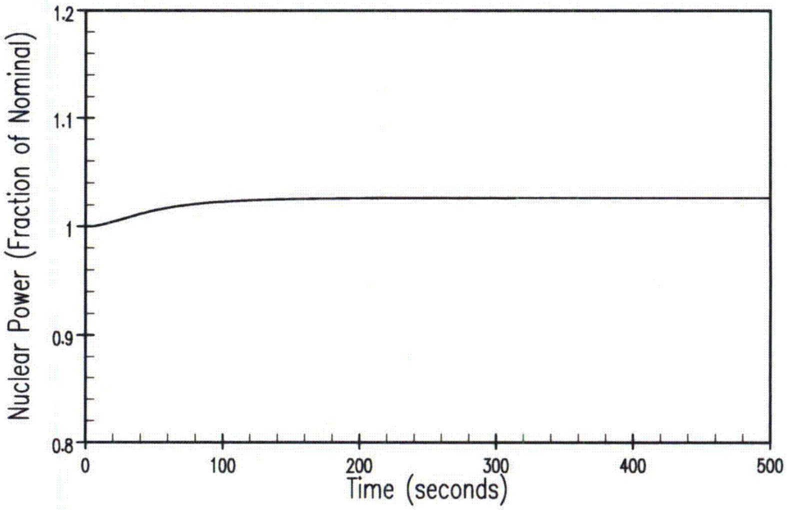
Figure 15.1.2-3

Feedwater Control Valve Malfunction Core Coolant Mass Flow



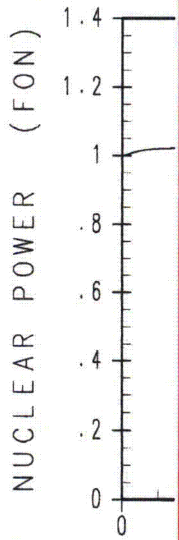


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Figure 15.1.3-1

Nuclear Power Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

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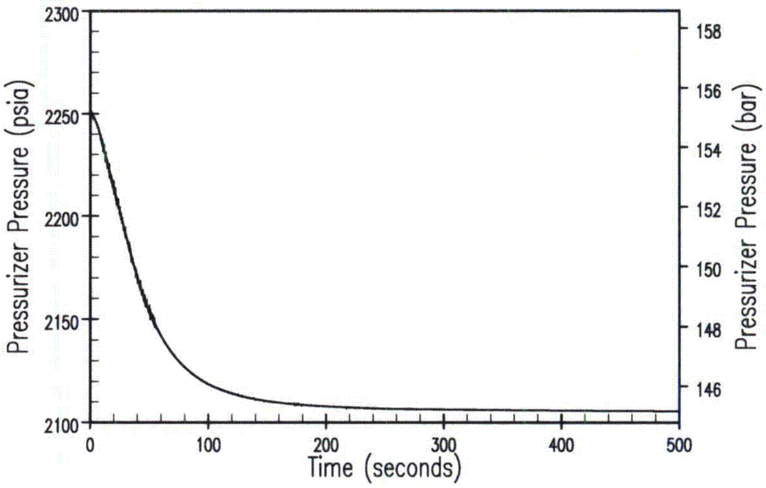

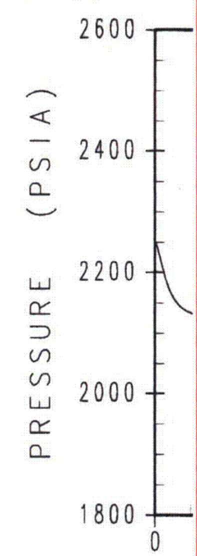

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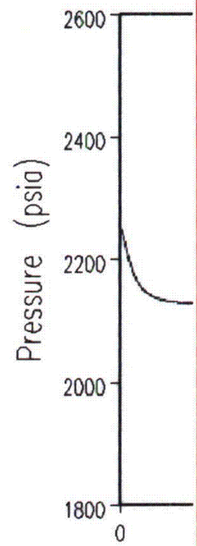

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Figure 15.1.3-2

Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

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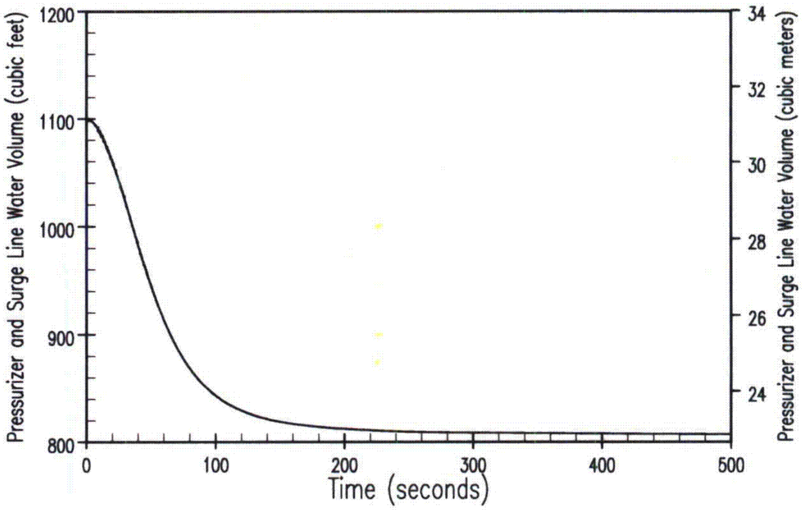
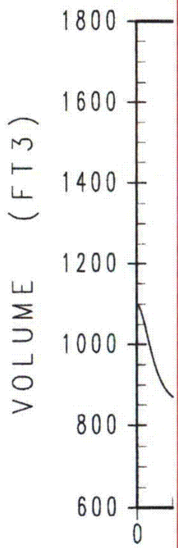


Figure 15.1.3-3

Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback



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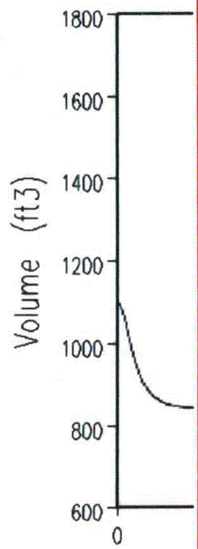

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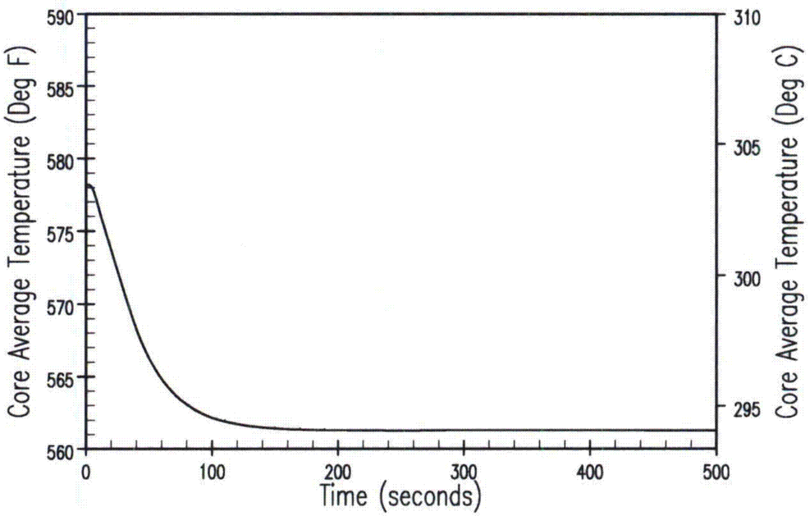

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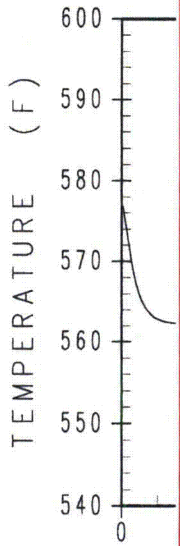

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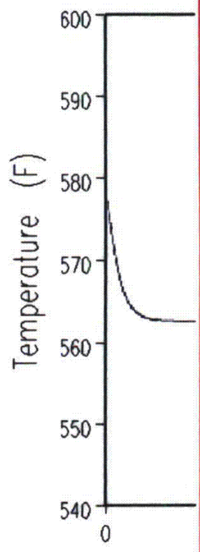

Figure 15.1.3-4

Core Average Temperature Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback

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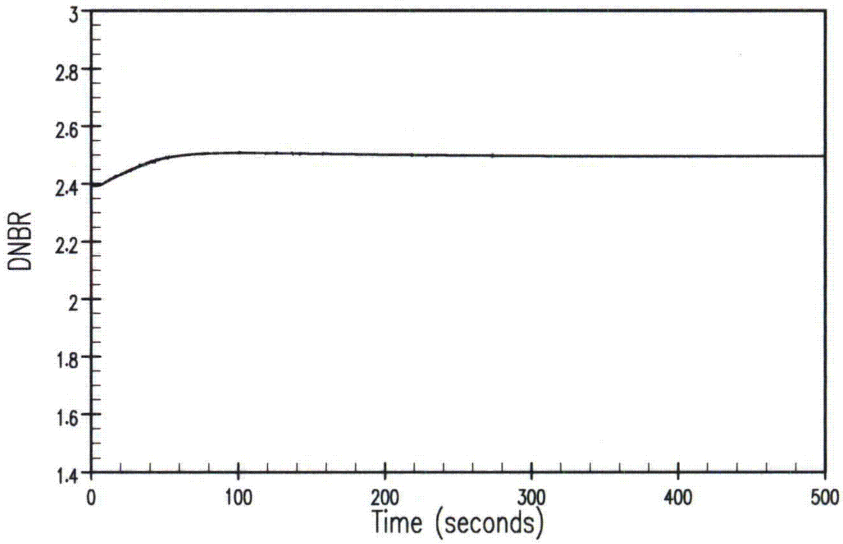
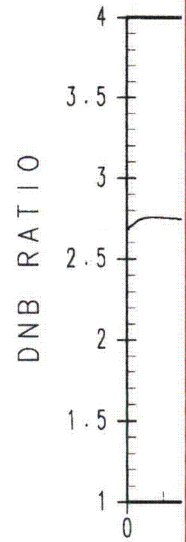


Figure 15.1.3-5

DNBR Versus Time for 10-percent Step Load Increase, Manual Control and Minimum Moderator Feedback



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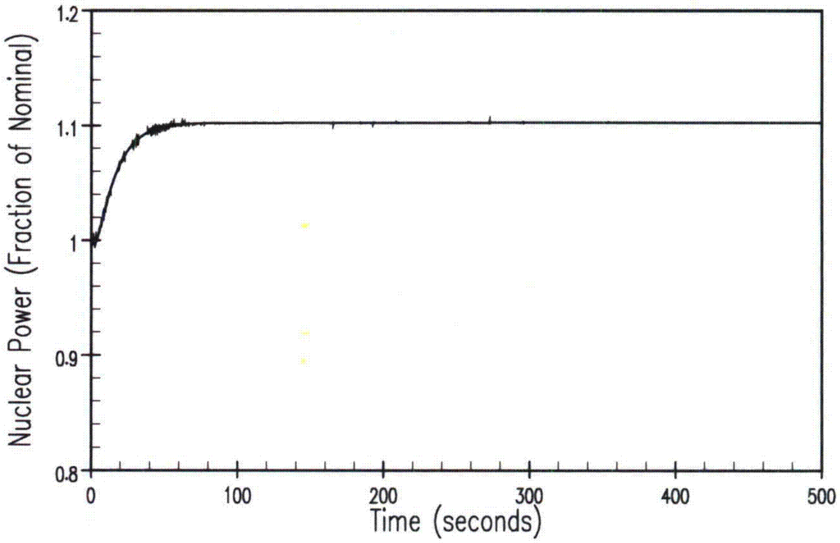

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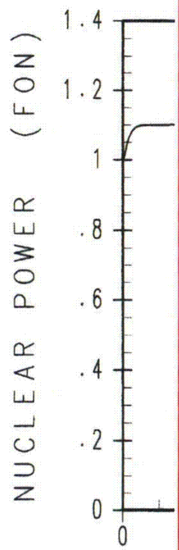

Figure 15.1.3-6

Nuclear Power Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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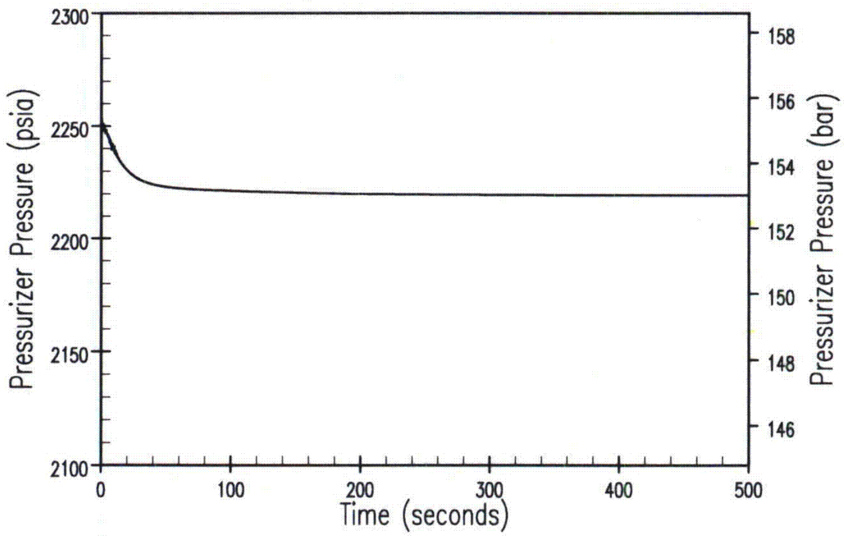

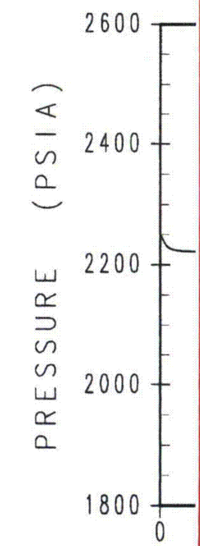

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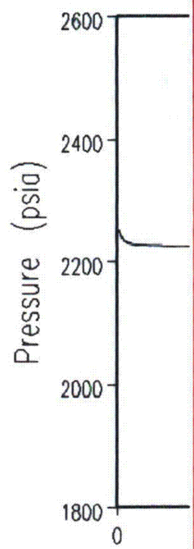

Figure 15.1.3-7

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Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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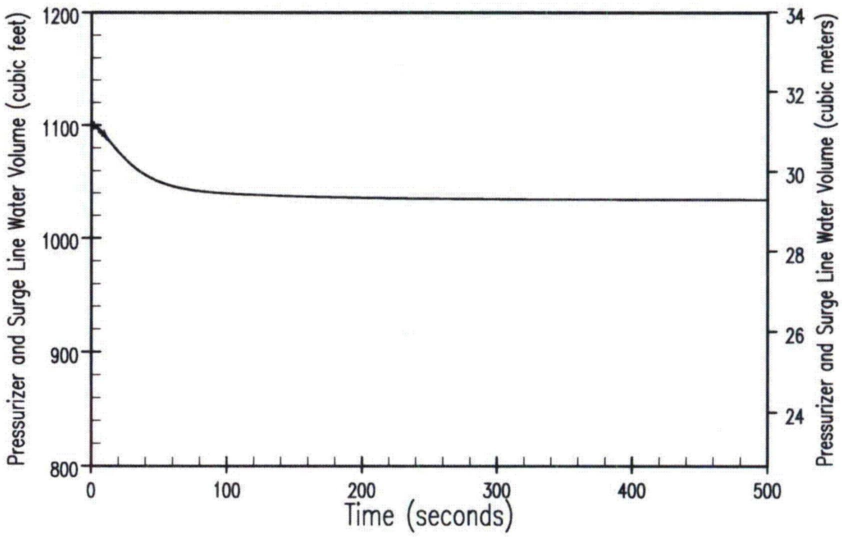

