
2.8 REACTOR SYSTEMS

2.8.1 Fuel System Design

2.8.1.1 Regulatory Evaluation

Final Safety Analysis Report (FSAR) Section 4.2 describes the fuel system design and licensing basis. The Comanche Peak Nuclear Power Plant (CPNPP) uses the Westinghouse 17x17 VANTAGE+ fuel design. No changes have been made to the fuel design currently being used as a result of the stretch power uprate (SPU).

The fuel system consists of an array of fuel rods, burnable poison rods, spacer grids and springs, end plates, and reactivity control rods. The fuel system has been reviewed to ensure that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs).
- Fuel system damage is never so severe as to prevent control rod insertion when it is required.
- The number of fuel rod failures is not underestimated for postulated accidents.
- Coolability is always maintained.

The review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system under normal operation, AOOs, and postulated accidents.

The acceptance criteria are based on:

- 10 CFR Part 50.46, insofar as it established standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance.
- General Design Criterion (GDC)-10, insofar as it requires that the reactor core be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs.
- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained.

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- GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any loss-of-coolant accident (LOCA).

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the CPNPP design relative to:

- 10 CFR Part 50.46, Acceptance Criteria for ECCS for Light Water Nuclear Power Reactors, is described in FSAR Section 15.6.5.
- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control and protection systems are designed with adequate margins to:

1. Ensure that fuel damage is not expected during normal core operation and operational transients (Condition I) or any transient conditions arising from occurrences of moderate frequency (Condition II).
2. Ensure return of the reactor to a safe shutdown state following infrequent incident (Condition III) events with only a small fraction of fuel rods damaged, although sufficient fuel damage might occur to preclude immediate resumption of operation.
3. Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components. FSAR Chapter 7 provides the details of the control and protections systems instrumentation design and logic. This information supports the FSAR Chapter 15 accident analysis, which shows that acceptable fuel design limits are not exceeded for Conditions I and II occurrences.

- GDC-27, Combined Reactivity Control Systems Capability, is described in FSAR Section 3.1.3.8.

CPNPP is provided with a means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. FSAR Chapters 4 and 9 discuss these means in detail. Combined use of the rod cluster control system and the chemical shim control system permit the necessary shutdown margin to be

maintained during long-term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full-out upon trip for this determination. FSAR Chapter 15 describes accident assumptions in detail.

- GDC-35, Emergency Core Cooling, is addressed in FSAR Section 3.1.4.6.

An ECCS is provided to cope with any LOCA in the plant design basis. Abundant cooling water is available in an emergency to transfer heat from the core at a rate such that the core is maintained in a coolable geometry and that the cladding-metal reaction is limited to a negligible amount. Adequate design provisions are made to assure performance of the required safety functions even with a single failure.

FSAR Section 6.3 includes details of the capability of the systems. FSAR Chapter 15 includes an evaluation of the adequacy of the system functions. Performance evaluations are conducted in accordance with 10 CFR Part 50.46 and Appendix K to 10 CFR Part 50.

FSAR Section 4.2 describes fuel system design. It addresses mechanical design of the reactor core components and their physical arrangement.

2.8.1.2 Technical Evaluation

2.8.1.2.1 Introduction

FSAR Section 4.2 describes the fuel system design and licensing basis. CPNPP uses the Westinghouse VANTAGE+ fuel design. No changes have been made to the design currently being used as a result of the SPU.

Fuel rod performance for the fuel is shown to satisfy the fuel rod design bases on a region-by-region basis. The design bases for Westinghouse VANTAGE+ fuel is discussed in Reference 1. The evaluation of fuel rod performance at SPU conditions is based on the same methods and models (PAD 4.0) used in the current licensing basis. Compliance with the GDC-10 SAFDL criteria for reload cycles is confirmed via the approved reload methodology of WCAP-9272 (Reference 2).

2.8.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Mechanical Performance

The effects of the SPU on the fuel mechanical design are limited to induced changes in the core flow rates and operating temperatures. The impacts of these changes on the fuel have been analyzed. The fuel design analyses that could be impacted are the fuel assembly lift forces and hold-down force margin. Analyses have been performed to demonstrate that the fuel assembly lift force margin requirement is met for the SPU without any modifications to the current fuel assembly design.

Seismic/LOCA

The effects of the SPU on the seismic/LOCA performance of the fuel mechanical design have been analyzed to confirm that all acceptance criteria and regulatory requirements are met. The criteria for the seismic loading design are that fragmentation of the fuel rod must not occur as a result of the seismic loads and the control rod insertability must be maintained. In addition, coolable geometry of the core must be maintained.

