Materials Reliability Program: Characterization of U.S. Pressurized Water Reactor (PWR) Fleet Operational Transients (MRP-393)

2014 TECHNICAL REPORT

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Abstract

Field data from the French pressurized water reactor (PWR) fleet and laboratory data from Halden Research Program strongly suggest that current irradiation assisted stress corrosion cracking (IASCC) initiation models based on laboratory data developed from static testing may underpredict crack initiation if significant variation in component loading occurs during plant operation. These observations support a systematic evaluation of the effects of dynamic straining on IASCC initiation rates in order to better understand the significance of such effects for the operating fleet.

The objective of this Materials Reliability Program (MRP) report is to characterize the operational transients that affect the U.S. PWR fleet. This information will subsequently be used to support an international collaborative project studying the effect of transients on IASCC initiation. The results may also be used at a later date to further quantify the effect of transients on specific reactor internals components such as bolts.

The information used to characterize the transients comes from actual plant data collected for the purpose of fatigue monitoring. As such, it reflects the typical, actual severity of transients, as opposed to conservative, idealized definitions used for structural design. In addition, while several events are postulated as contingencies for conservative design analyses, some rarely, if ever, occur in practice. Thus, the only events presented here are ones that normally occur in plants and for which data are available.

Keywords
IASCC initiation
Operational transients
Fatigue monitoring
Plant heatup
Plant cooldown

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Section 1: Introduction

Field data from the French Pressurized Water Reactor (PWR) fleet and laboratory data from Halden Research Program strongly suggest that current Irradiation Assisted Stress Corrosion Cracking (IASCC) initiation predictive models based on laboratory data developed from static testing may under predict initiation if significant variation in component loading occurs during plant operation. Specifically, cracking observed in baffle bolts at some plants in the French fleet has occurred at fluence values well below the current predictive curves. However, this behavior has not been observed at other plants in the French fleet, nor has it been observed in the U.S. fleet. In the laboratory, IASCC crack initiation has been observed following inadvertent load transients. These observations support a systematic evaluation of the effects of dynamic straining on IASCC initiation rates to better understand the significance of such effects to the operating fleet.

The objective of this technical document is to characterize the operational transients that affect the U.S. PWR fleet. This information will subsequently be used to support an international collaborative project studying the effect of transients on initiation of IASCC. The results may also be used at a later date to further quantify the effect of the transients on specific reactor internals components such as bolts.

The information used to characterize the transients comes from actual plant data collected for the purpose of fatigue monitoring. As such, it reflects the typical, actual severity of transients, as opposed to conservative, idealized definitions used for structural design. In addition, while several events are postulated as contingencies for conservative design analyses, in practice some rarely if ever occur in practice. Thus, only the events that normally occur in plants and where data is available are presented here.

Section 2: Westinghouse Designed PWR Transients

Transients affecting reactor pressure vessels in Westinghouse plants are listed in the final safety analysis report (FSAR) and in the surveillance procedures for each such U.S. plant. The Reference [1] license renewal application, publically available on the U.S. NRC website, provides an example listing of those transients, along with numbers of design cycles and the actual cycles projected to 60 years. Transient names and typical design cycles related to the reactor pressure vessel are listed in Table 2-1. Plant data is available to characterize the transients including the Reactor Coolant System (RCS) hot and cold leg temperatures (indicated as THOT and TCOLD in the sections that follow), the primary system pressure, and the reactor power level in percent.

Table 2-1
Westinghouse Transients Affecting RPV

Transient Description	FSAR Design Cycles
Normal Condition Transients	
Plant heatup at 100°F/hr	200
Plant cooldown at 100°F/hr	200
Unit loading at 5% of full power per min	13,200
Unit unloading at 5% of full power per min	13,200
Step increase 10% of full power	2,000
Step decrease 10% of full power	2,000
Large step decrease with steam dump	200
Steady state fluctuations, Initial	1.5E5
Steady state fluctuations, Random	3.0E6
Feedwater cycling at hot shutdown	2000
Loop out of service	80
Unit loading between 0 and 15% of full power	500
Unit unloading between 0 and 15% of full power	500
Boron concentration equalization	26,400

Table 2-1 (continued)
Westinghouse Transients Affecting RPV

Transient Description	FSAR Design Cycles
Reactor Coolant Pump Operation	Various
Reduced Temperature Return to Power	2000
Refueling	80
Turbine Roll Test	20
Primary Side Leak Test	200
Secondary Side Leak Test	80
Upset Condition Transients	
Loss of load	80
Loss of power	40
Partial loss of flow	80
Reactor Trip from full power	400
Inadvertent RCS depressurization	20
Inadvertent startup of inactive loop	10
Control rod drop	80
Inadvertent safety injection actuation	60
Excessive feedwater flow	30

2.1 Normal Condition Transients

2.1.1 Plant Heatup

For all PWRs, heatup operations of the Reactor Coolant System are generally limited to a maximum rate of change of 100°F/hr due to material ductility considerations. Several different methods of heatup procedures are used; however, for analysis purposes it is generally conservative to represent the transient as a constant temperature increase from cold shutdown temperature to the no load temperature.

Plants are typically designed to accommodate 200 of these events in a 40 year life. Actual accumulation rates generally produce estimates of around 60 to 200 cycles at 60 years of operation, with refueling outages now typically taking place every 18 months.

Temperature and pressure time histories of a typical transient are shown in Figure 2-1. Table 2-2 provides typical operational transient parameters for the example provided. Power production occurs following the event during the unit loading events discussed below.

Table 2-2
Plant Heatup – Typical Operational Parameters

Minimum Power	0	%
Maximum Power	0	%
Minimum THOT	116.9	°F
Maximum THOT	557.4	°F
Minimum TCOLD	115.8	°F
Maximum TCOLD	558.3	°F
40 Year Cycles	200	

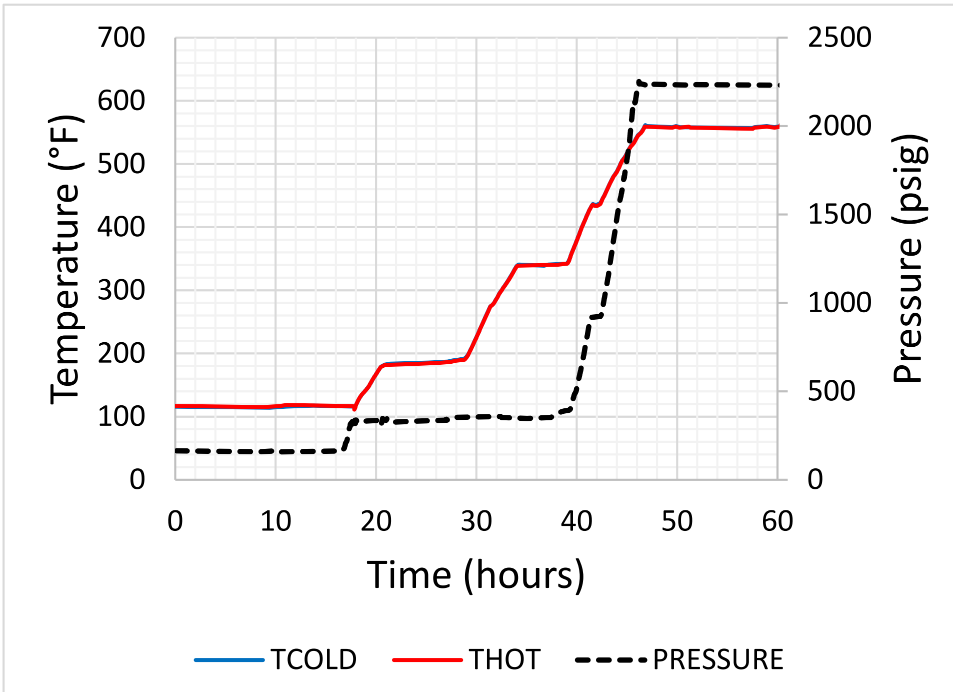

Figure 2-1
Plant Heatup – RCS Temperatures and Pressure

2.1.2 Plant Cooldown

Cooldown operations of the Reactor Coolant System are also generally limited to a maximum rate of change of 100˚F/hr . For analysis purposes it is generally conservative to represent the transient as a constant temperature decrease from the no load temperature to cold shutdown.

Plants are typically designed to accommodate 200 of these events in a 40 year life. Actual accumulation rates generally produce estimates of around 60 to 200 cycles at 60 years of operation, with refueling outages now typically taking place every 18 months.

Temperature and pressure time histories of a typical transient are shown in Figure 2-2. Table 2-3 provides typical operational transient parameters for the example provided.