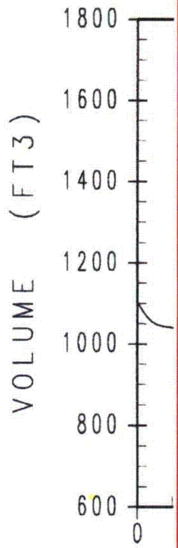

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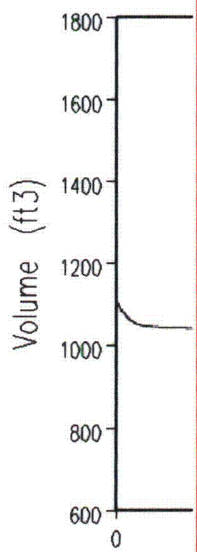

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Figure 15.1.3-8

Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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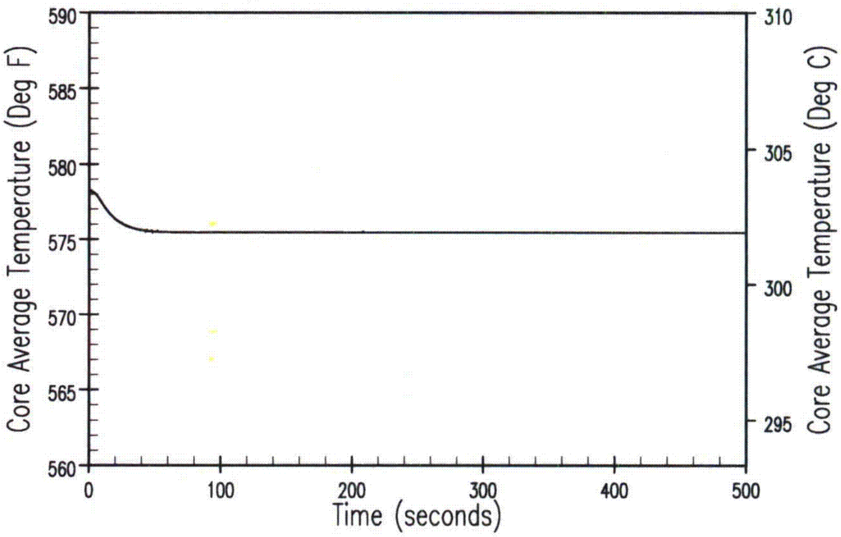

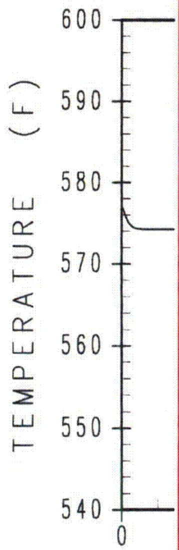

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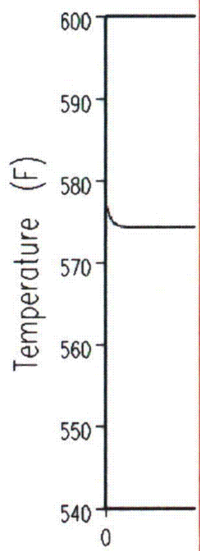

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Figure 15.1.3-9

Core Average Temperature Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback

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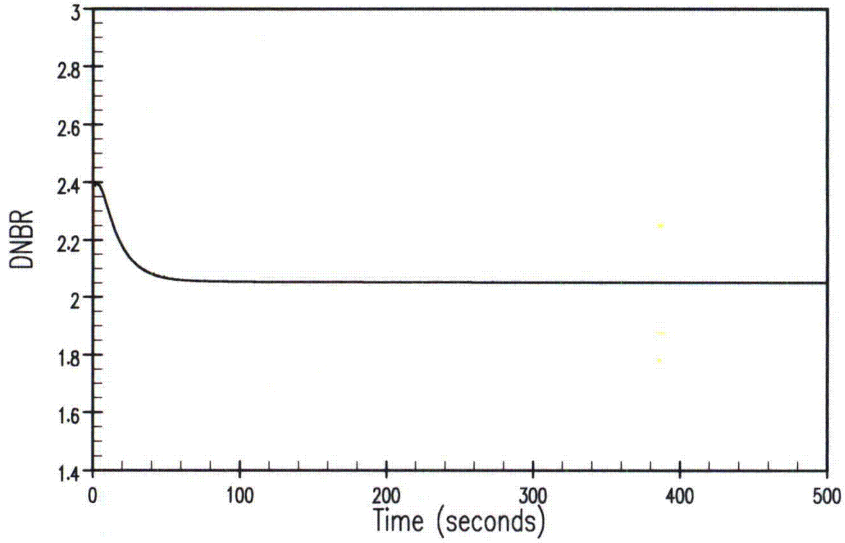

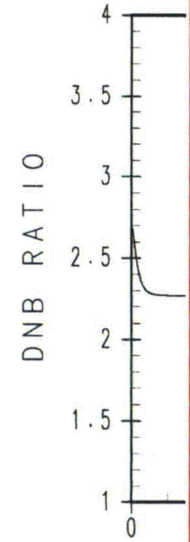

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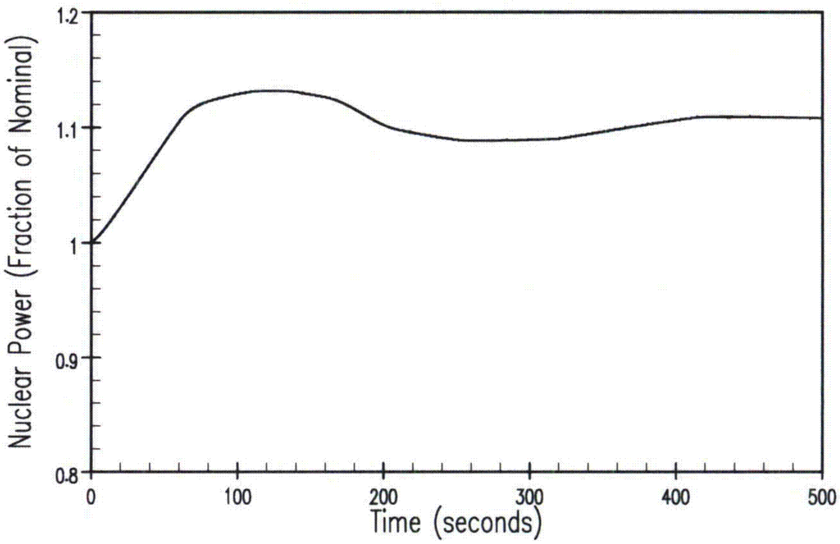
Figure 15.1.3-10

DNBR Versus Time for 10-percent Step Load Increase, Manual Control and Maximum Moderator Feedback



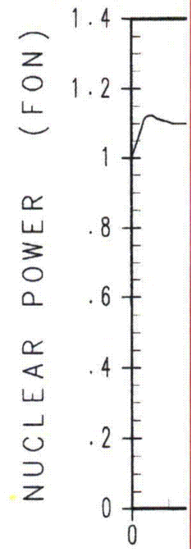


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Figure 15.1.3-11

Nuclear Power Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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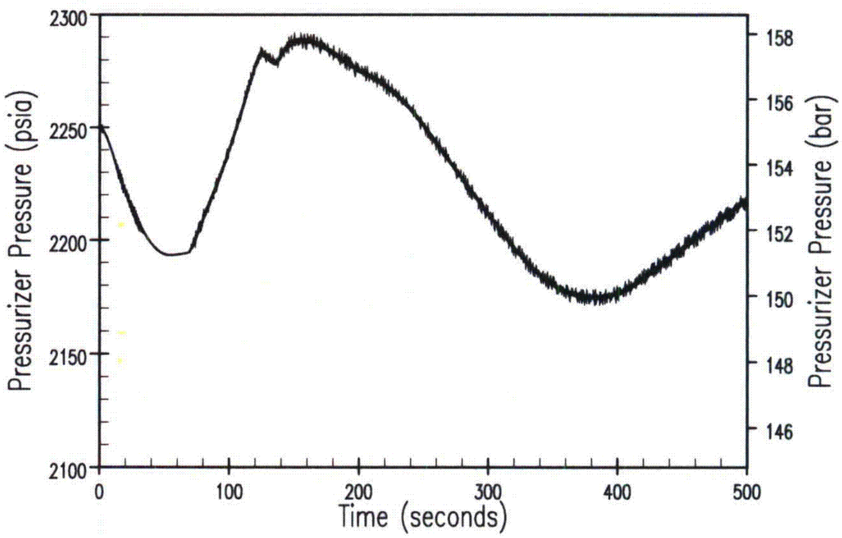

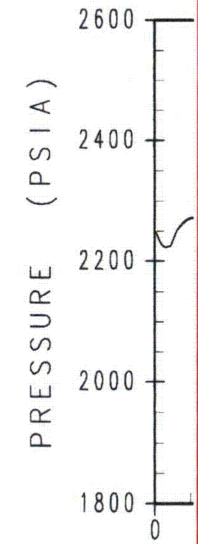

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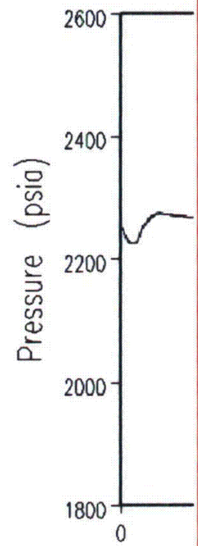

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Figure 15.1.3-12

Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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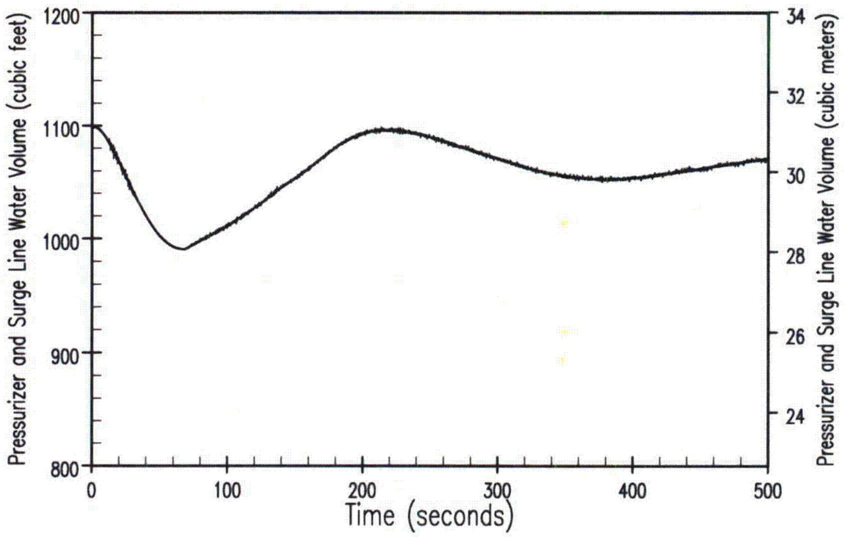

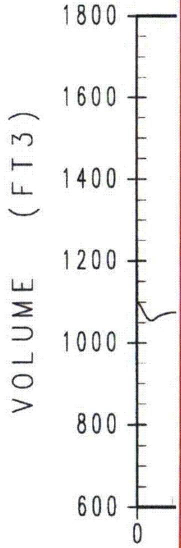

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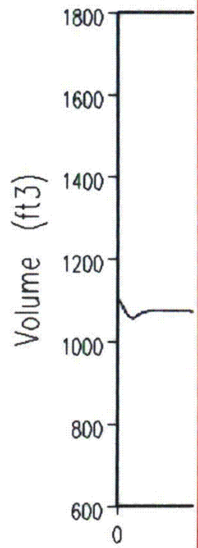

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Figure 15.1.3-13

Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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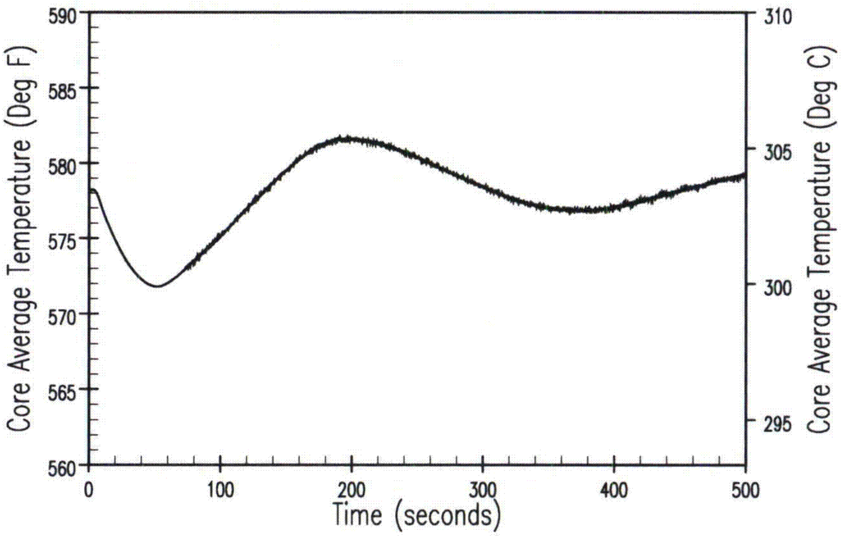

Figure 15.1.3-14

Core Average Temperature Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback

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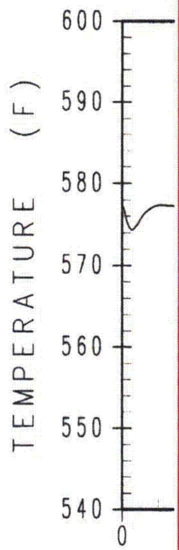

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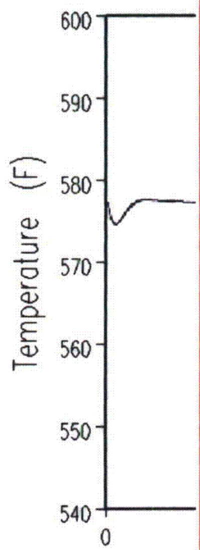