Table 2-3
Plant Cooldown – Typical Operational Parameters

Minimum Power	0	%
Maximum Power	0	%
Minimum THOT	170.1	°F
Maximum THOT	558.1	°F
Minimum TCOLD	171.1	°F
Maximum TCOLD	557.3	°F
40 Year Cycles	200	

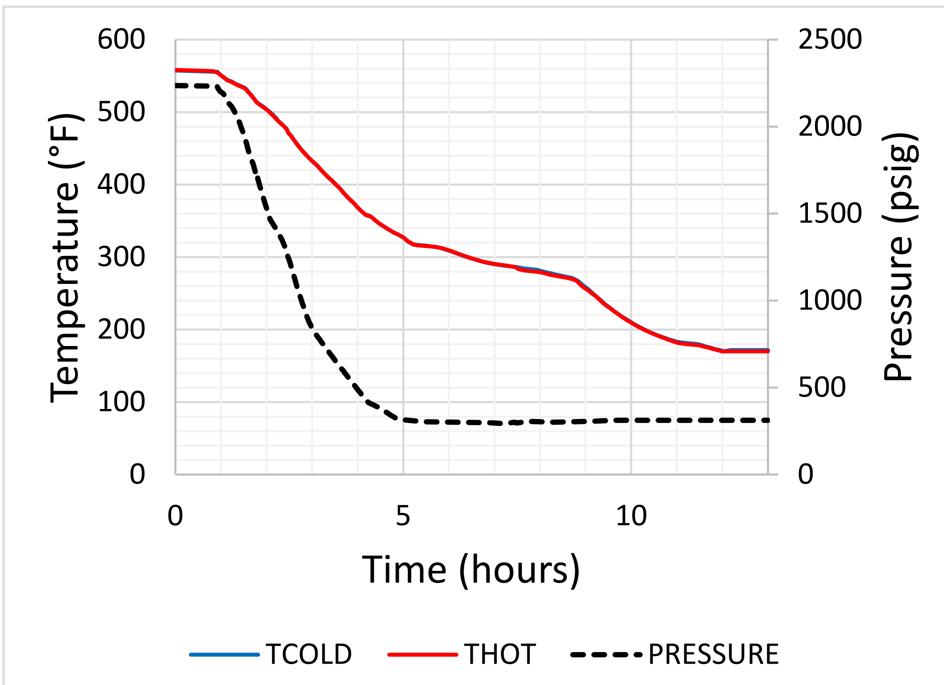

Figure 2-2
Plant Cooldown – RCS Temperatures and Pressure

2.1.3 Unit Loading at 5% of Full Power per Minute

This event is counted whenever there is a gradual increase in reactor power (average rate ≤ 5% per minute) while reactor power is above 15%. Actual rates typically occur much more slowly, as shown below.

Plants are typically designed to accommodate 11,200 of these events. However, because nuclear plants in the U.S. do not typically load follow, the actual number is far fewer (e.g. << 1000).

Power and temperature time histories of a typical transient are shown in Figure 2-4. Table 2-4 provides typical operational transient parameters for the example provided. For the most conservative calculation of stresses, it can be assumed that the temperature excursions below occur much quicker, coincident with a maximum 5% power increase per minute from 15 to 100% power.

Table 2-4
Unit Loading 5% of Full Power per Minute – Typical Operational Parameters

Minimum Power	15	%
Maximum Power	100	%
Minimum THOT	566.4	°F
Maximum THOT	617.05	°F
Minimum TCOLD	552.85	°F
Maximum TCOLD	557.98	°F
40 Year Cycles	13,200	

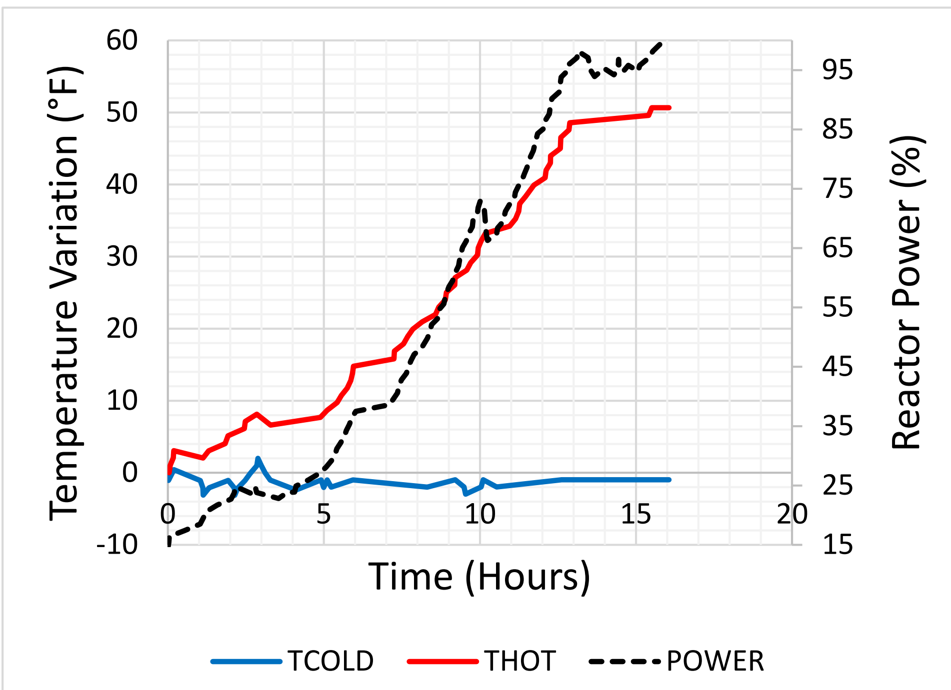

Figure 2-3
Unit Loading at 5% of Full Power per Minute – RCS Temperatures and Power

2.1.4 Unit Unloading at 5% of Full Power per Minute

This event is counted whenever there is a gradual decrease in reactor power (average rate ≤ 5% per minute) while reactor power is above 15%. Actual rates typically occur much more slowly, as shown below.

The plant is typically designed to accommodate 13,200 of these events. However, because nuclear plants in the U.S. do not typically load follow, the actual number projected to 60 years of operation is far fewer (e.g. << 1000).

Power and temperature time histories of a typical transient are shown in Figure 2-4. Table 2-5 provides typical operational transient parameters for the example provided. For the most conservative calculation of stresses, it can be assumed that the temperature excursions below occur much quicker, coincident with a maximum 5% power decrease per minute from 100 to 15% power.

Table 2-5
Unit Unloading 5% of Full Power per Minute – Typical Operational Parameters

Minimum Power	14.7	%
Maximum Power	100.7	%
Minimum THOT	569.02	°F
Maximum THOT	618.52	°F
Minimum TCOLD	552.55	°F
Maximum TCOLD	556.65	°F

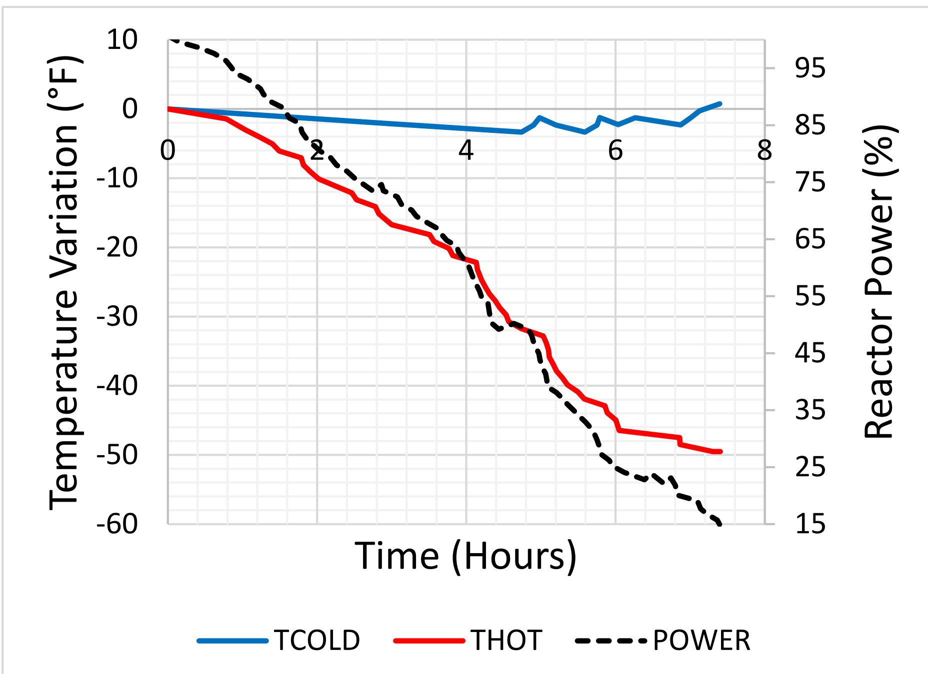

Figure 2-4
Unit Unloading at 5% of Full Power – RCS Temperatures and Power

2.1.5 Step Increase and Decrease 10% of Full Power

This event occurs when there is a sudden increase or decrease in turbine load demand of 10% or less, while reactor power is above 15%. This is typically a result of disturbances in the electrical grid while the plant is on-line. The automatic reactor control system, restores equilibrium in the RCS by matching reactor power to the turbine demand, while avoiding a reactor trip.

A plant is typically designed to accommodate 2,000 of these events (or 50 per year). However, actual events are rare. Monitoring systems may conservatively count these events when the power instrument reading artificially fluctuates. Postulated events may be conservatively characterized as Large Step Load Decrease with Steam Dump described in Section 2.1.6. Typical operating parameters for an actual event are shown in Table 2-6 and Table 2-7.

Table 2-6
Step Load Increase – Typical Operating Parameters

Minimum Power	90	%
Maximum Power	100	%
Minimum THOT	625	°F
Maximum THOT	619	°F
Minimum TCOLD	557	°F
Maximum TCOLD	551	°F

Table 2-7
Step Load Decrease – Typical Operating Parameters

Minimum Power	90	%
Maximum Power	100	%
Minimum THOT	612	°F
Maximum THOT	619	°F
Minimum TCOLD	557	°F
Maximum TCOLD	563	°F

2.1.6 Large Step Load Decrease with Steam Dump

This event occurs when there is a large step decrease in turbine load and reactor power level. The plant is designed to accommodate a large (up to 50%) step decrease in power using the steam dump as a heat sink without initiating a reactor trip or steam generator safety valve actuation.

The plant is typically designed to accommodate 200 of these events, or 5 per year. However, actual number of cycles at 60 years is typically less than 20.

Power, pressure, and temperature time histories of a typical transient are shown in Figure 2-5 through Figure 2-7. Table 2-8 provides typical operational transient parameters for the example provided. For the most conservative calculation of stresses, it can be assumed that the temperature excursions below occur much quicker, coincident with a maximum 5% power decrease per minute from 100 to 15% power.