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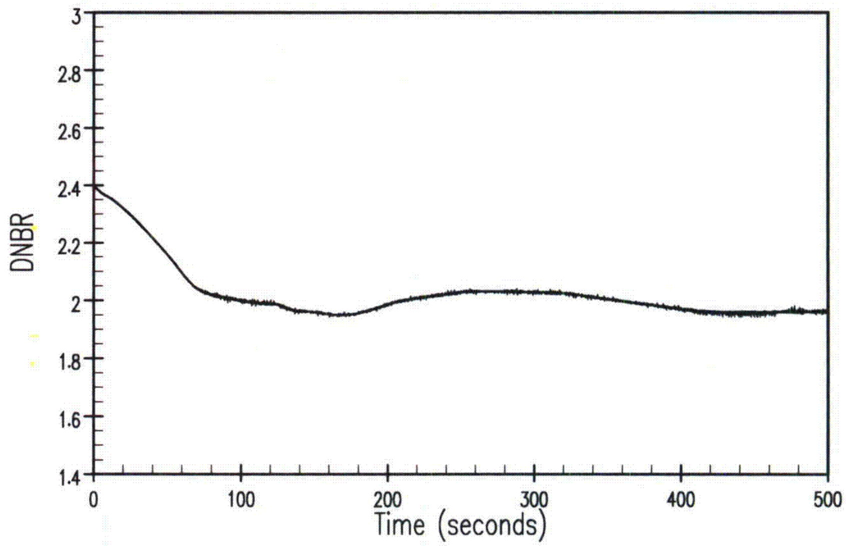

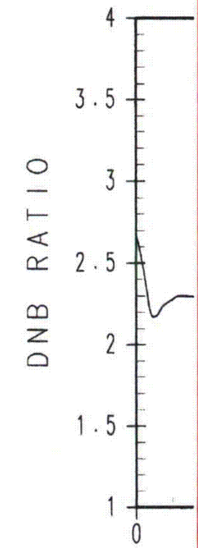

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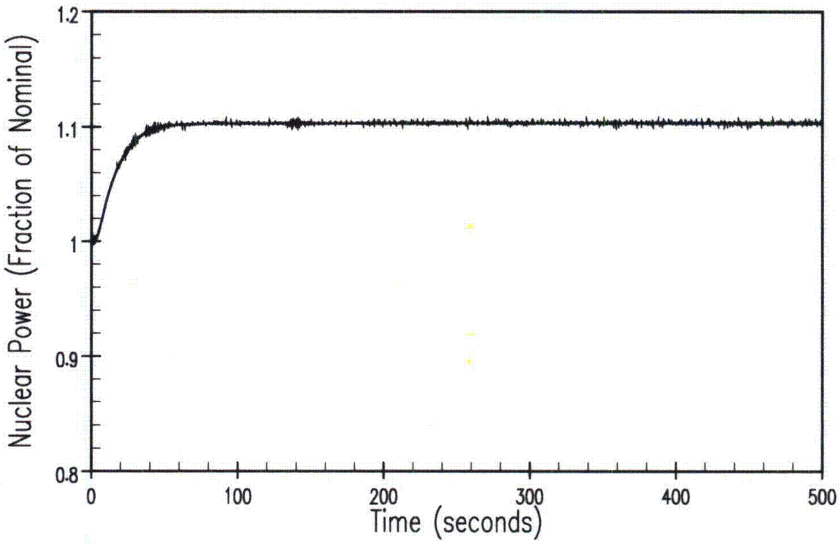
Figure 15.1.3-15

DNBR Versus Time for 10-percent Step Load Increase, Automatic Control and Minimum Moderator Feedback



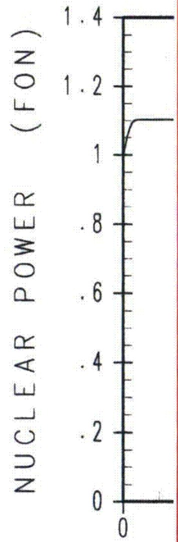


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Figure 15.1.3-16

Nuclear Power Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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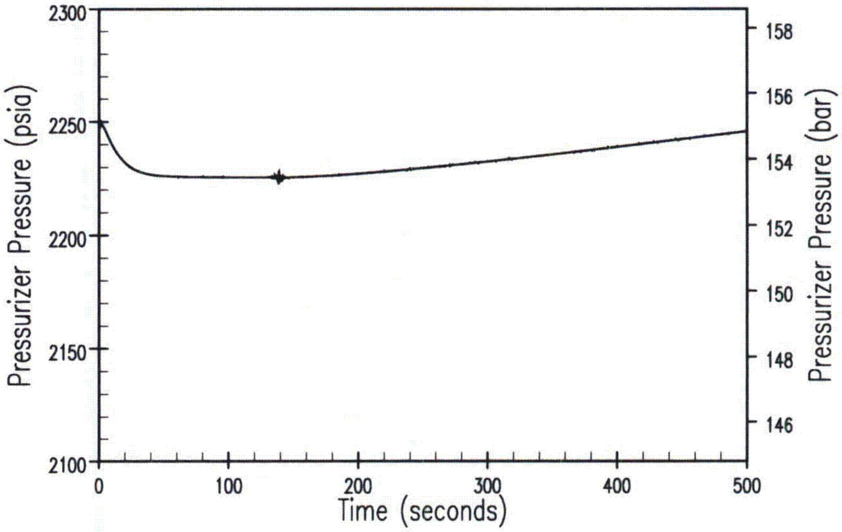

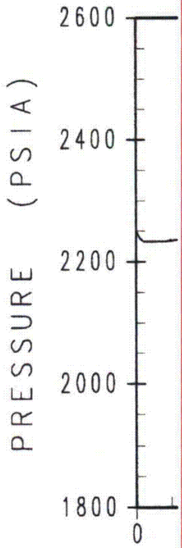

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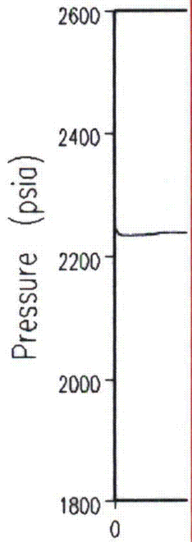

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Figure 15.1.3-17

Pressurizer Pressure Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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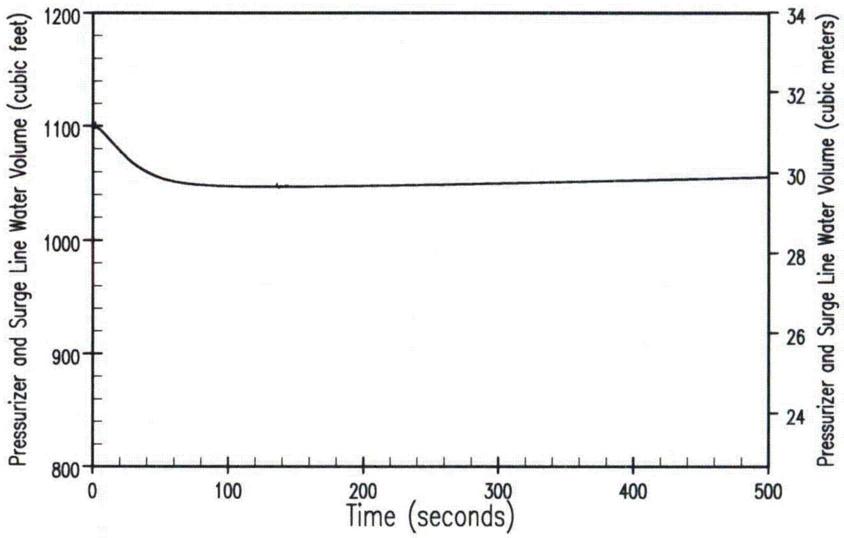

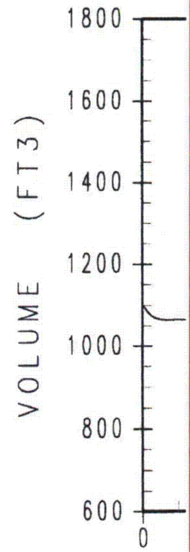

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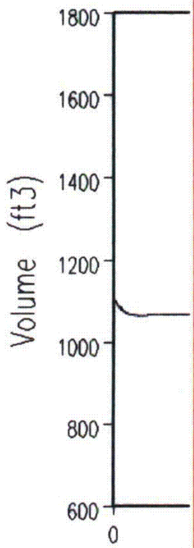

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Figure 15.1.3-18

Pressurizer Water Volume Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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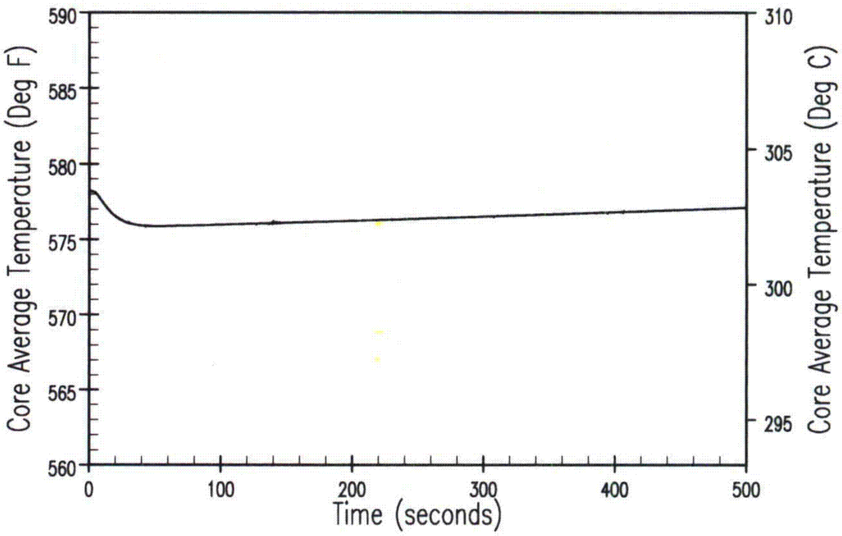

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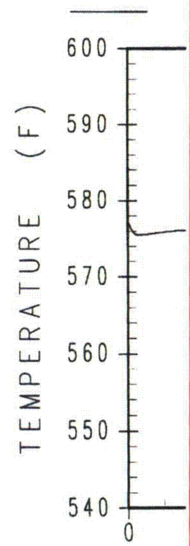

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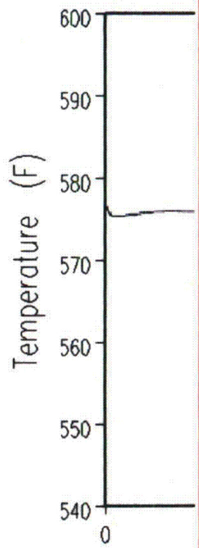

Figure 15.1.3-19

Core Average Temperature Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback

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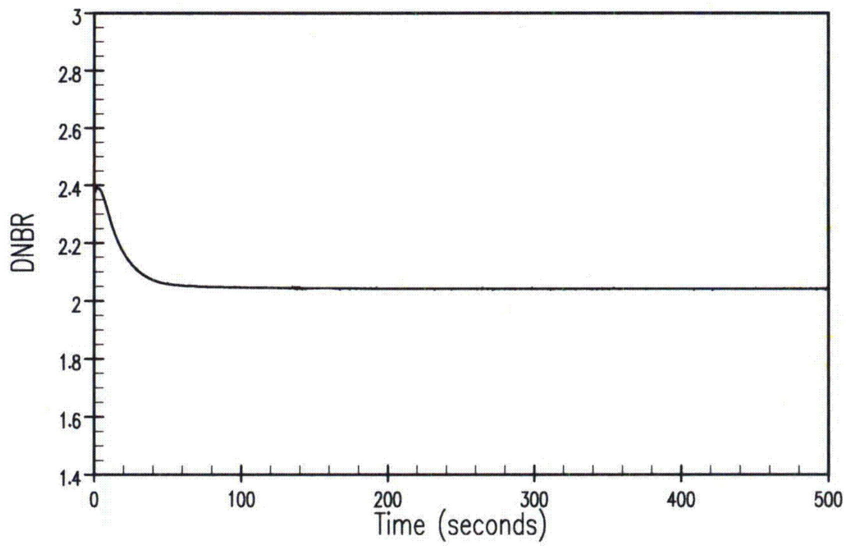

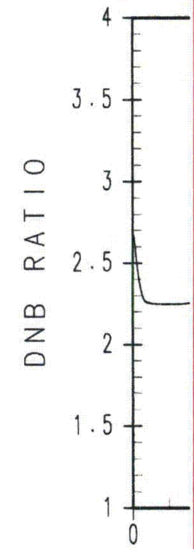

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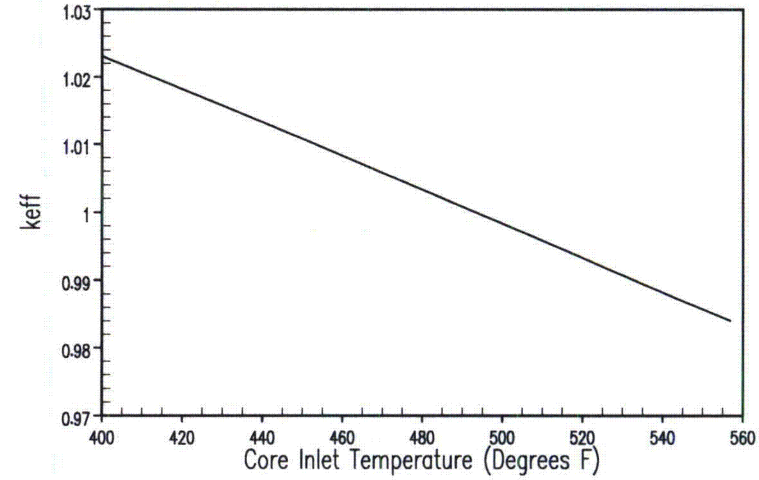
Figure 15.1.3-20

DNBR Versus Time for 10-percent Step Load Increase, Automatic Control and Maximum Moderator Feedback



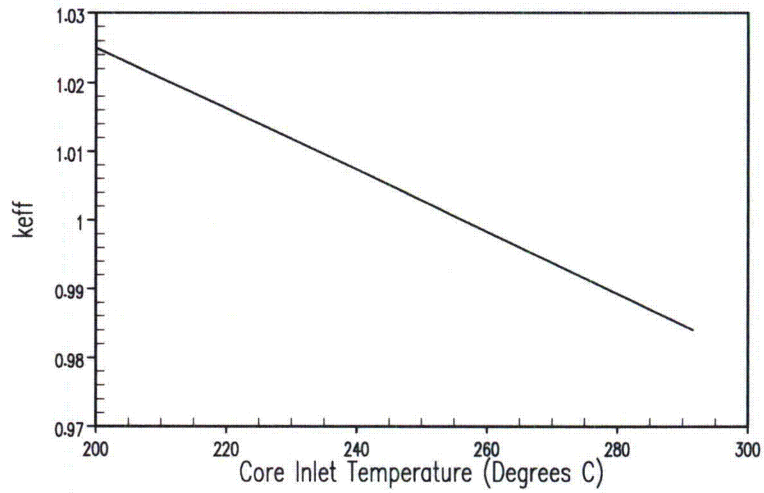
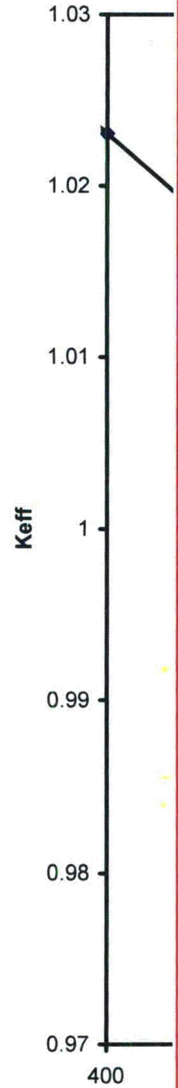