Table 2-8
Large Step Load Decrease – Typical Operational Parameters

Minimum Power	5	%
Maximum Power	100	%
Minimum THOT	549.13	°F
Maximum THOT	616.51	°F
Minimum TCOLD	550.81	°F
Maximum TCOLD	560.51	°F

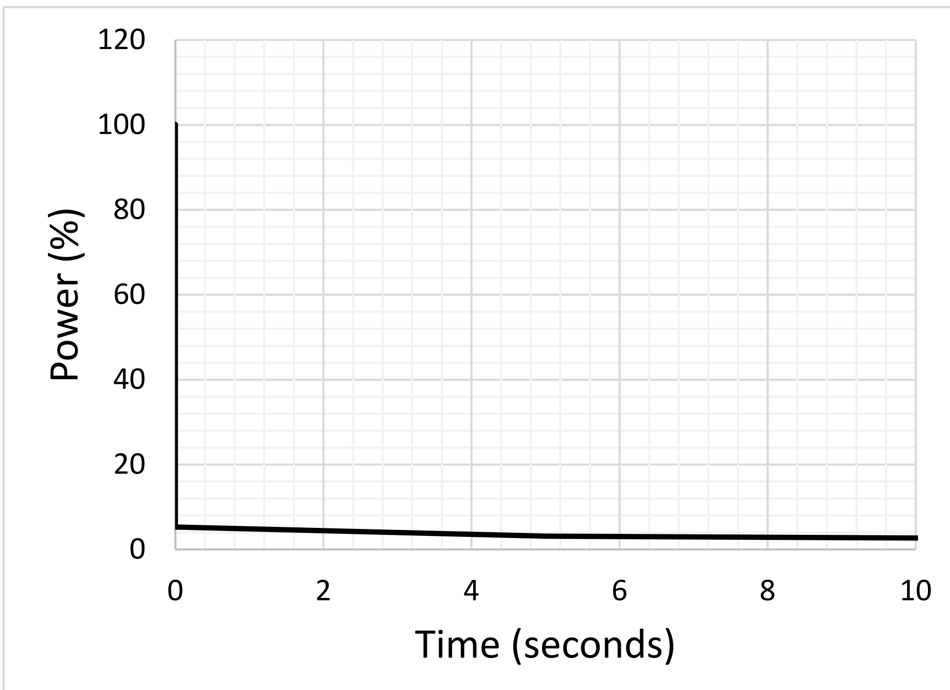

Figure 2-5
Large Step Load Decrease - Power

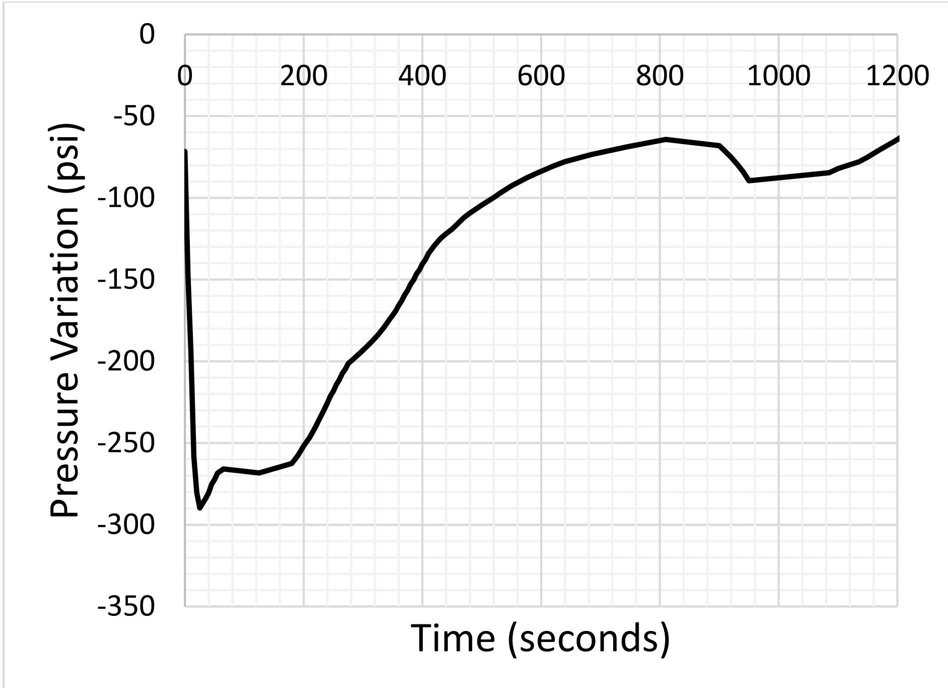

Figure 2-6
Large Step Load Decrease – RCS Pressure

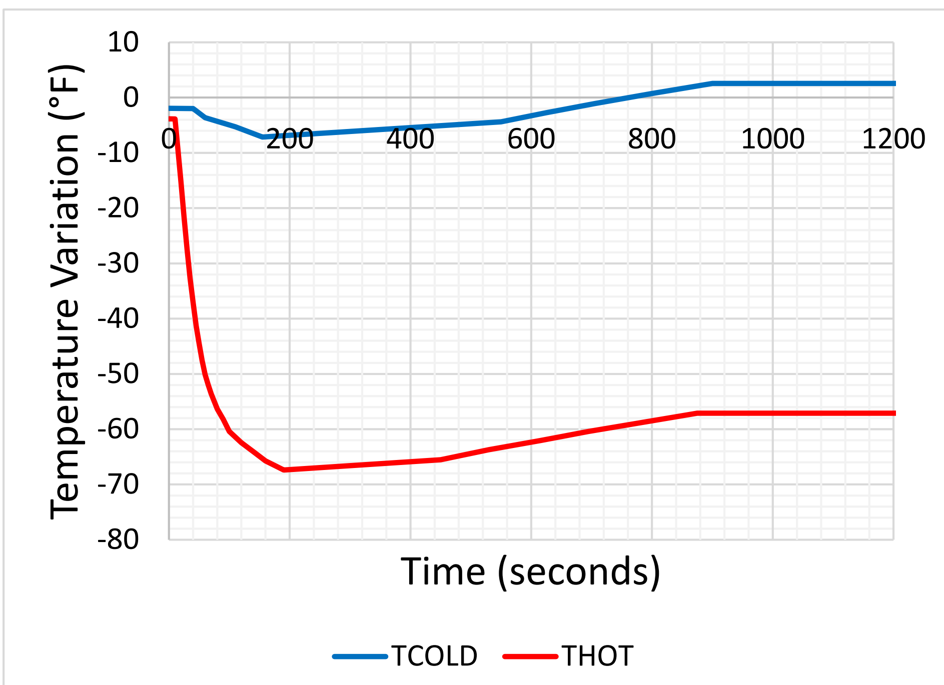

Figure 2-7
Large Step Load Decrease – RCS Temperatures

2.1.7 Steady State Fluctuations

A large number of steady state fluctuations in RCS temperature and pressure are conservatively assumed for design purposes. Actual data shows only small fluctuations (< ± 0.5 ˚F) at normal operation. The effects may therefore be assumed to be negligible.

2.1.8 Feedwater Cycling

Feedwater cycling occurs when cold auxiliary feedwater is introduced intermittently (slug feeding) into a relatively hot steam generator (S/G) during hot standby, no load, heatup or cooldown conditions. Although the design transients assume conservative hot and cold leg temperature excursions, the effects of feedwater cycling on the primary side of the plant are usually benign. As can be seen in Figure 2-8, significant feedwater cycling is occurring, as indicated by auxiliary feedwater flow instruments being cycled on and off during the hot standby period. Although the impact of this cycling is significant on the secondary side of the plant, the measured hot and cold leg temperature fluctuations on the primary side are essentially negligible (< ±2˚F).

2000 cycles of feedwater cycling are typically postulated for the 40 year design life. Actual cycles at 60 years vary significantly among plants and can be much less than or in excess of the design values. However, as discussed above, impacts of feedwater cycling on the reactor pressure vessel are expected to be negligible. No power is produced during these events.

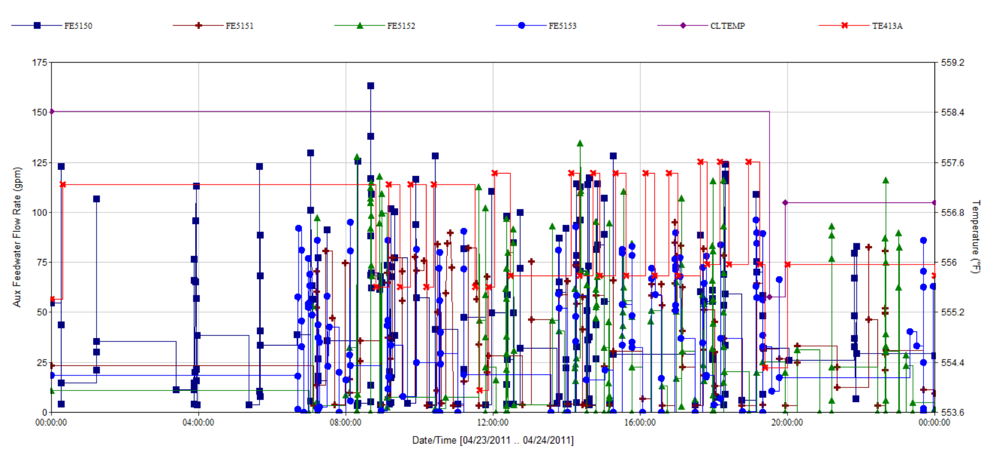

Figure 2-8
Feedwater Cycling – RCS Temperatures

2.1.9 Loop Out of Service

Plants with 3 and 4 loops are capable of operating at lower power levels with one loop out of service while the others are running. Few to no U.S. plants are believed to be licensed to operate in this manner. As such, no measured data of these transients are available.

80 cycles of are typically postulated for the 40 year design life (2 per year). Actual cycles at 60 years are expected to be zero for all U.S. Westinghouse designed plants.

2.1.10 Unit Loading Between 0 and 15% of Full Power

This event reflects the transient changes that occur at the start of power ascension, from hot standby (0% power) to low-power operation (up to 15% power). This is a normal event, assumed to occur at a steady rate over approximately 30 minutes. However, power ascension typically moves at a slower rate and may include interruptions, as shown below.

500 cycles are typically postulated for design purposes, however actual plant cycles projected to 60 years of operation may be considerably less (e.g., less than 200).

Power and temperature time histories of a typical transient are shown in Figure 2-9. Table 2-9 provides typical operational transient parameters for the example provided.

Table 2-9
Unit Loading Between 0 and 15% of Full Power – Operational Parameters

Maximum Power	15%
Minimum Power	0%
40 Year Cycles	500

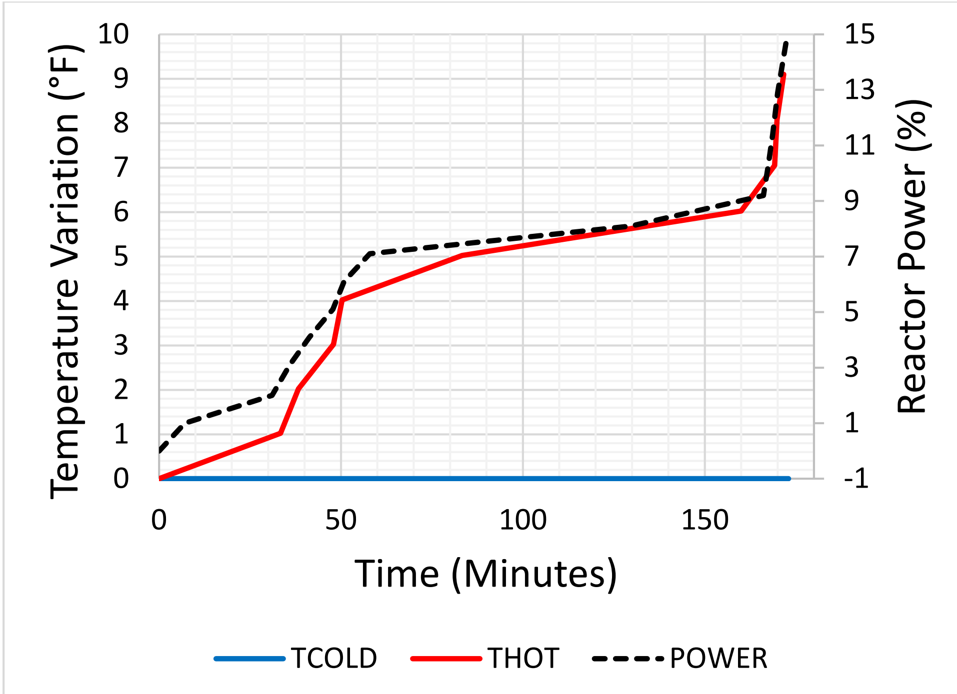

Figure 2-9
Unit Loading Between 0 and 15% of Full Power – Reactor Coolant Temperatures and Power

(Note: RCS pressure remains constant at normal operating value. Temperatures start at no-load conditions; e.g. 557°F.)

2.1.11 Unit Unloading Between 0 and 15% of Full Power

This event reflects the end of a normal power reduction, from low-power operation (~ 15% power) to hot standby (0% power). This is a normal event, assumed to occur at a steady rate over approximately 5 minutes. Actual rates are typically slower.

500 cycles are typically postulated for design purposes, however actual plant cycles projected to 60 years of operation may be considerably less (e.g., less than 200).

Power and temperature time histories of a typical transient are shown in Figure 2-10. Table 2-10 provides typical operational transient parameters for the example provided.

Table 2-10
Unit Unloading Between 0 and 15% of Full Power – Operational Parameters

Maximum Power	15%
Minimum Power	0%
40 Year Cycles	500

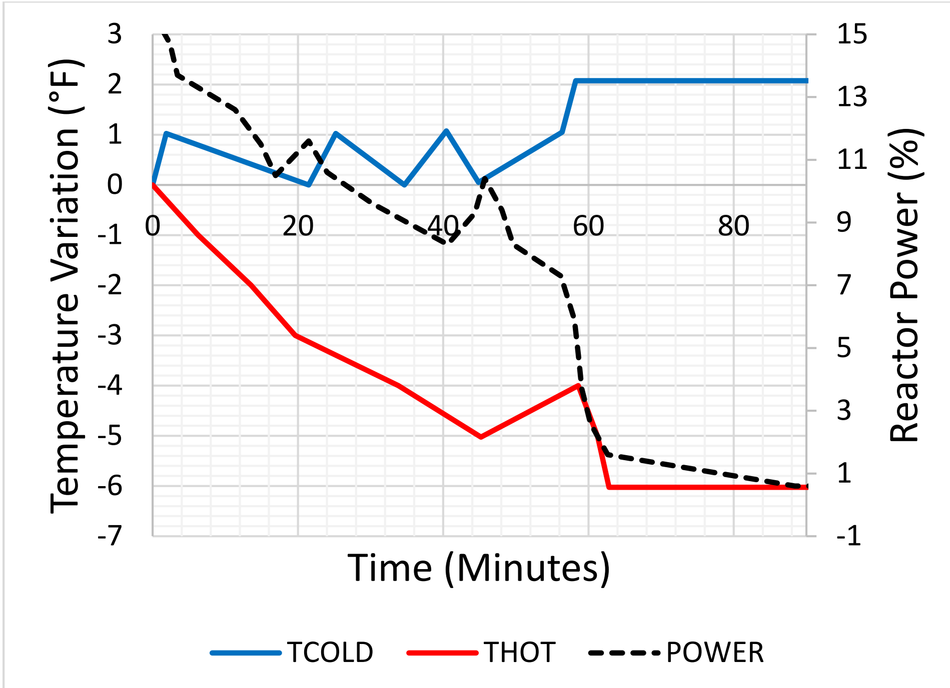

Figure 2-10
Unit Unloading Between 0 and 15% of Full Power – Reactor Coolant Temperatures and Power

(Note: RCS pressure remains constant at normal operating value. Hot and cold legs start at normal load program temperatures at the 15% power level.)

2.1.12 Boron Concentration Equalization

26,400 of these cycles are typically postulated for the 40 year design, based on assumed twice daily load following cycles. Few to no U.S. PWRs currently load follow, and as such do not typically count these cycles.

Actual cycles at 60 years are expected to be zero for most U.S. Westinghouse designed plants. Moreover, the effects of this transient are negligible with respect to temperature excursions in the reactor pressure vessel. Only RCS pressure varies slightly due to the spray cycling that occurs during the event.

2.1.13 Reactor Coolant Pump Startup and Shutdown

Reactor Coolant Pump (RCP) start and stop operations are normal operations considered in the design of the plant. Although this has an effect on the flow and the pressure downstream of the RCP, the impact on RCS temperature is negligible.

3,800 cycles of various pump operations are typically postulated for design. However, the impact on the reactor pressure vessel temperature and power levels is negligible.

2.1.14 Reduced Temperature Return to Power

This event pertains to plants that perform load-follow operations. U.S. PWRs do not typically count these cycles, because they do not perform load following, and example transients based on plant data are not available.

2,000 cycles of are typically postulated for the 40 year design life (1 per week). Actual cycles at 60 years are expected to be zero for all U.S. Westinghouse designed plants.

2.1.15 Refueling

This event encapsulates several load cases and operating conditions that occur in conjunction with a typical refueling outage. It accounts for a zero load state in the fatigue analyses for various RCS components, initiation of flow from the Refueling Water Storage Tank (RWST) into the loops at cold conditions, and unloading and reloading of the fuel.

80 cycles are postulated for the 40 year design life (twice per year). Refueling outages typically occur at 18 month intervals. Because the power is zero and temperatures are cold during this transient, it is considered to have a negligible effect.

2.1.16 Turbine Roll Test

This event is intended to occur only as part of hot functional testing, prior to initial plant operation. The RCS is heated by operating the reactor coolant pumps, prior to rolling the turbine using the secondary side. The plant cooldown rate during the turbine roll can exceed the 100°F/hr design rate for normal cooldowns.

Because the tests are typically performed only during initial startup, no plant data from monitoring systems are available. 20 cycles are typically postulated for design, but actual cycles for most plants are expected to be less than 10. For analysis purposes, a reactor temperature drop from the no-load operating temperature of approximately 100˚F at 400 ˚F/hr may be assumed. No power is produced during this operation.

2.1.17 Primary Side Leak Test

The primary side leak test was originally intended to be performed after each opening of the primary RCS. During this test, the RCS pressure is raised to at least 2335 psig (approx. design pressure), while the temperature is simultaneously raised to maintain margin to the RPV material ductility requirements.

200 cycles are typically postulated for the 40 year design. However, this test is not typically performed on a routine basis. 10 or fewer at 60 years is a typical cycle projection.

For analysis purposes, it may be assumed that the reactor temperature rises at 100°F/hr from 70°F to 300°F. During the same period, the reactor pressure may be assumed to rise from 0 to 2335 psig. No power is produced during this operation.

2.1.18 Secondary Side Leak Test

A Secondary Side Leak Test pressurizes the secondary side of the steam generator to check for leaks such as in the manway closure. 80 cycles are typically assumed for the 40 year design. In practice, the test is rarely performed. Actual cycles at 60 years are expected to be less than 10.

For analysis purposes, it may be assumed that the reactor temperature rises at 100°F/hr from 70°F to 250°F. During the same period, the reactor pressure may be assumed to rise from 0 to 630 psig. No power is produced during this operation.

2.2 Upset Condition (Occasional) Transients

2.2.1 Loss of Load

This event occurs when there is a large, rapid step-decrease in turbine load from full power without an immediate reactor trip (due to a failure of the automatic reactor trip). The reactor trip eventually occurs several seconds later from the high pressurizer water level signal.

80 cycles are typically postulated for design purposes, however, in practice the event is rare. A review of multiple monitoring systems provided no instrument data available for an actual transient. Temperature excursions for the event are expected to be fairly minor (< ±50°F). Characterizing the event as a reactor trip is reasonable, however the rate of occurrence is expected to be essentially zero per year.