Figure 15.1.4-1

K_{eff} Versus Core Inlet Temperature
Steam Line Break Events



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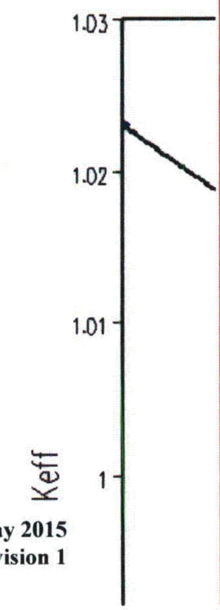

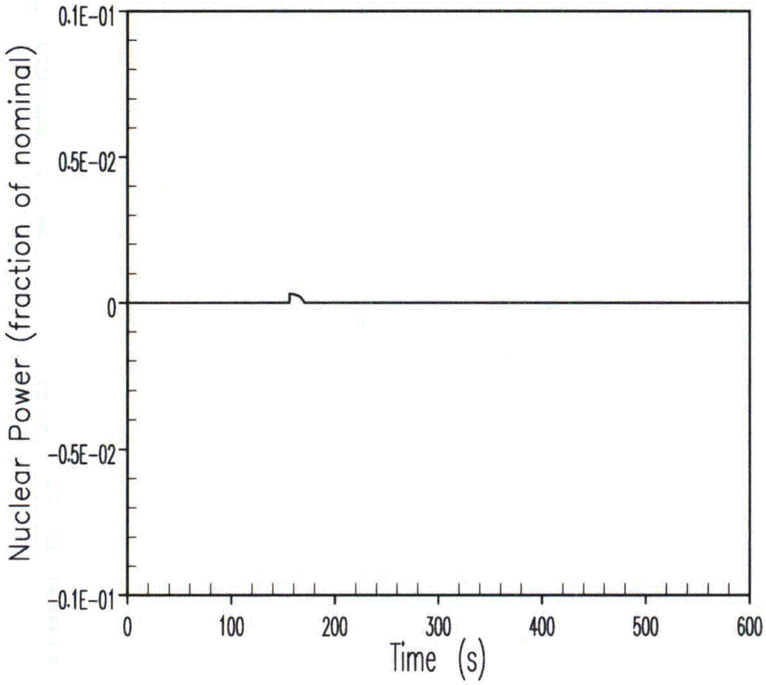

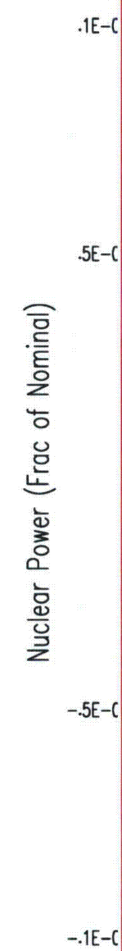

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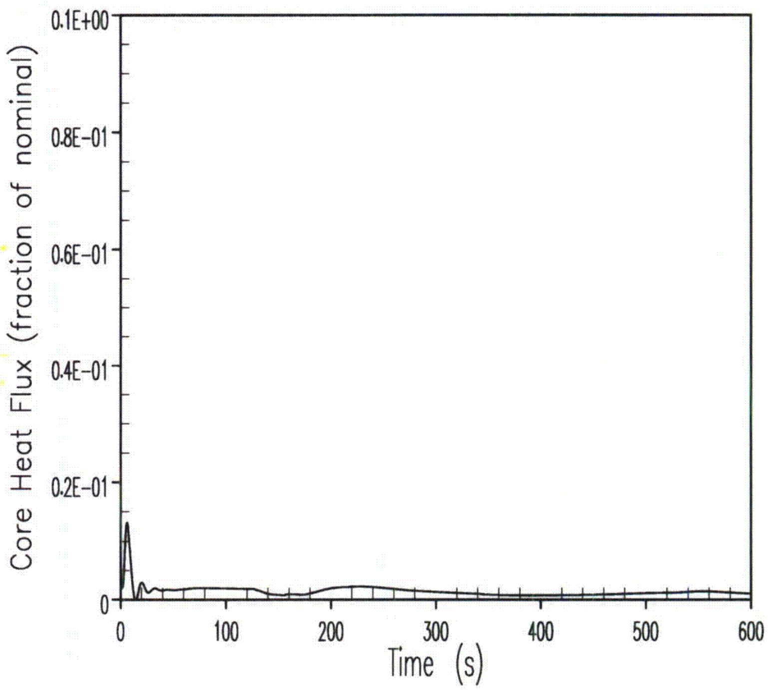
Figure 15.1.4-2

Nuclear Power Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



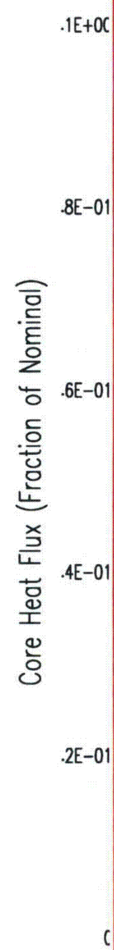


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Figure 15.1.4-3

Core Heat Flux Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

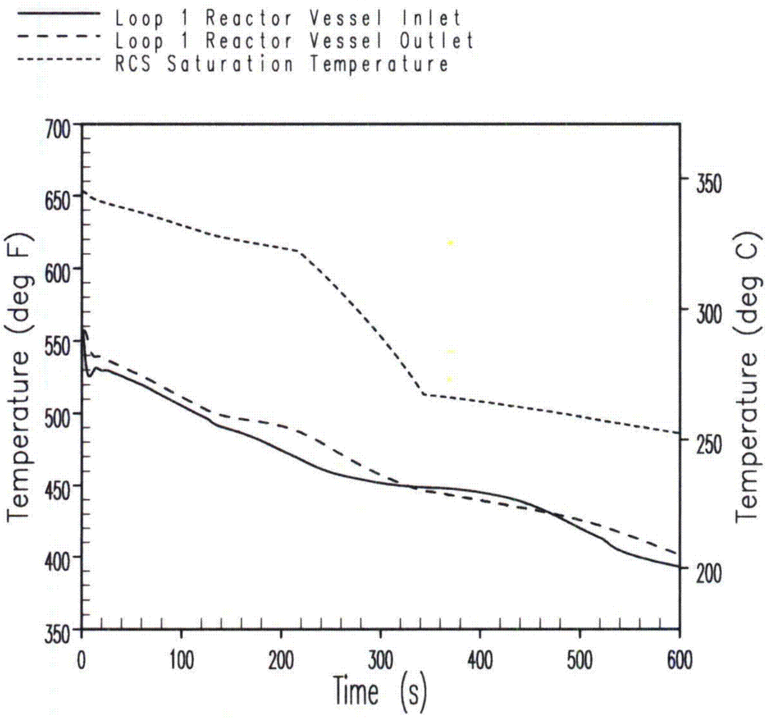
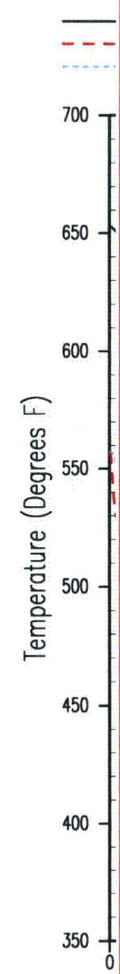


Figure 15.1.4-4

Loop 1 Reactor Coolant Temperatures
Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1-60



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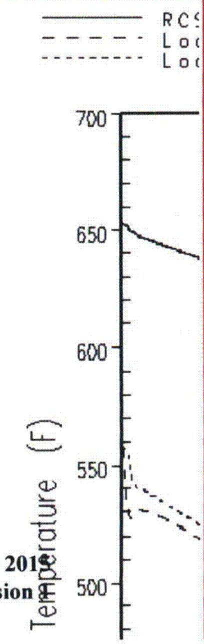


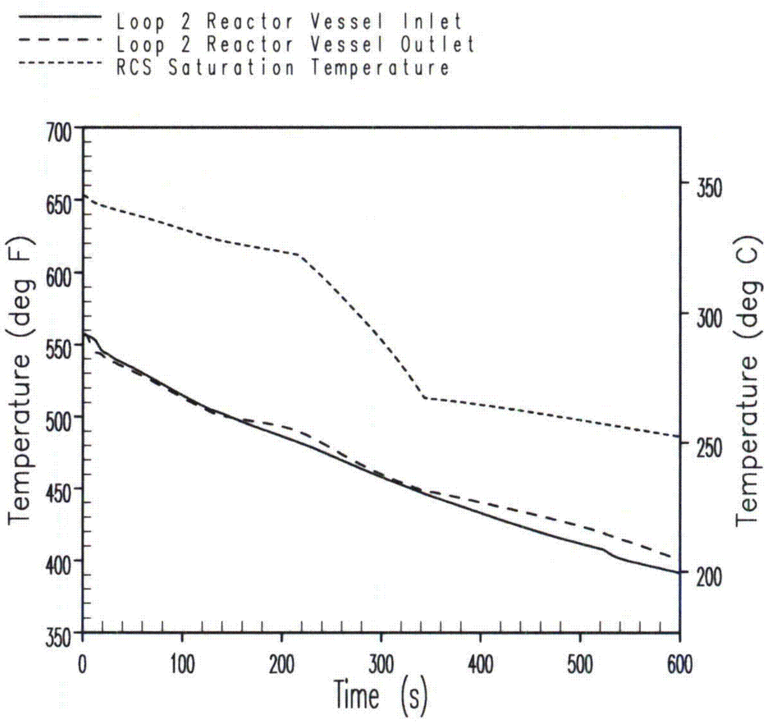
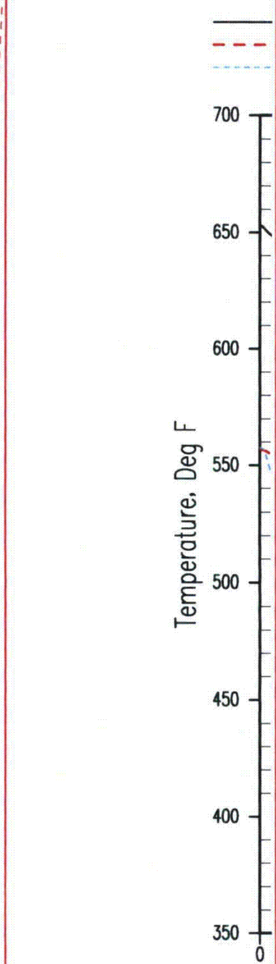


Figure 15.1.4-5

Loop 2 (Faulted Loop) Reactor Coolant Temperatures
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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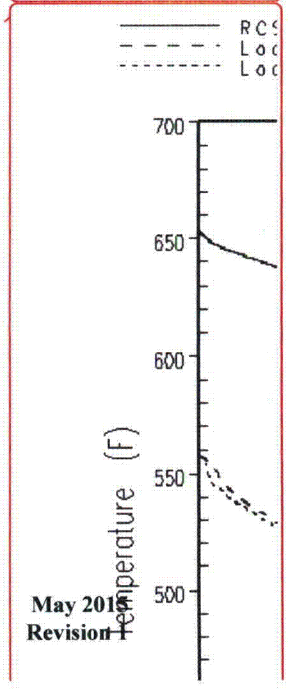

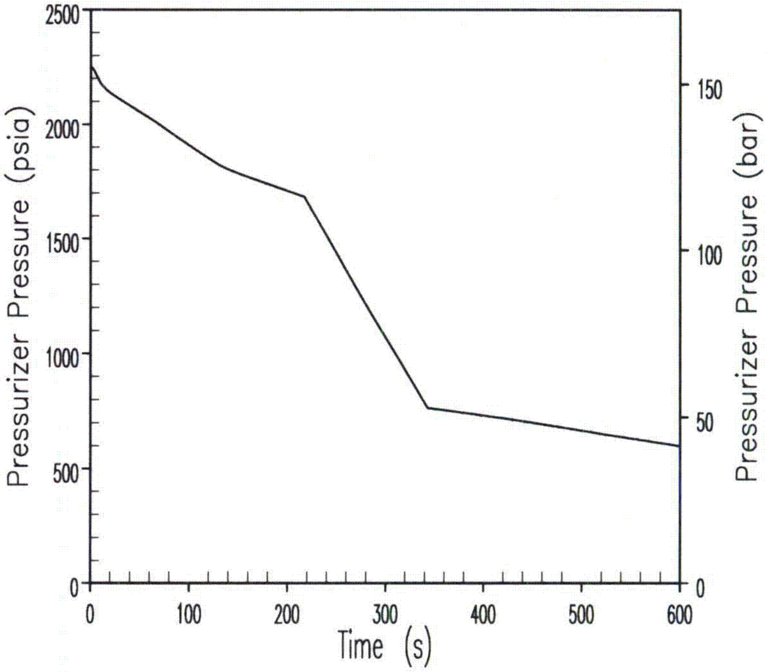

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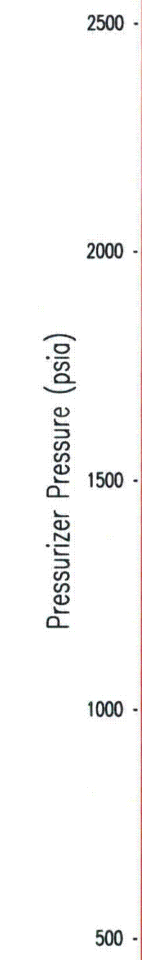


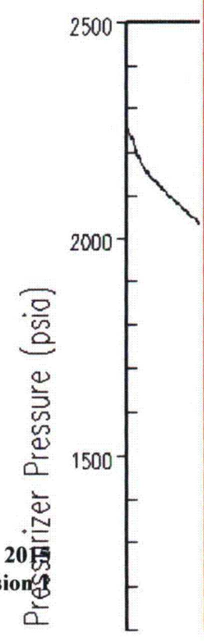


Figure 15.1.4-6

Pressurizer Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1-62

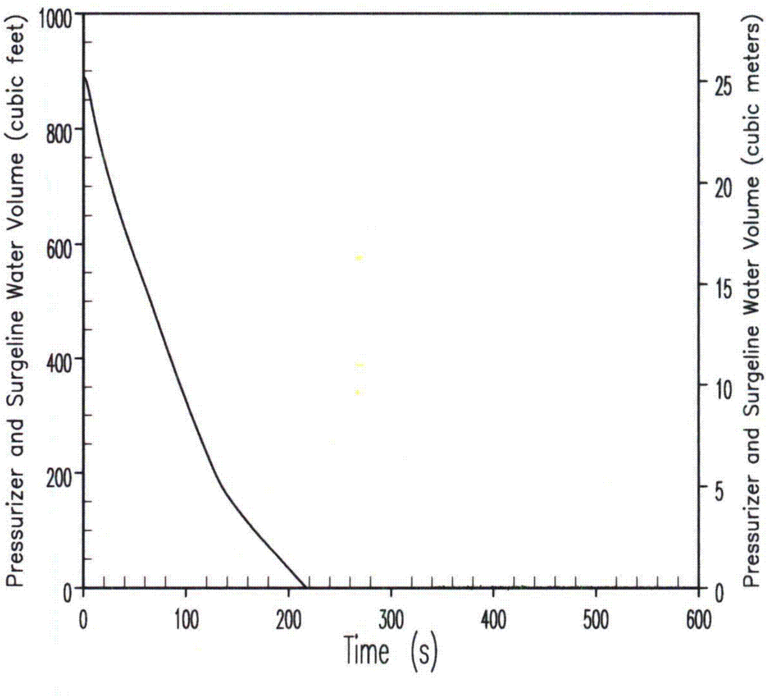
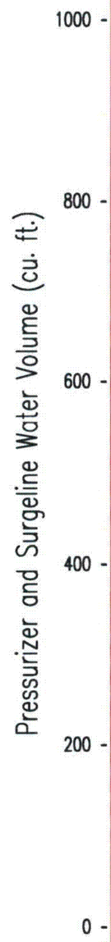


Figure 15.1.4-7

Pressurizer and Surgeline Water Volume Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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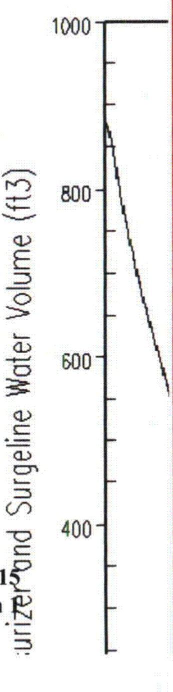

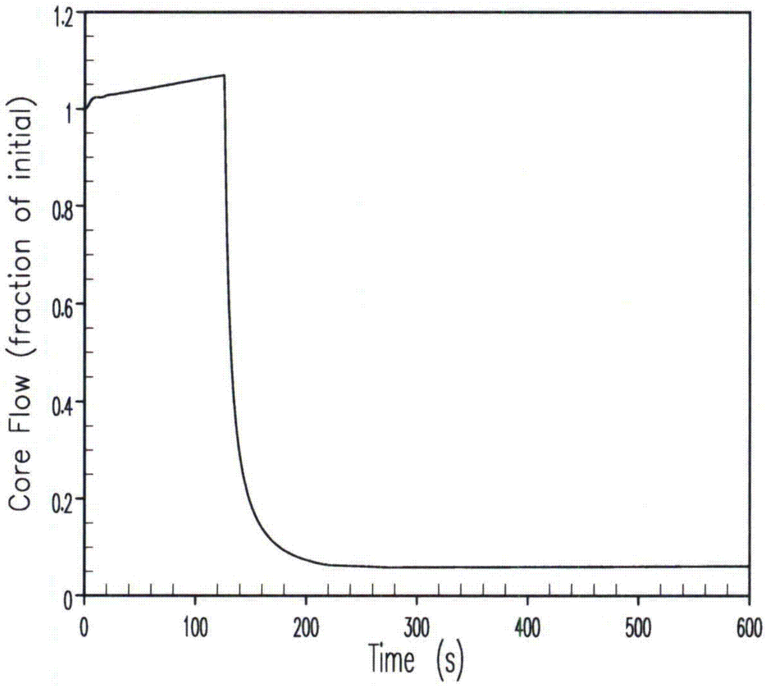

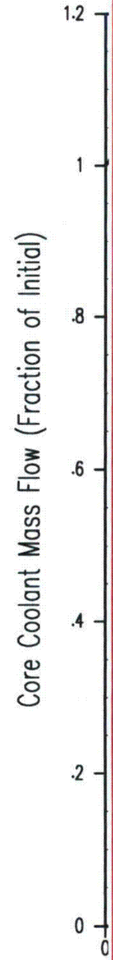

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Figure 15.1.4-8

Core Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1-64

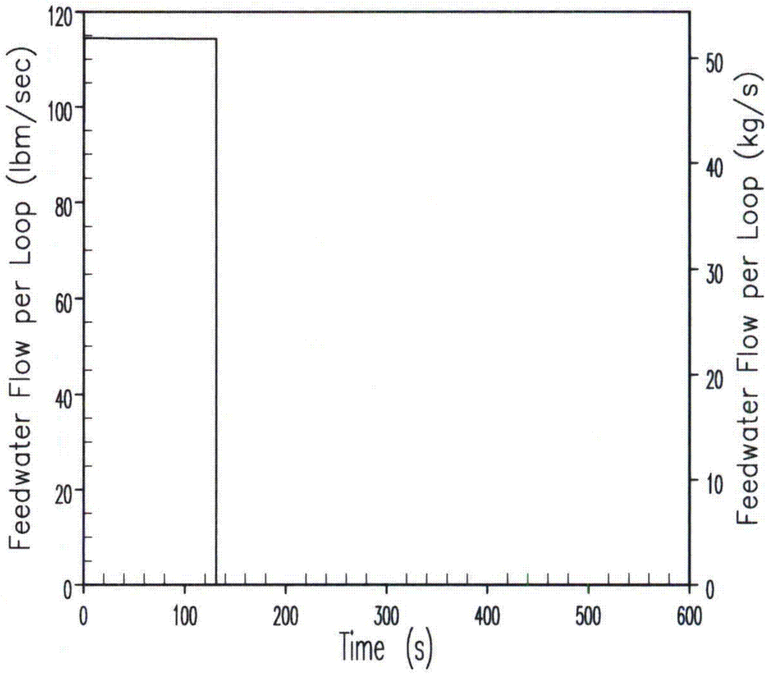
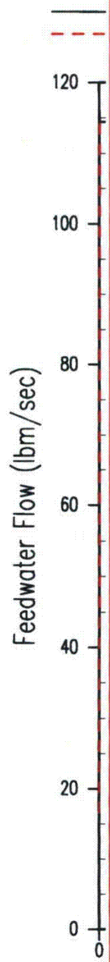


Figure 15.1.4-9

Feedwater Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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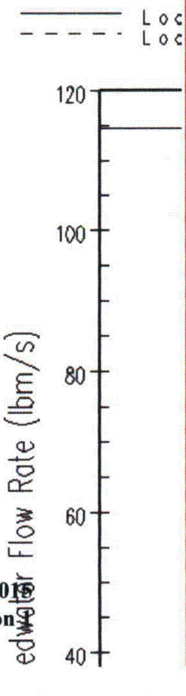

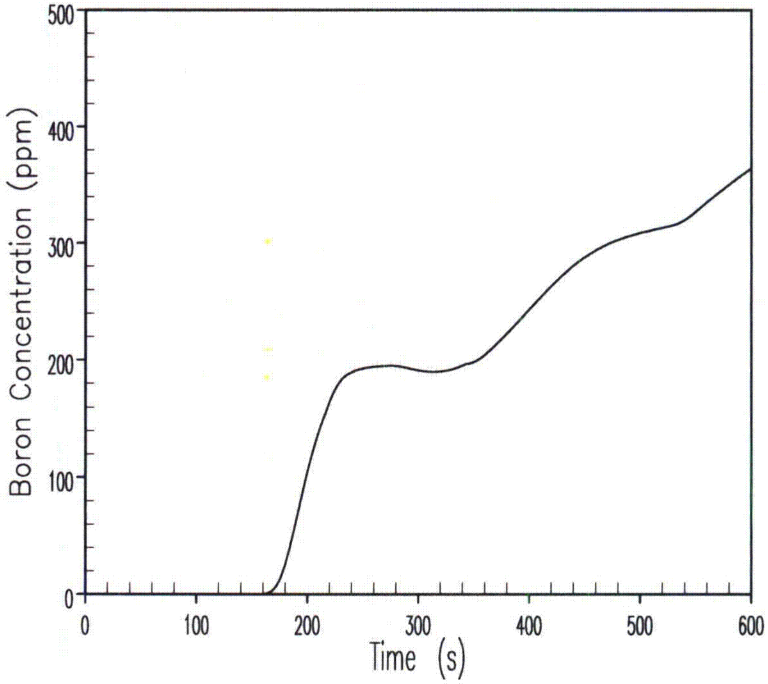

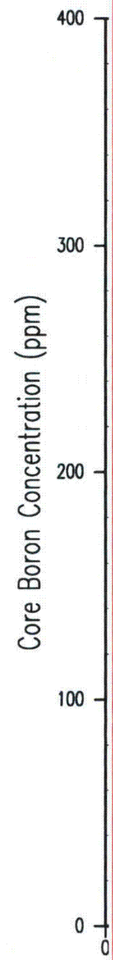

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Figure 15.1.4-10

Core Boron Concentration Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve

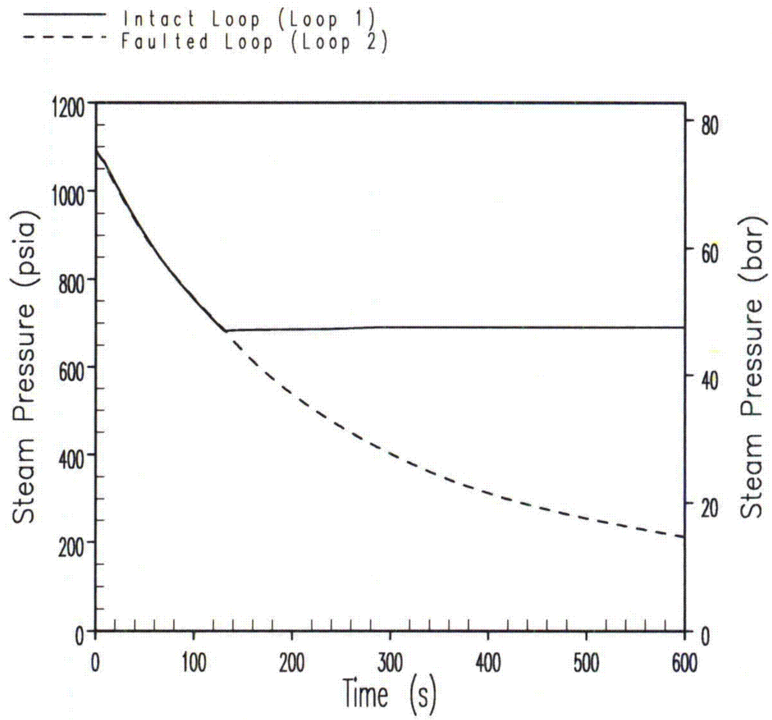
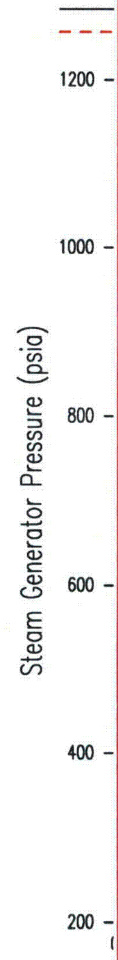


Figure 15.1.4-11

Steam Pressure Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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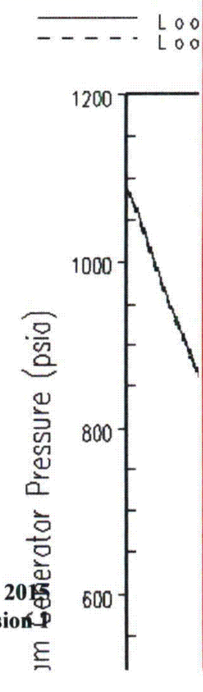


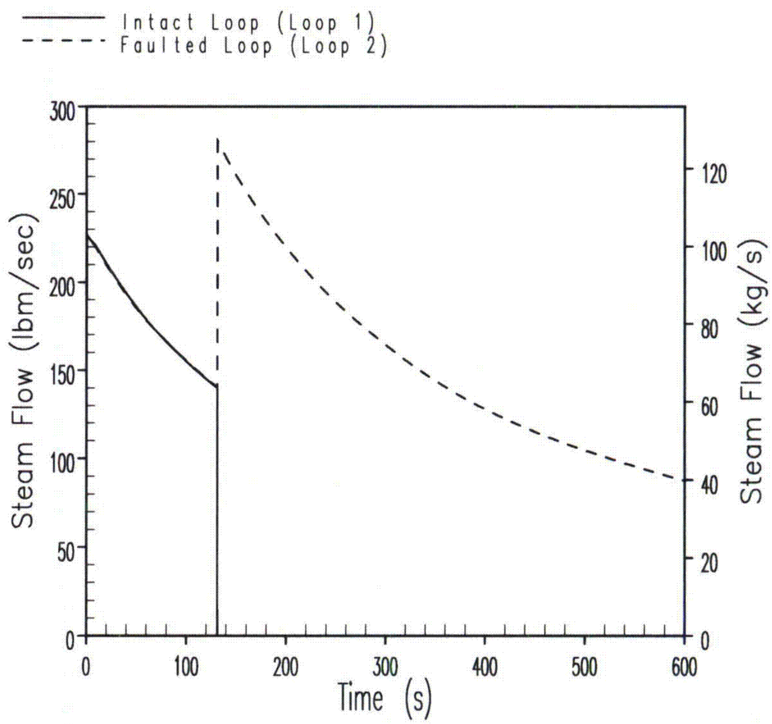
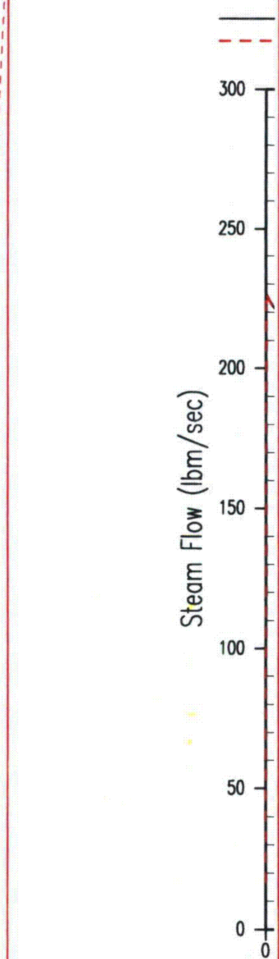


Figure 15.1.4-12

Steam Flow Transient
Inadvertent Opening of a Steam Generator Relief or Safety Valve



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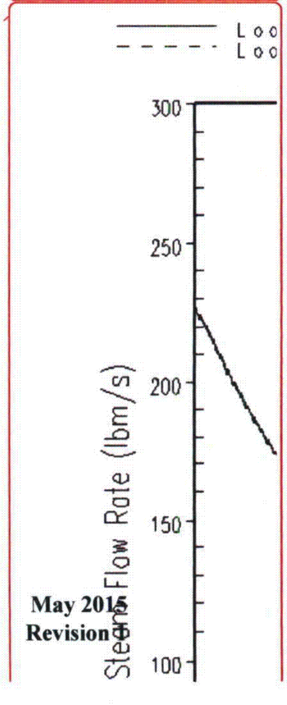