2.2.2 Loss of Power

This transient involves the loss of all offsite power to the station when the plant is operating followed by reactor and turbine trips. 40 cycles are typically postulated for the 40 year design (once per year). However, similar to Loss of Load, the event rarely occurs in actuality, and example data is not generally available. Temperature excursions are expected to have a negligible impact on thermal stresses during the transient.

2.2.3 Partial Loss of Flow

This transient involves the loss of RC flow in one loop with the reactor at full power, due to the loss of a reactor coolant pump. This event results in a reactor trip followed by a turbine trip. 80 cycles are typically postulated for the 40 year design (twice per year). Flow reversal in the affected loop pushes cold leg water back through the steam generator (getting even colder) and into the RPV hot leg nozzle causing a large reduction in temperature. However, the event rarely occurs in actuality, and example data is not generally available. Characterizing the event as a reactor trip is reasonable, however the rate of occurrence is expected to be essentially zero per year.

2.2.4 Reactor Trip from Full Power

Reactor Trips immediately stop power production and produce pressure and temperature decreases in the Reactor Coolant System.

400 of these events are typically postulated for design purposes (10 times per year over the 40 year life of the plant). They are categorized into three different cases (A, B, and C) based on the extent of secondary side cooling. Each sub-type is discussed in the sections below.

2.2.4.1 Reactor Trip from Full Power, with No Cooldown (Case A)

The Case A reactor trip requires a RCS pressure drop no greater than a given value (a typical threshold in a monitoring system is 250 psid) and the safety injection signal to not actuate during the events.

230 cycles of the Case A trips are typically postulated for design purposes, however actual plant cycles projected to 60 years of operation may be considerably less (e.g., half the design value).

Pressure and temperature time histories of a typical transient are shown in Figure 2-11 through Figure 2-12. Table 2-11 provides typical operational transient parameters.

Table 2-11
Reactor Trip A – Operational Parameters

Maximum Power	100%
Minimum Power	0%
40 Year Cycles	230

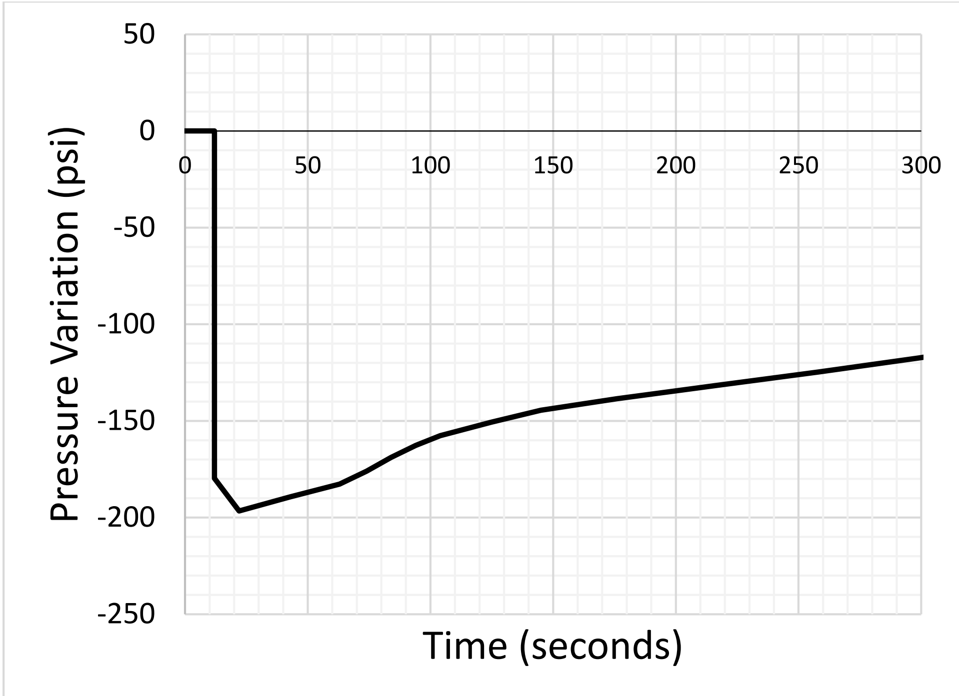

Figure 2-11
Reactor Trip A – Reactor Coolant Pressure

(Note: Power drops from 100% to 0% immediately following reactor trip signal at Time=0. Pressure returns to original value consistent with normal heatup)

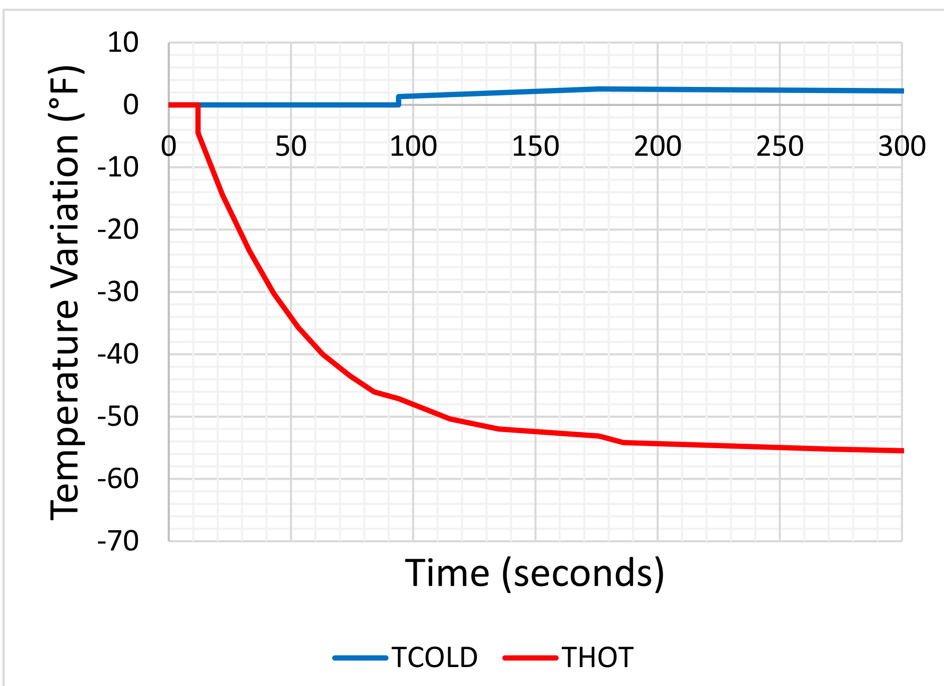

Figure 2-12
Reactor Trip A – Reactor Coolant Temperatures

2.3 Reactor Trip from Full Power, with Cooldown and No Safety Injection (Case B)

The Case B reactor trip requires a pressure drop greater than a given value (a typical threshold in a monitoring system is 250 psid) and the safety injection signal to not actuate during the events.

160 cycles of the Case B trips are typically postulated for design purposes, however actual plant cycles projected to 60 years of operation may be considerably less (e.g., half the design value).

Pressure and temperature time histories of a typical transient are shown in Figure 2-13 through Figure 2-14. Table 2-12 provides typical operational transient parameters.

Table 2-12
Reactor Trip B – Operational Parameters

Maximum Power	100%
Minimum Power	0%
40 Year Cycles	230

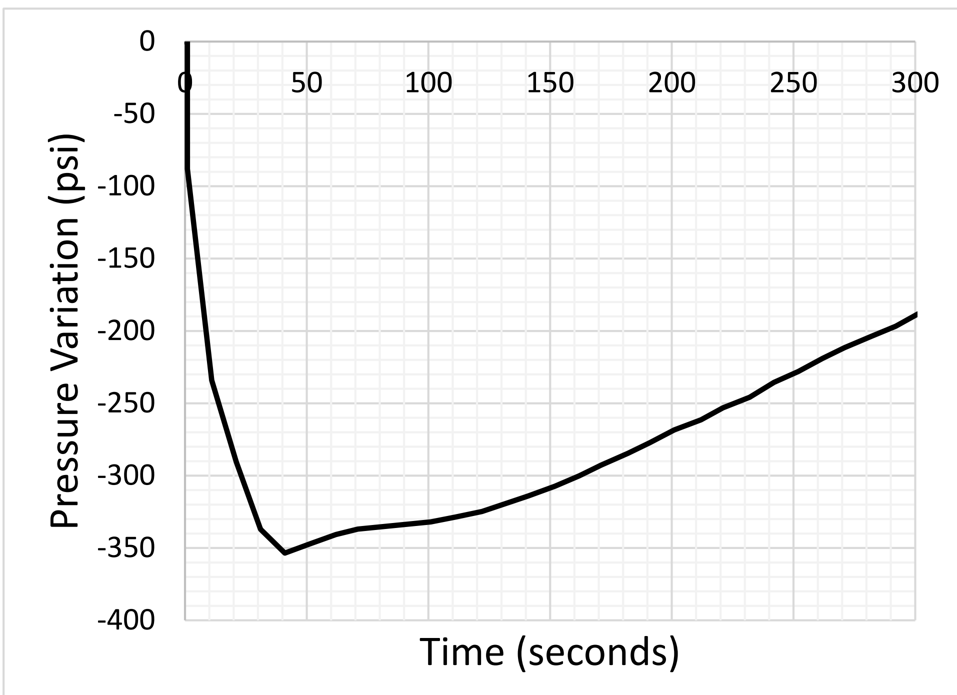

Figure 2-13
Reactor Trip B – Reactor Coolant Pressure

(Note: Power drops from 100% to 0% immediately following reactor trip signal at Time=0. Pressure returns to original value consistent with normal heatup)