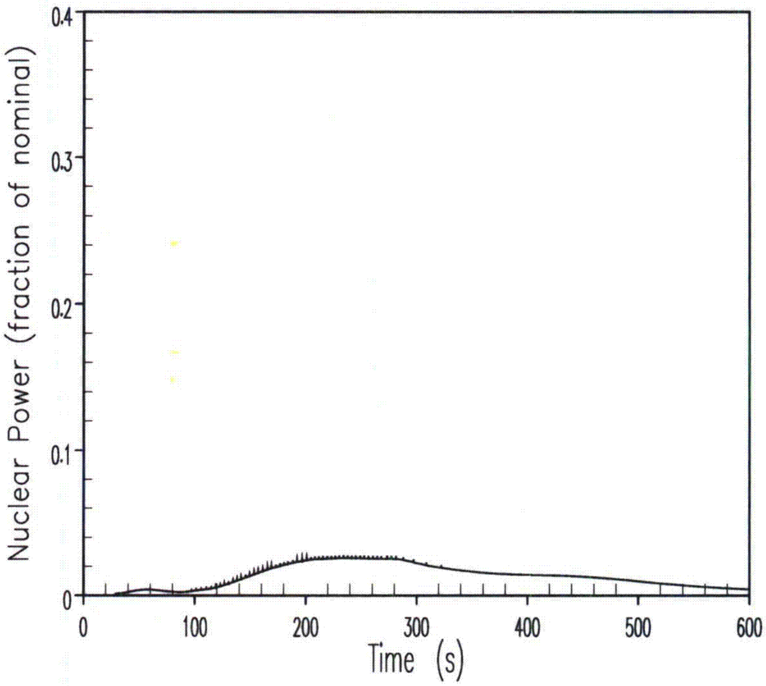

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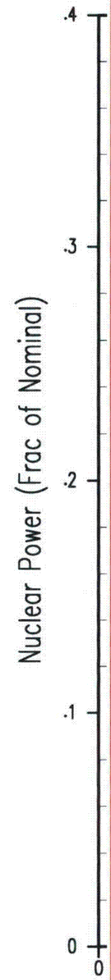

Figure 15.1.5-1

Nuclear Power Transient Steam System Piping Failure

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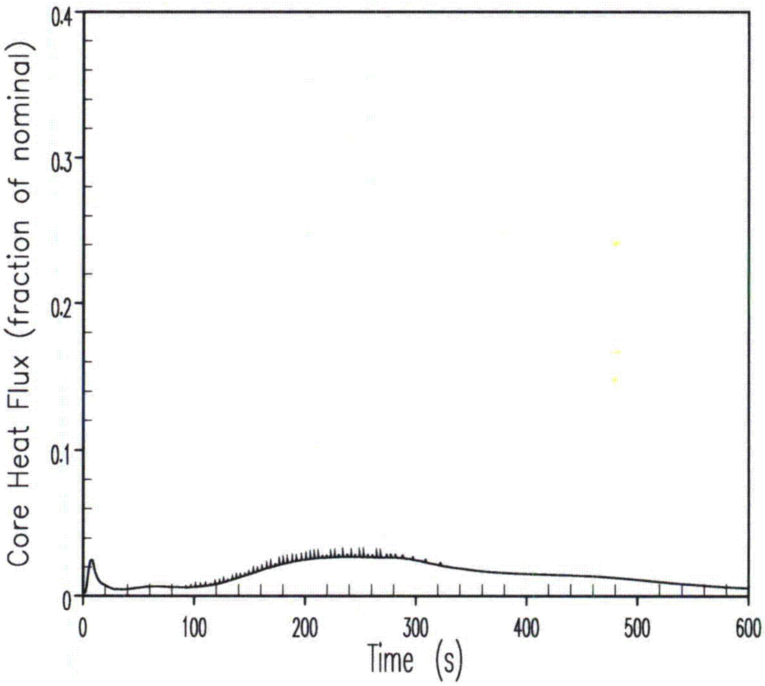

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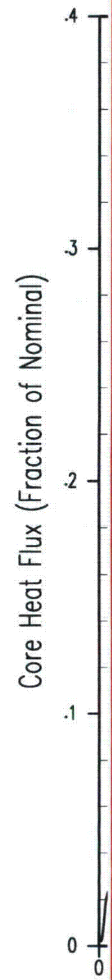


Figure 15.1.5-2

Core Heat Flux Transient Steam System Piping Failure

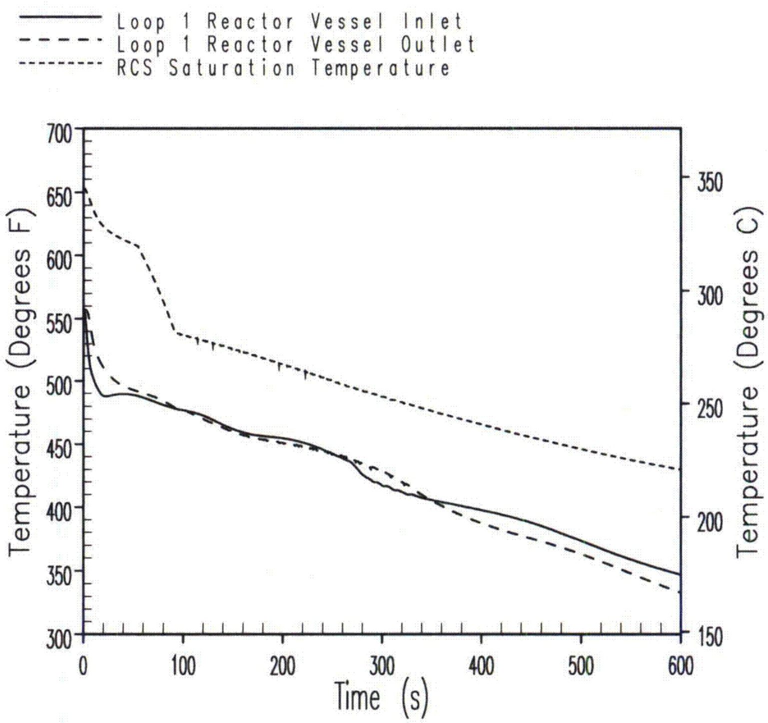
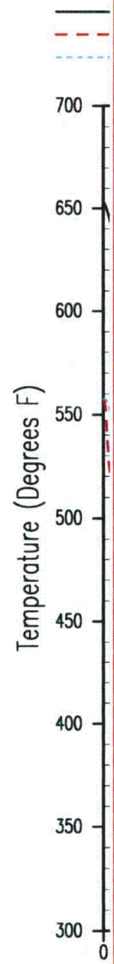


Figure 15.1.5-3

Loop 1 Reactor Coolant Temperatures
Steam System Piping Failure



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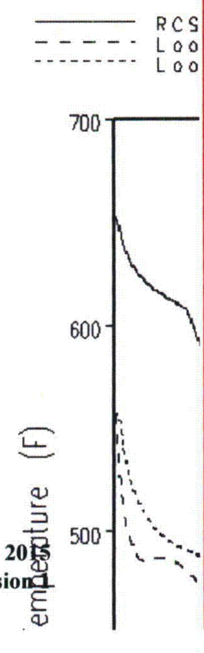


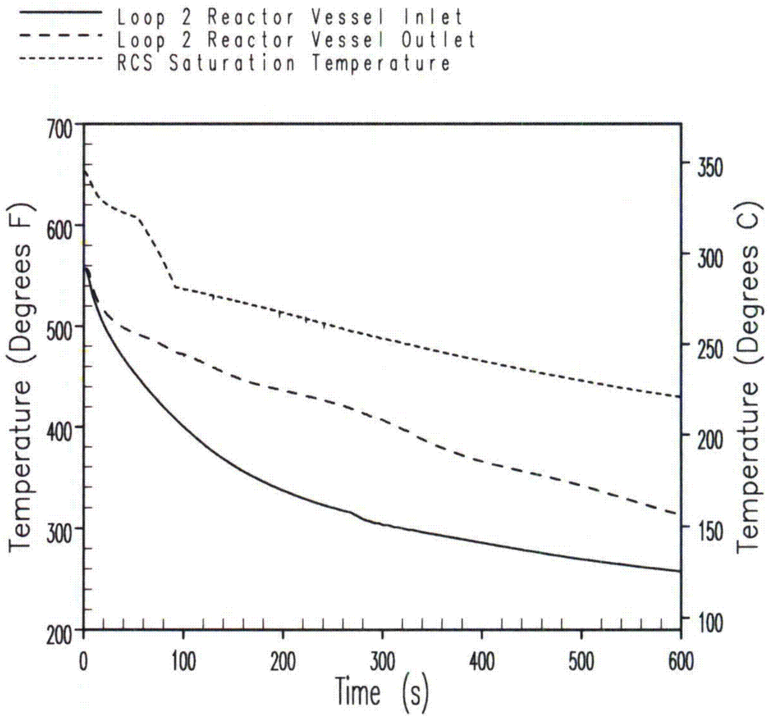
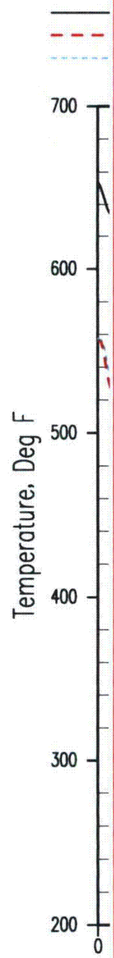


Figure 15.1.5-4

Loop 2 Reactor Coolant Temperatures
Steam System Piping Failure



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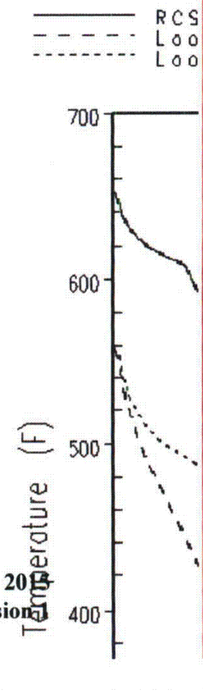

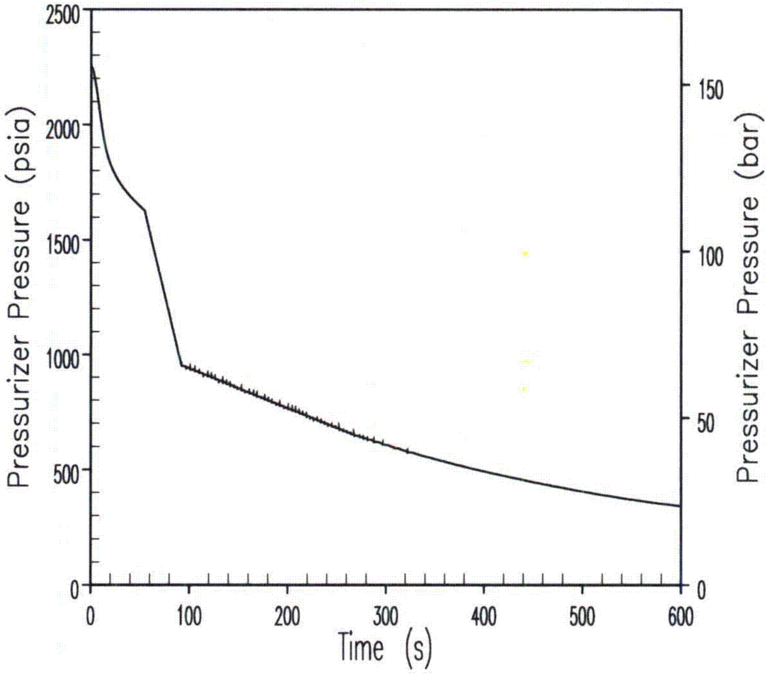

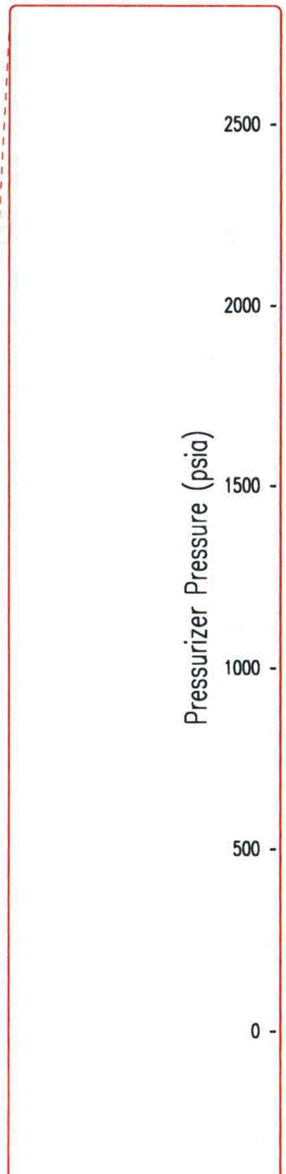

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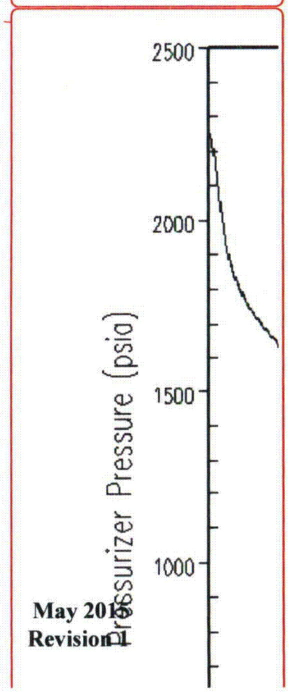

Figure 15.1.5-5

Pressurizer Pressure Transient
Steam System Piping Failure

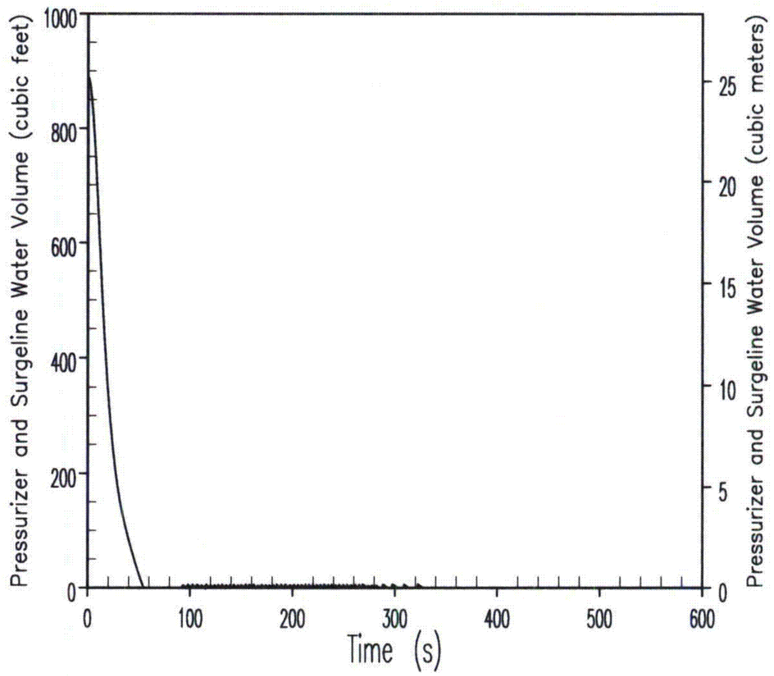

Figure 15.1.5-6

Pressurizer and Surgeline Water Volume Transient
Steam System Piping Failure

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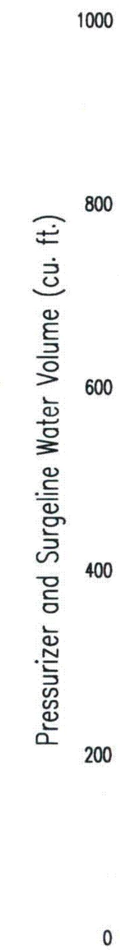

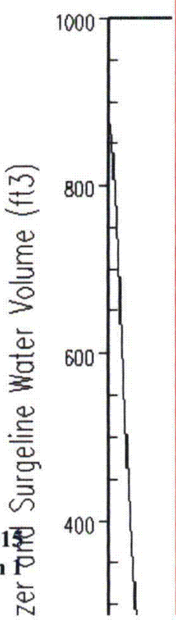

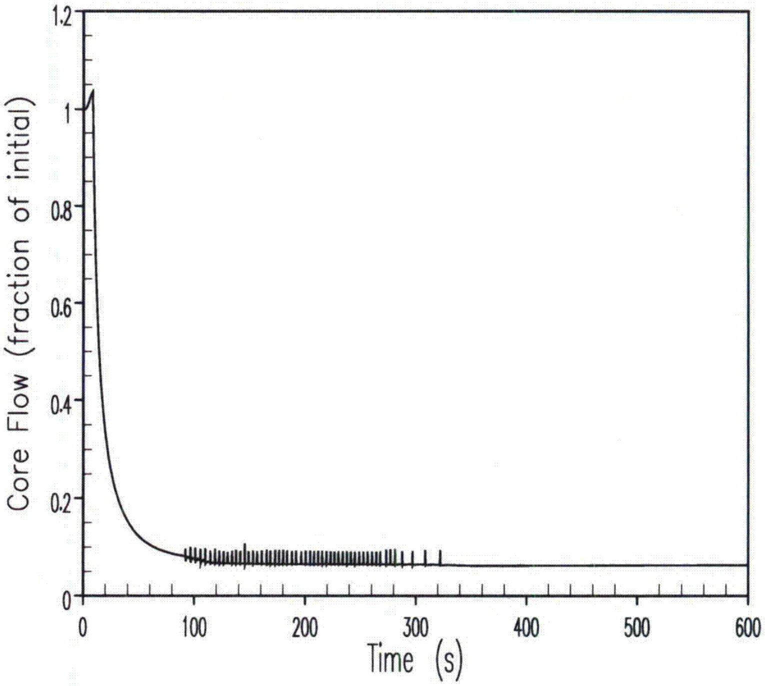

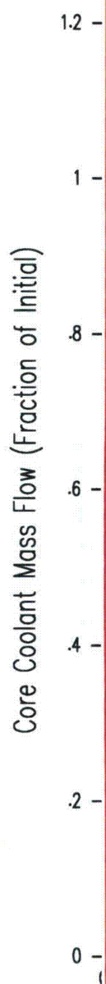

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Figure 15.1.5-7

Core Flow Transient Steam System Piping Failure

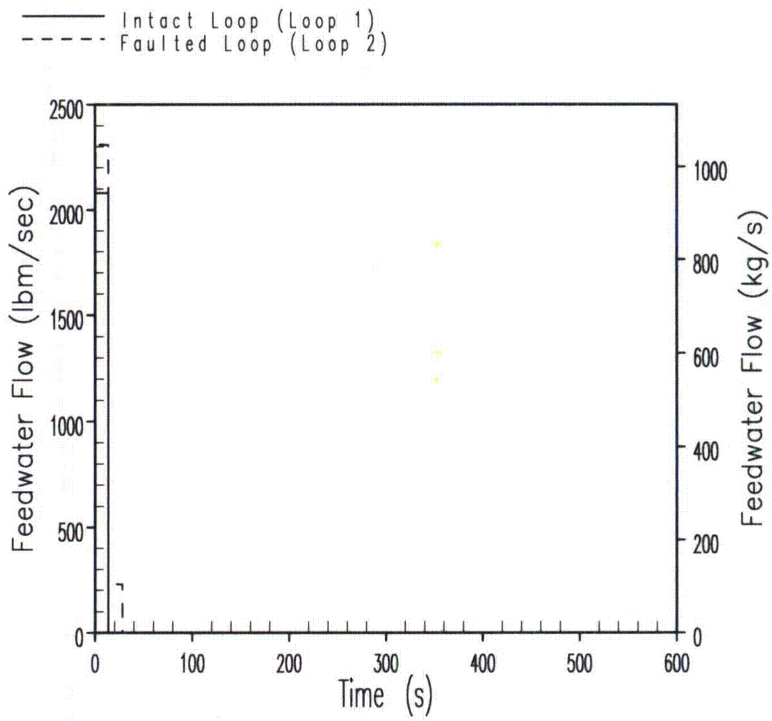
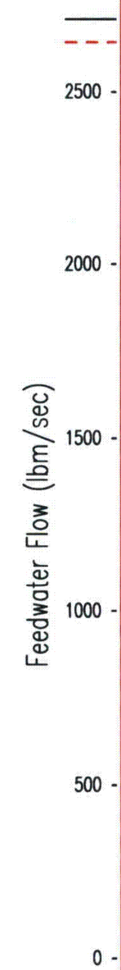


Figure 15.1.5-8

Feedwater Flow Transient Steam System Piping Failure



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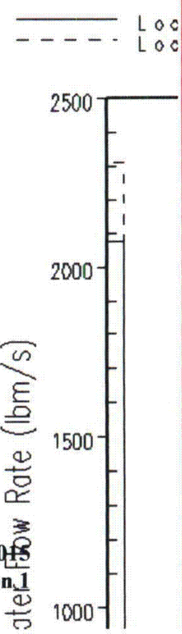

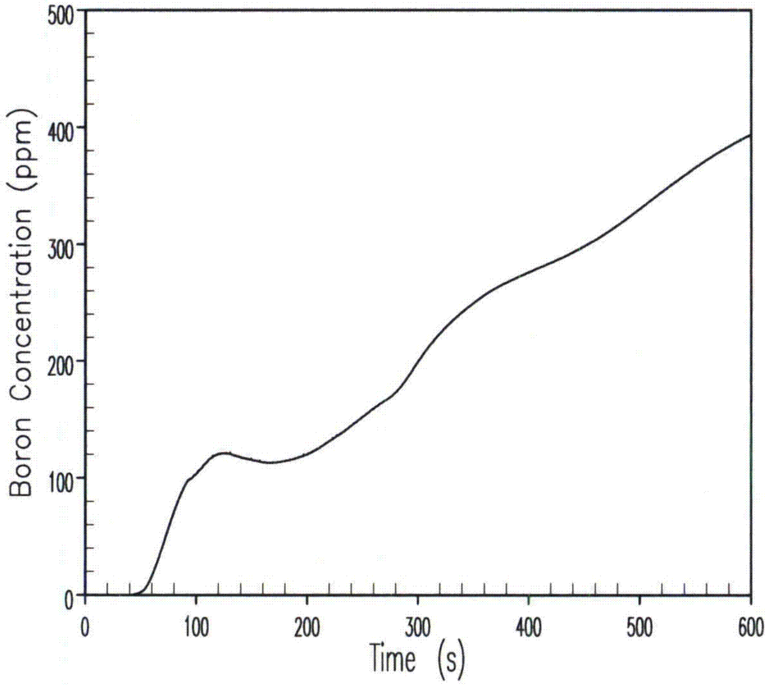

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Figure 15.1.5-9

Core Boron Concentration Transient Steam System Piping Failure

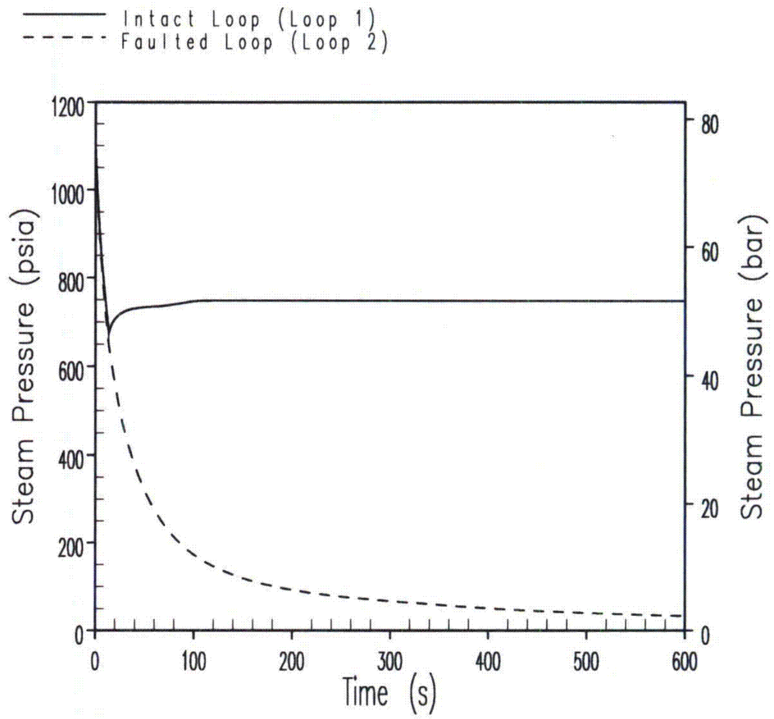
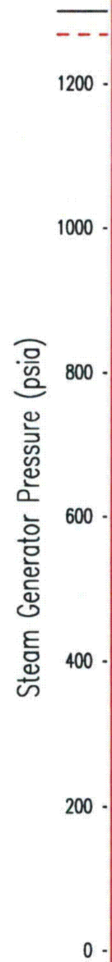


Figure 15.1.5-10

Steam Pressure Transient Steam System Piping Failure



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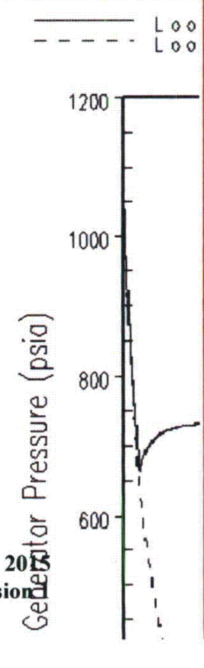


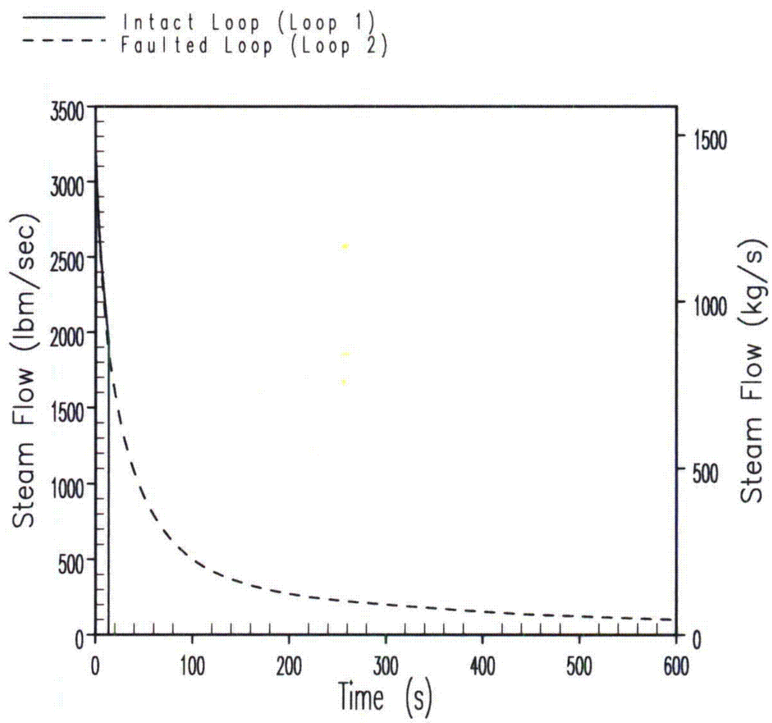
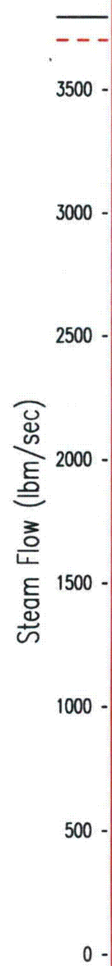


Figure 15.1.5-11

Steam Flow Transient Steam System Piping Failure



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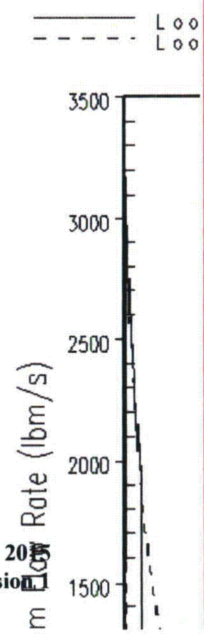


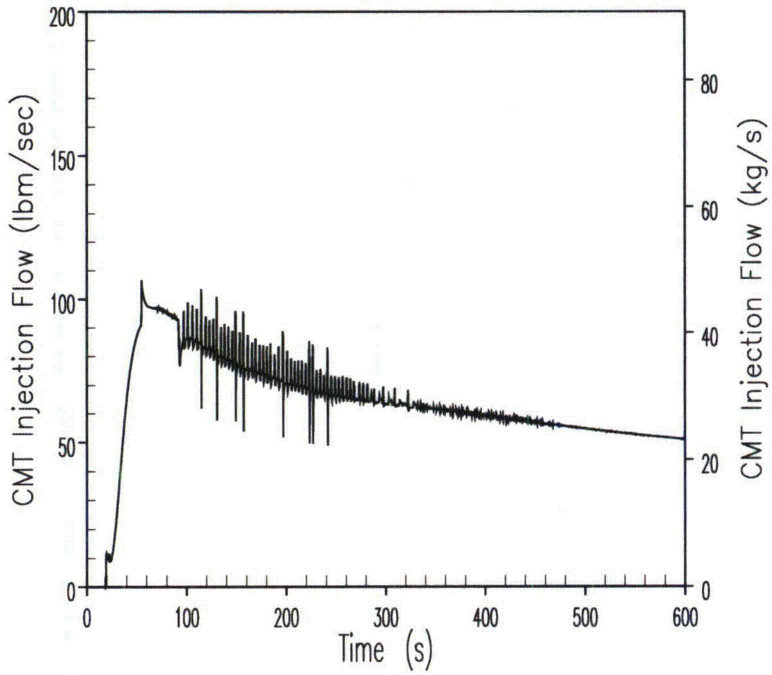
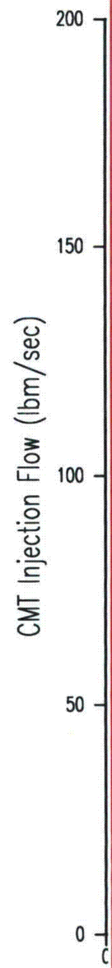


Figure 15.1.5-12

Core Makeup Tank Injection Flow
Steam System Piping Failure



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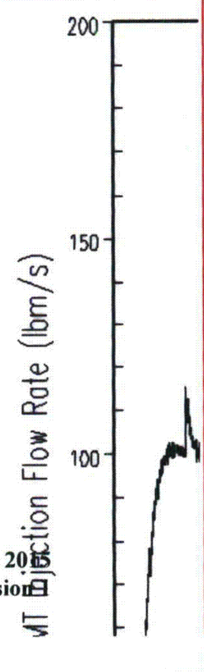