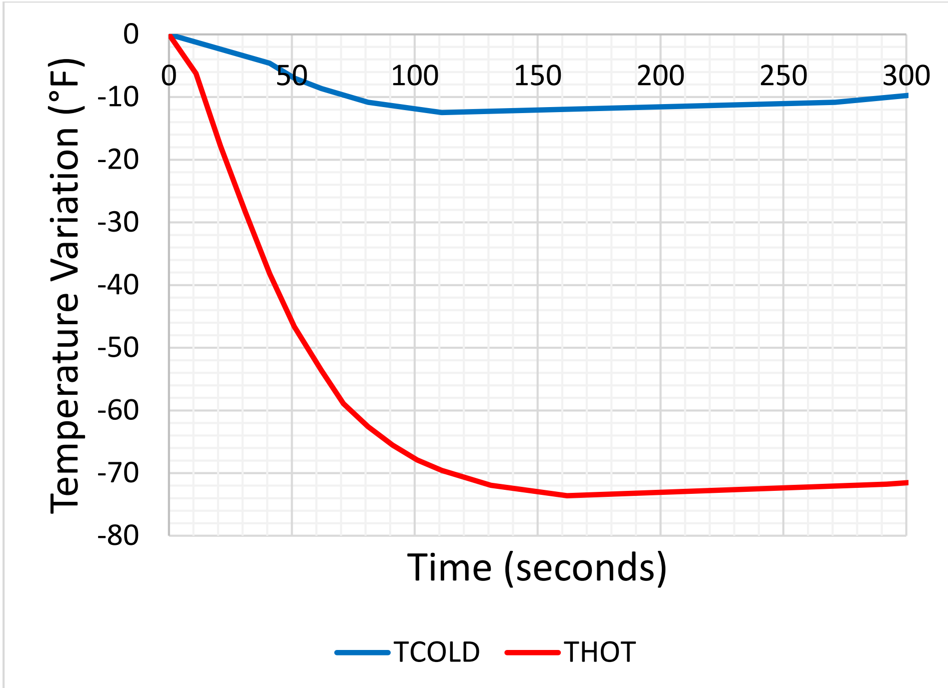

Figure 2-14
Reactor Trip B – Reactor Coolant Temperatures

2.3.1 Reactor Trip from Full Power, with Cooldown and Safety Injection (Case C)

The Case C reactor trip requires the safety injection signal to actuate during the events.

10 cycles of the Case A trips are typically postulated for design purposes, however actual plant cycles projected to 60 years of operation are rare.

Pressure and temperature time histories of an actual transient are shown in Figure 2-15 through Figure 2-16. Table 2-13 provides typical operational transient parameters.

Table 2-13
Reactor Trip C – Operational Parameters

Maximum Power	100%
Minimum Power	0%
40 Year Cycles	10

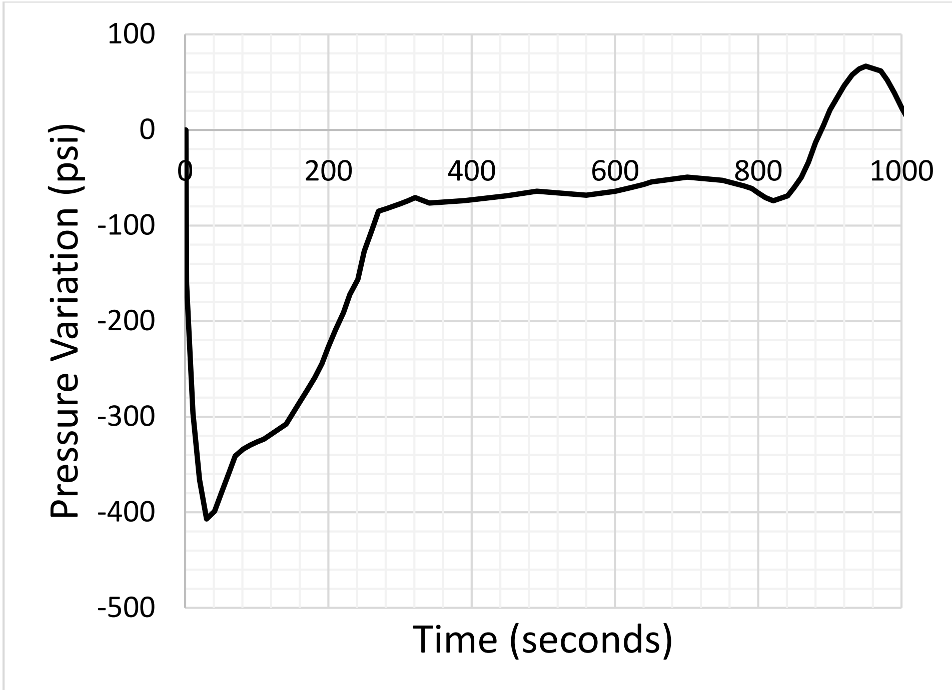

Figure 2-15
Reactor Trip C – Reactor Coolant Pressure

(Note: Power drops from 100% to 0% immediately following reactor trip signal at Time=0.)

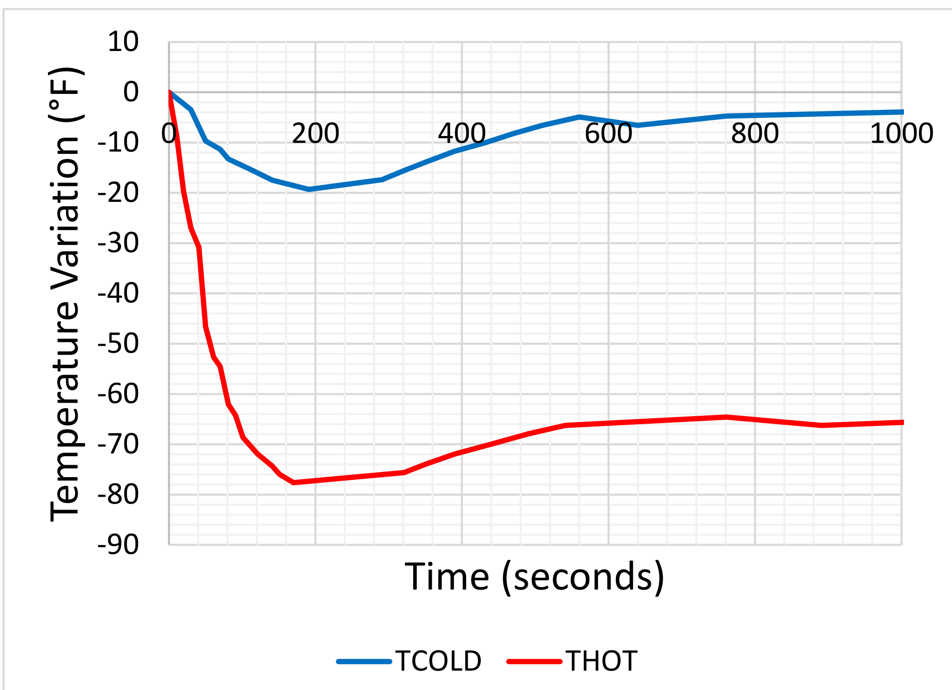

Figure 2-16
Reactor Trip C – Reactor Coolant Temperatures

2.3.2 Inadvertent RCS Depressurization

The Inadvertent RCS Depressurization event accounts for those transients where a large and rapid pressure drop occurs. This pressure drop can be caused by actuation of a pressurizer safety valve, or inadvertent opening of a pressurizer power operated relief valve (PORV). The rapid pressure drop can trip the reactor and initiate the safety injection system to inject coolant into the RCS.

Twenty cycles are typically postulated for design. In practice, actual transients such as this are rare to nonexistent, and no known data is available to illustrate the transient.

2.3.3 Inadvertent Startup of Inactive Loop

This event occurs any time the plant trips due to the unplanned startup of an inactive RCP. This situation can only happen if the plant is initially operating at reduced power with a loop out of service. Because no U.S. plants are believed to be licensed to operate in this manner, no plant data for this event is available. Ten cycles are typically postulated for a 40 year design, but no actual occurrences are expected.

2.3.4 Control Rod Drop

The event occurs when control rods are dropped (i.e. fully inserted) due to the failure of a component, causing a subsequent reactor trip.

Eighty events are typically postulated for the 40 year design life. However, in practice the event is rare, and no plant data is available to characterize the behavior. The reactor trip transient is a reasonable approximation.

2.3.5 Inadvertent Safety Injection Actuation

This event occurs following a spurious safety injection signal. The reactor is tripped, followed by high head safety injection delivered by the charging pump.

Sixty cycles are typically postulated for the 40 year design life, however actual occurrences are rare. The Reactor Trip, Case C event discussed above is a reasonable approximation of the transient characteristics if any are to be considered.

2.3.6 Excessive Feedwater Flow

This event represents an unintended increase in feedwater flow, caused for example by the failure of a feedwater control valve, leading to a reactor trip and safety injection.

Thirty events are typically postulated for the 40-year design life, however in pactice the event is rare. The Reactor Trip, Case C event discussed above is a reasonable approximation of the transient characteristics if any are to be considered.

Section 3: Combustion Engineering (C-E) Designed PWR Transients

Transients affecting reactor pressure vessels in C-E plants are listed in the final safety analysis report (FSAR) and in the surveillance procedures for each such U.S. plant. The Reference [2] license renewal application (LRA), publically available on the U.S. NRC website, provides an example listing of those transients, along with numbers of design cycles. The Reference [3] LRA provides some example projected numbers of cycles at 60 years, using a conservative projection method.

Plant data is available to characterize the transients including the Reactor Coolant System (RCS) hot and cold leg temperatures (indicated as THOT and TCOLD in the sections that follow), the primary system pressure, and the reactor power level in percent. Only those transients related to the reactor vessel are included here. Transient names and typical design cycles related to the reactor pressure vessel are listed in Table 3-1.