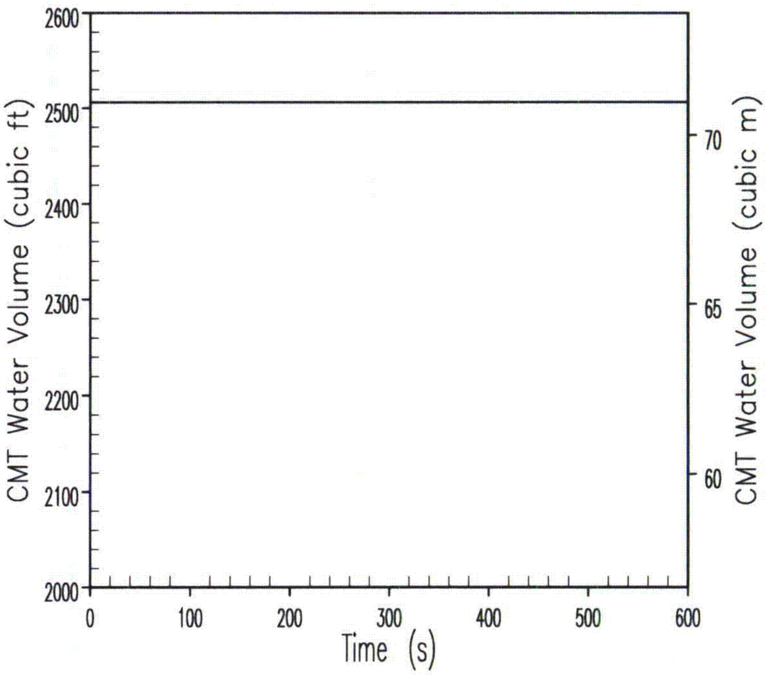

Figure 15.1.5-13

Core Makeup Tank Water Volume Steam System Piping Failure

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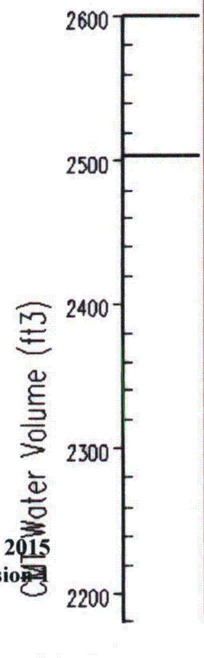

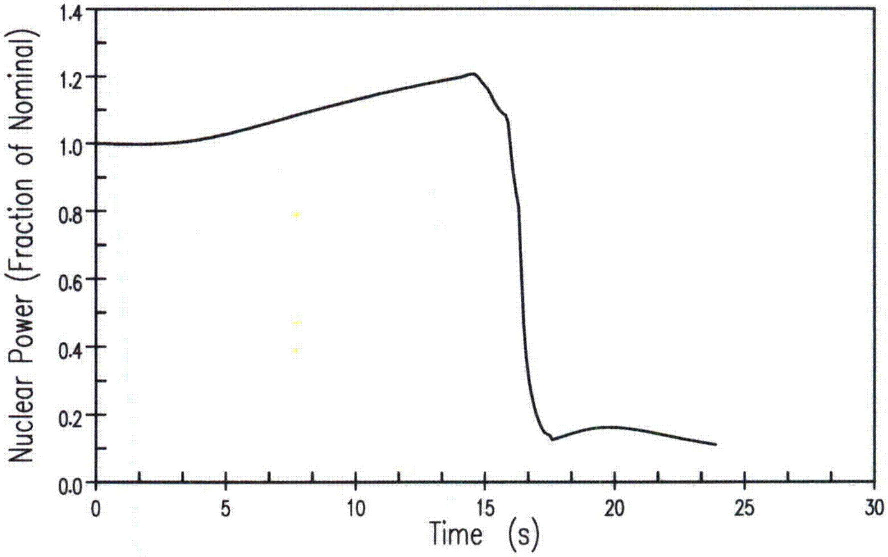

Figure 15.1.5.5-1
Nuclear Power Transient
Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

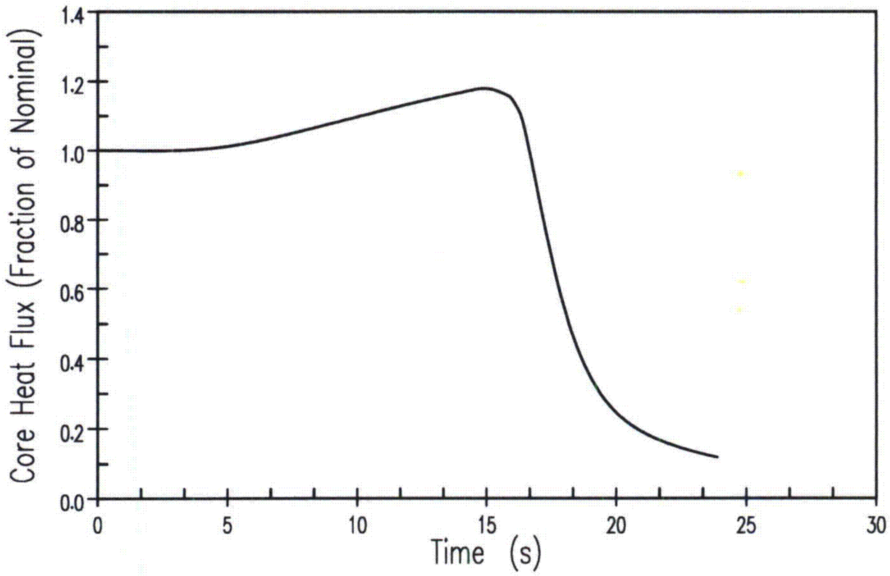

Figure 15.1.5.5-2

Core Heat Flux Transient

Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

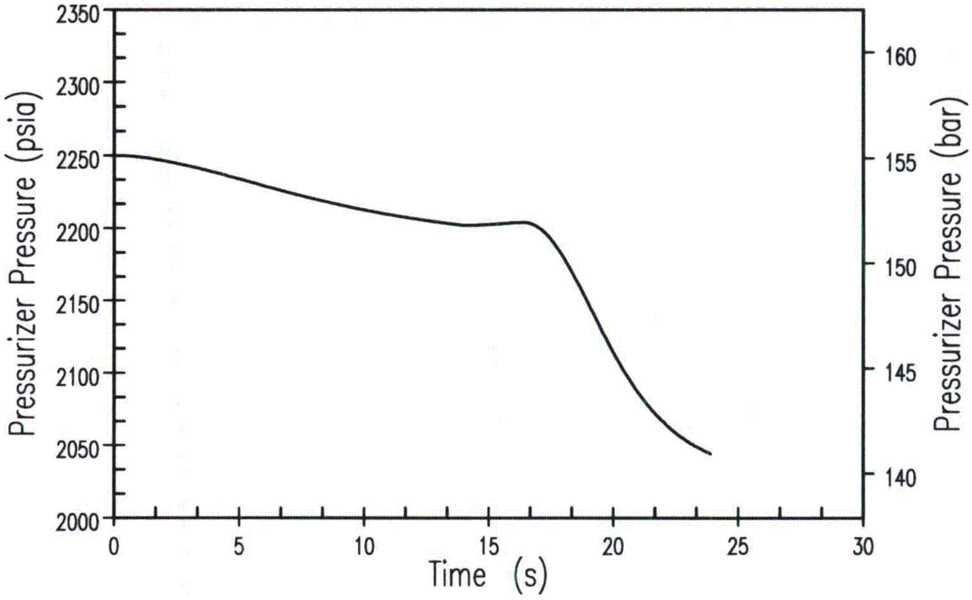

Figure 15.1.5.5-3
Pressurizer Pressure Transient
Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

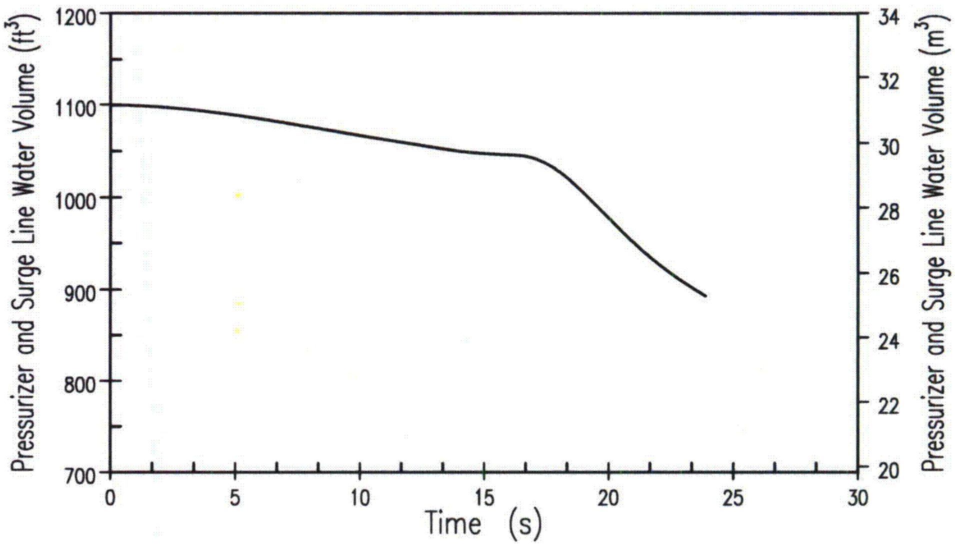

Figure 15.1.5.5-4

Pressurizer Water Volume Transient

Steam System Piping Failure at Full Power – 0.87 ft² Break Size

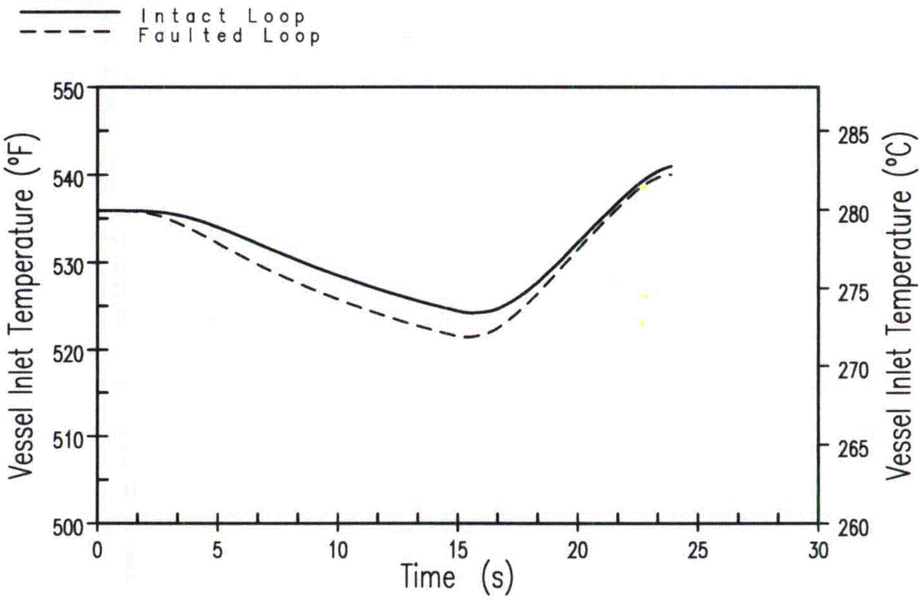

Figure 15.1.5.5-5

Vessel Inlet Temperature Transient

(Intact and Faulted Loops)

Steam System Piping Failure at Full Power – 0.87 ft^2 Break Size

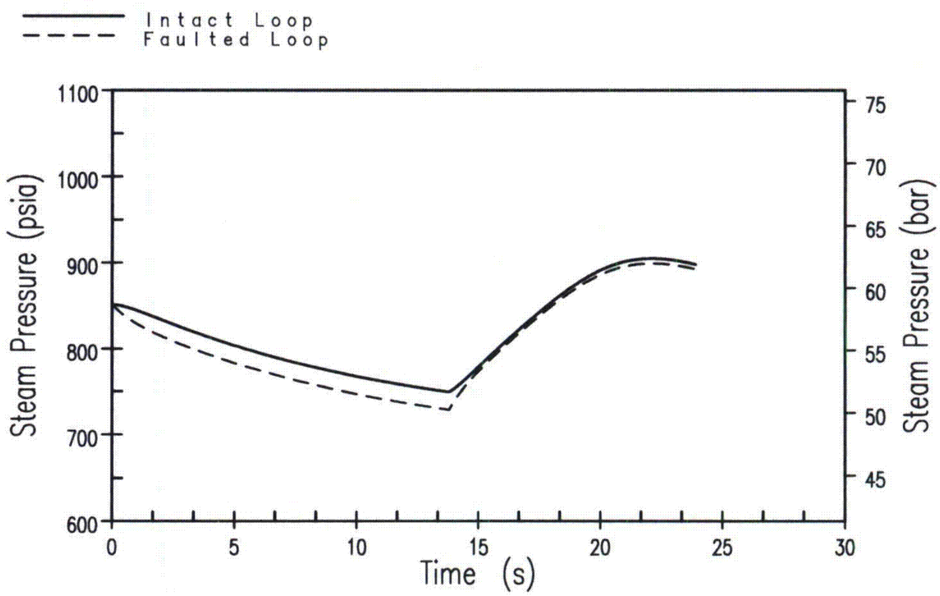

Figure 15.1.5.5-6

Steam Generator Pressure Transient

(Intact and Faulted Loops)

Steam System Piping Failure at Full Power – 0.87 ft² Break Size

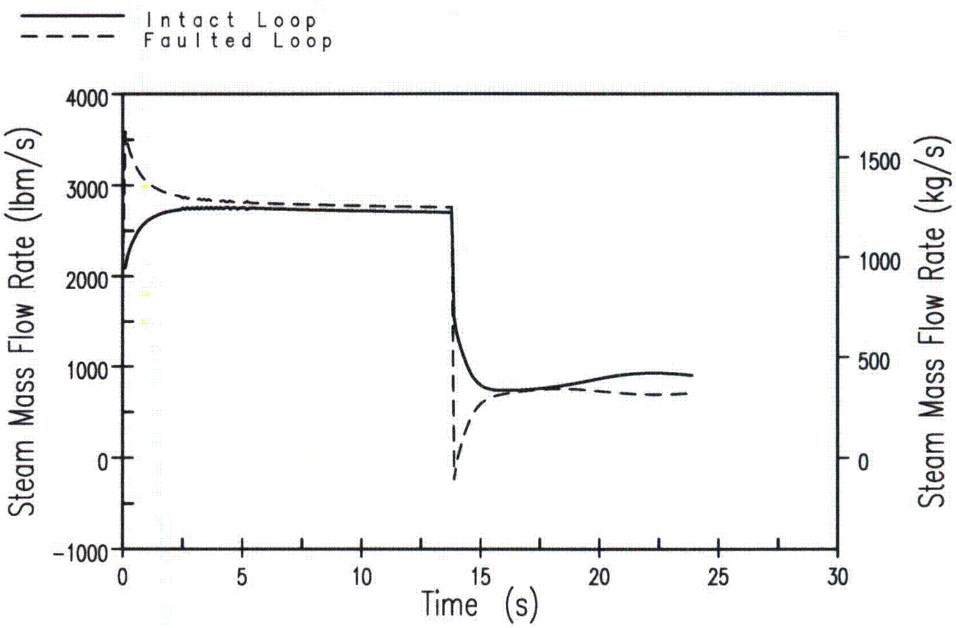

Figure 15.1.5.5-7

Steam Flow Transient(Intact and Faulted Loops)

Steam System Piping Failure at Full Power – 0.87 ft² Break Size

Figures 15.1.6-1 through 15.1.6-8 not used.