Table 3-1
C-E Transients Affecting RPV

Transient Description	FSAR Design Cycles
Normal Condition Transients	
Plant Heatup, 100°F/hr	500
Plant Cooldown, 100°F/hr	500
Plant Loading, 5%/min	15,000
Plant Unloading, 5%/min	15,000
10% Step Load Increase	2,000
10% Step Load Decrease	2,000
Upset Condition Transients	
Loss of Reactor Coolant Flow	40
Loss of power	40
Turbine Trip without Immediate Reactor Trip	40
Reactor Trip from Full Power	400
Inadvertent RCS Depressurization	20
Reactor Trip (400), Loss of Reactor Coolant Flow (40) or Turbine Trip (40) (combined transient)	480

3.1 Normal Condition Transients

3.1.1 Plant Heatup, 100°F/hr

Heatup operations of the RCS are generally limited to a maximum rate of temperature change of 100˚F/hr due to material ductility considerations. It is generally conservative to represent the transient as a constant temperature increase from cold shutdown temperature to the no load temperature.

Plants are designed to accommodate 500 of these events in a 40 year life. Actual accumulation rates generally produce estimates of around 60 to 200 cycles at 60 years of operation, with refueling outages now typically take place every 18 months.

Temperature and pressure time histories of a typical transient are shown in Figure 3-1. Table 3-2 provides typical operational transient parameters for the example provided. Power production occurs following the event during the unit loading events discussed below.

Table 3-2
C-E Plant Heatup – Typical Operational Parameters

Minimum Power	0	%
Maximum Power	0	%
Minimum THOT	185.0	°F
Maximum THOT	564.9	°F
Minimum TCOLD	187.8	°F
Maximum TCOLD	565.7	°F
40 Year Cycles	500	

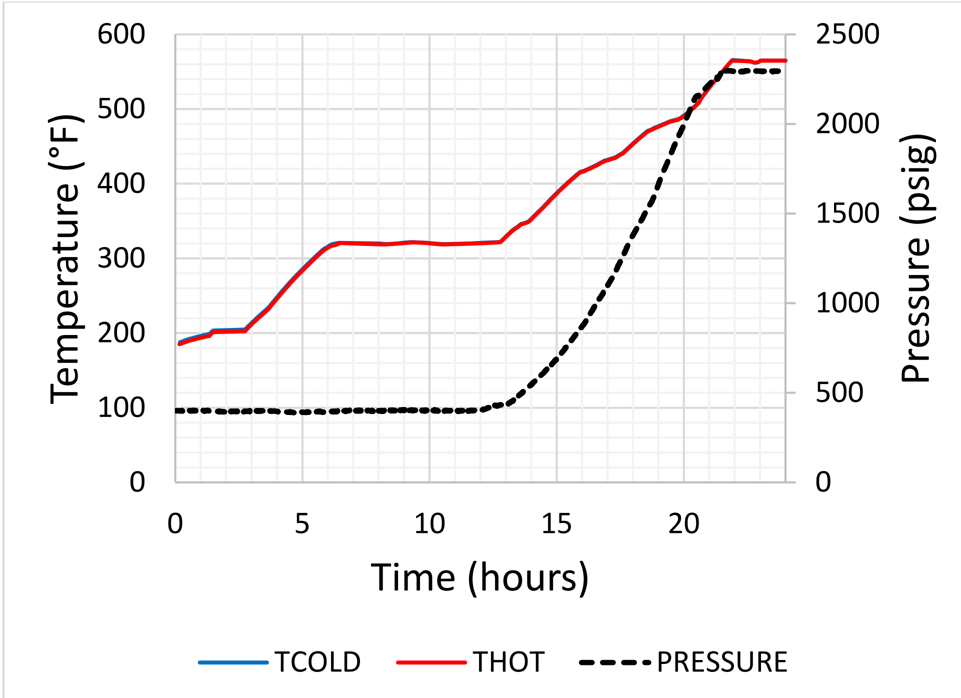

Figure 3-1
C-E Plant Heatup – RCS Temperatures and Pressure

3.1.2 Plant Cooldown, 100°F/hr

Cooldown operations of the Reactor Coolant System are also generally limited to a maximum rate of temperature change of 100°F/hr due to material ductility considerations. For analysis purposes it is generally conservative to represent the transient as a constant temperature decrease from the no load temperature to cold shutdown.

Plants are designed to accommodate 500 of these events in a 40 year life. Actual accumulation rates generally produce estimates of around 60 to 200 cycles at 60 years of operation, with refueling outages now typically taking place every 18 months.

Temperature and pressure time histories of a typical transient are shown in Figure 3-2. Table 3-3 provides typical operational transient parameters for the example provided.

Table 3-3
C-E Plant Cooldown – Typical Operational Parameters

Minimum Power	0	%
Maximum Power	0	%
Minimum THOT	169.2	°F
Maximum THOT	565.4	°F
Minimum TCOLD	170.8	°F
Maximum TCOLD	565.2	°F
40 Year Cycles	500	

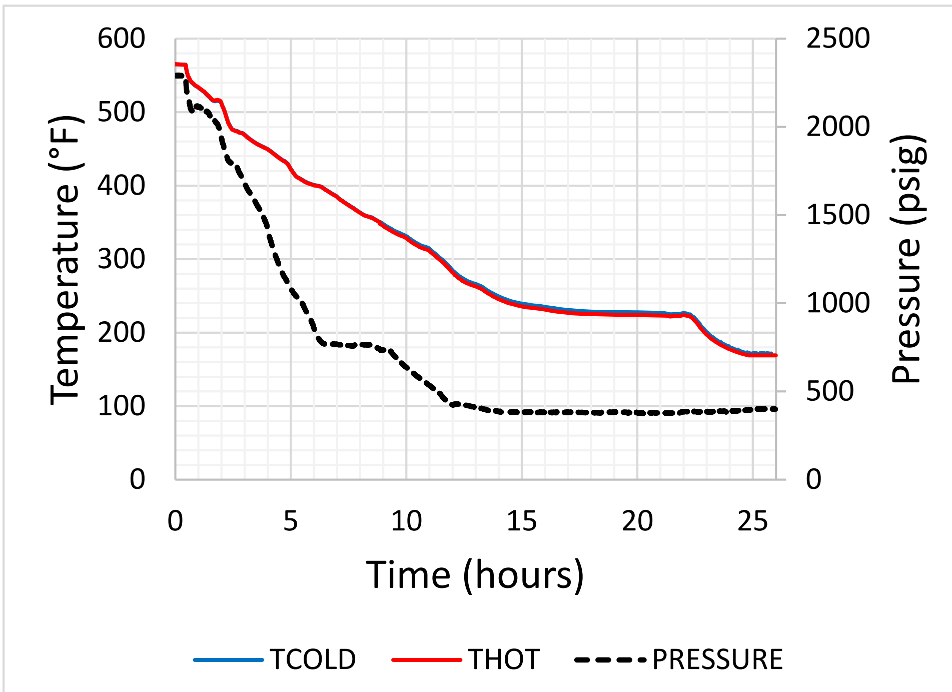

Figure 3-2
C-E Plant Cooldown – RCS Temperatures and Pressure

3.1.3 Plant Loading, 5%/min

This event is counted whenever there is a gradual increase in reactor power (average rate ≤ 5% per minute) while reactor power is above 15%. Actual rates typically occur much more slowly, as shown below.

Plants are designed to accommodate 15,000 of these events. However, because nuclear plants in the U.S. do not typically load follow, the actual number is far fewer (e.g. << 1000).

Power and temperature time histories of a typical transient are shown in Figure 3-3. Table 3-4 provides typical operational transient parameters for the example provided.

Table 3-4
C-E Plant Loading, 5%/min – Typical Operational Parameters

Minimum Power	18.6	%
Maximum Power	88.8	%
Minimum THOT	571.4	°F
Maximum THOT	608.5	°F
Minimum TCOLD	556.4	°F
Maximum TCOLD	561.7	°F
40 Year Cycles	15,000	

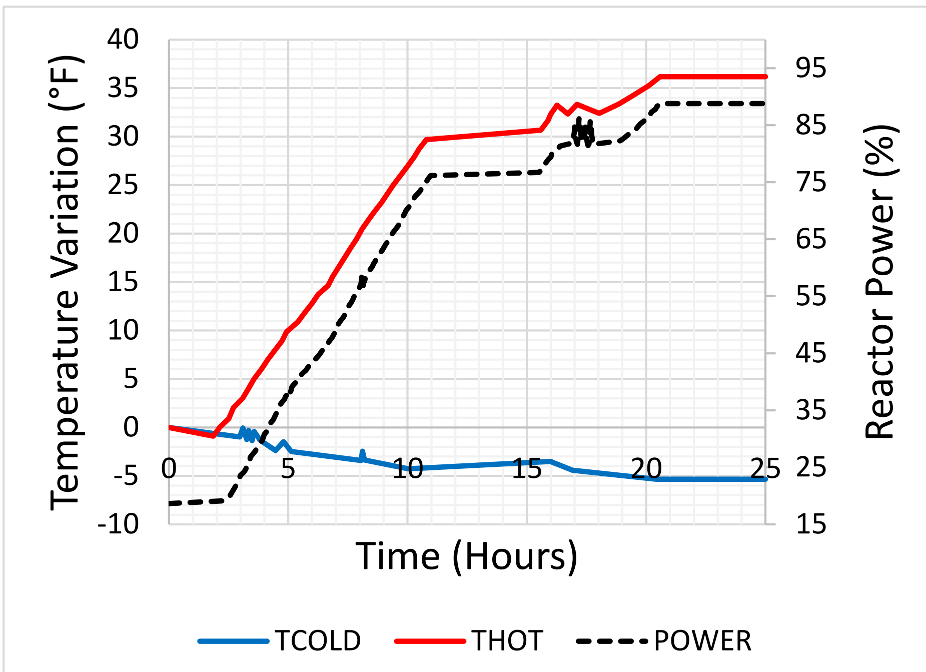

Figure 3-3
C-E Plant Loading, 5%/min – RCS Temperatures and Power

3.1.4 Plant Unloading, 5%/min

This event is counted whenever there is a gradual decrease in reactor power (average rate ≤ 5% per minute) while reactor power is above 15%. Actual rates typically occur much more slowly, as shown below.

The plant is designed to accommodate 15,000 of these events. However, because nuclear plants in the U.S. do not typically load follow, the actual number projected to 60 years of operation is far fewer (e.g. << 1000).

Power and temperature time histories of a typical transient are shown in Figure 3-4. Table 3-5 provides typical operational transient parameters for the example provided. For the most conservative calculation of stresses, it can be assumed that the temperature excursions below occur much quicker, coincident with a maximum 5% power decrease per minute from 100 to 15% power.

Table 3-5
C-E Plant Unloading, 5%/min – Typical Operational Parameters

Minimum Power	26.2	%
Maximum Power	90.7	%
Minimum THOT	562.6	°F
Maximum THOT	610.6	°F
Minimum TCOLD	556.8	°F
Maximum TCOLD	565.3	°F
40 Year Cycles	15,000	

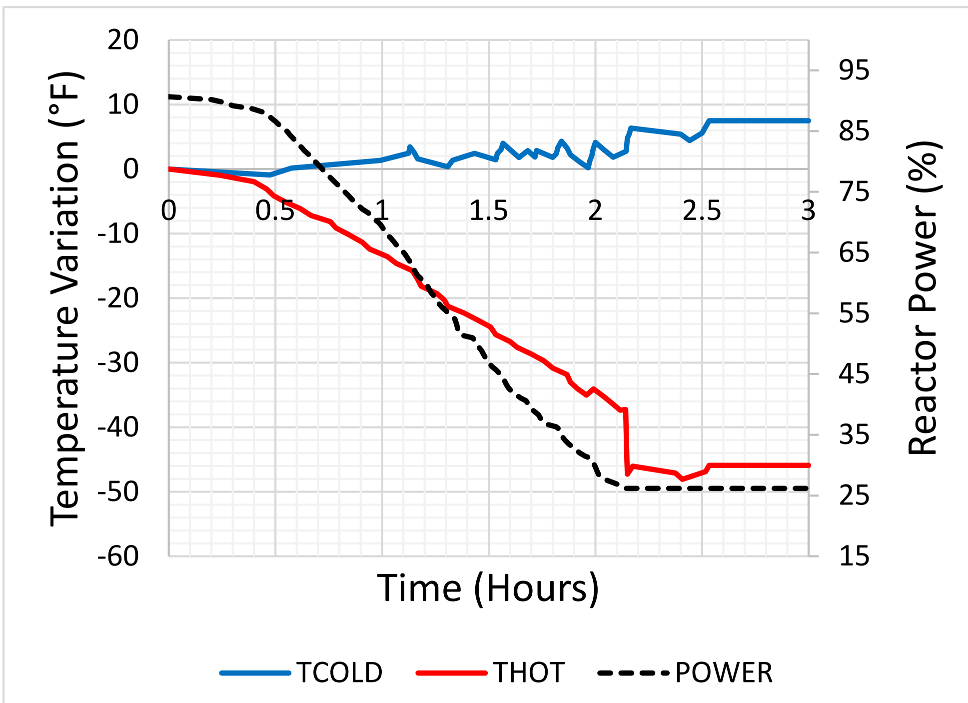

Figure 3-4
C-E Plant Unloading, 5%/min – RCS Temperatures and Power

3.1.5 10% Step Increase and Decrease

This event considers step power changes of 10% of full load, increasing in the 15% to 100% of full load range and decreasing in the 100% to 25% of full load range

2000 cycles each for the 40 year design life is based on normal operation involving one cycle per week for 50 weeks of the year. However, actual events are rare. Monitoring systems may conservatively count these events when the power instrument reading artificially fluctuates (while temperatures stay the same) or make conservative assumptions about rates of accumulation in the absence of data. Postulated events may be conservatively characterized similar to the Westinghouse Large Step Decrease with Steam Dump in Section 2.1.6.

3.2 Upset Condition (Occasional) Transients

3.2.1 Reactor Trip, Loss of Flow or Load

This event envelopes the Reactor Trip, Loss of Flow and Loss of Load events, and typically consists of 480 cycles total.

400 of the cycles are assumed to come from Reactor Trips from full load, based on approximately one reactor trip per month for the life of the plant. 40 of the cycles are assumed to be caused by Turbine Trip (followed by a Reactor Trip), based on one Reactor Trip per year for the life of the plant. The remaining 40 are assumed to come from Loss of Flow when at 100% power, based on one Reactor Trip per year for the life of the plant, resulting from failure of electrical supply to the reactor coolant pumps. Actual cycles projected to 60 years are expected to be less than half of the values assumed for design.

Pressure and temperature time histories of a typical transient are shown in Figure 3-5 and Figure 3-6. Table 3-6 provides typical operational transient parameters.

Table 3-6
C-E Reactor Trip, Loss of Flow or Load – Operational Parameters

Maximum Power	100%
Minimum Power	0%
40 Year Cycles	480

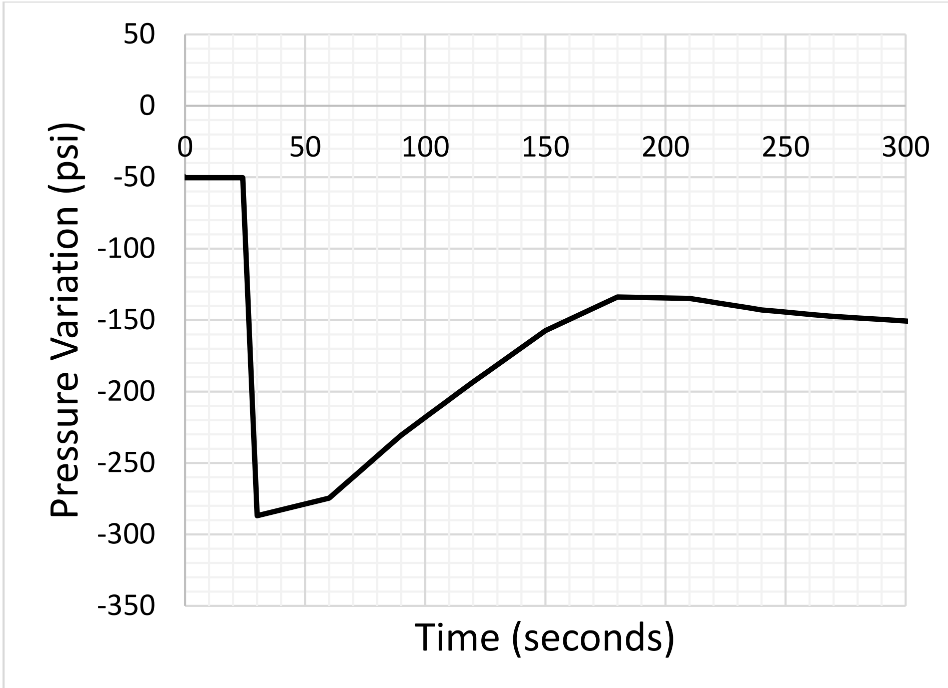

Figure 3-5
C-E Reactor Trip – Reactor Coolant Pressure

(Note: Power drops from 100% to 0% immediately following reactor trip signal at Time=0. Pressure returns to original value consistent with normal heatup)

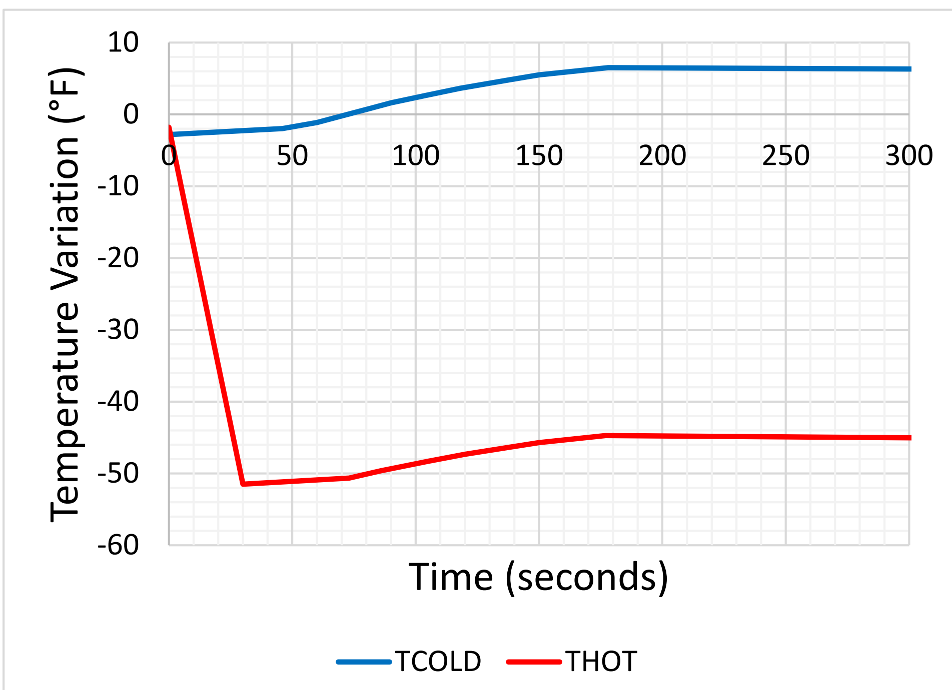

Figure 3-6
C-E Reactor Trip – Reactor Coolant Temperatures

Section 4: References

1. License Renewal Application, Callaway Plant Unit 1, Table 4.3-2, "Transient Accumulations and Projections." http://www.nrc.gov/reactors/operating/licensing/renewal/applications/callaway.html

2. License Renewal Application, St. Lucie Unit 2, Table 3.9-2, "Transients Used in Design and Fatigue Analysis." http://www.nrc.gov/reactors/operating/licensing/renewal/applications/st-lucie.html

3. License Renewal Application, Palo Verde Generating Station Unit 1, Unit 2, and Unit 3, Table 4.3-3, "APS Fatigue Cycle Count Verification (Composite Worst-Case Unit), and Projections." http://www.nrc.gov/reactors/operating/licensing/renewal/applications/palo-verde.html

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