Fuel Rod Performance

The fuel rod design analysis is performed on a cycle-specific basis. In support of the SPU, a fuel rod design analysis has been performed on a representative uprated core design that modeled three cycles at uprated conditions. Both this reference analysis and the cycle-specific analysis consider compliance for all fuel designs in the core. The fuel rod design evaluation for each region incorporates all appropriate design features of the region (such as plenum length, or presence of annular pellets in the axial blanket region) and considers the rod powers or fuel duty associated with the fuel region. Analysis of integral fuel burnable absorber (IFBA) rods includes the geometry changes necessary to model the presence of the integral burnable absorber (ZrB_2 coated pellets), and conservatively models the gas release from the ZrB_2 coating.

Fuel rod design evaluations for the fuel were performed using NRC-approved models (References 1 and 3) and NRC-approved design criteria and methods (References 4 and 5) to demonstrate that all fuel rod design criteria are satisfied. The fuel rod design criteria given below are verified by evaluating the predicted performance of the limiting fuel rod, defined as the rod that gives the minimum margin to the design limit.

The NRC-approved PAD 4.0 code, which incorporates models (References 1 and 3) for in-reactor behavior, is used to calculate the fuel rod performance over its irradiation history. PAD is the principal design tool for evaluating fuel rod performance. PAD iteratively calculates the interrelated effects of temperature, pressure, clad elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power. PAD 4.0 is a best-estimate fuel rod performance model, and in most cases the design criterion evaluations are based on a best-estimate plus uncertainties approach. A statistical convolution of individual uncertainties due to design model uncertainties, fabrication uncertainties, and dimensional uncertainties is used. An evaluation of the NRC-approved oxidation criterion (Reference 6) for structural component oxidation was also performed.

The fuel rod design criteria which have been analyzed are:

- Rod internal pressure

The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward cladding creep during steady-state operation, and extensive departure from nucleate boiling ratio (DNB) propagation to occur.

- Cladding stress and strain

The design limit for cladding stress is that the volume average effective stress considering interference due to uniform cylindrical pellet-to-cladding contact, caused by pellet thermal expansion, pellet swelling, and uniform cladding creep, and pressure differences is less than the 0.2-percent offset yield stress, with due consideration to temperature and irradiation effects under Condition I and II modes of operation. While the cladding has some capability for accommodating plastic strain, the yield stress has been established as a conservative design limit. The design limit for fuel rod cladding strain during steady-state operation is that the total plastic tensile creep strain due to uniform cladding creep and uniform cylindrical fuel pellet expansion associated due to fuel swelling and thermal expansion is less than 1 percent from the unirradiated condition. The design limit for fuel rod cladding strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion during the transient is less than 1 percent from the pre-transient value. These limits are consistent with proven practice.

- Oxidation and hydriding

The design criteria related to cladding corrosion require that the ZIRLO™ clad metal-oxide interface temperature is maintained below specified limits to prevent a condition of accelerated oxidation, which would lead to cladding failure.

The best-estimate hydrogen pickup level in ZIRLO cladding is less than or equal to the specified limit at end of life (EOL) to preclude a loss of ductility due to hydrogen embrittlement by the formation of zirconium hydride platelets.

The ZIRLO structural component stresses will be consistent with the American Society of Mechanical Engineers (ASME) Code Section III requirements after accounting for thinning due to corrosion.

- Fuel temperature

For Condition I and II events, the fuel and reactor protection systems are designed to ensure that a calculated centerline fuel temperature does not exceed the fuel melting temperature criterion. The melting temperature is taken to be 5,080°F (unirradiated) and to decrease by 58°F per 10,000 MWD/MTU of fuel burnup. The intent of this criterion is to avoid a condition of gross fuel melting that can result in severe duty on the cladding. The concern here is based on the large volume increase associated with the phase change in the fuel, and the potential for loss-of-cladding integrity as a result of molten fuel/cladding interaction.

- Cladding fatigue

The design limit for cladding strain fatigue is that the fatigue usage factor is less than 1.0 to prevent reaching the material fatigue limit. That is, for a given strain range, the

number of strain fatigue cycles is less than that required for failure, considering a factor of safety of 2 on the stress amplitude and a factor of safety of 20 on the number of cycles. The concern of this criterion is the accumulated effect of short-term cyclic, cladding stress and strain, that results from daily load follow operation.

- **Cladding flattening**

The cladding flattening criterion prevents fuel rod failures due to long-term creep collapse of the fuel rod cladding into axial gaps formed within the fuel stack. Current fuel rod designs employing high density fuel with improved in-pile stability and helium backfill pressure provide adequate assurance that axial gaps large enough to allow cladding flattening will not form within the fuel stack.

- **Fuel rod axial growth**

The criterion is that fuel rods are designed with adequate clearance between the fuel rod ends and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly to preclude interference of these members. This criterion ensures that there is sufficient axial space to accommodate the maximum expected fuel rod growth without degradation of the assembly function.

2.8.1.2.3 Description of Analyses and Evaluations

Mechanical Performance

With respect to the mechanical performance of the fuel, the impacts on the fuel of the core flow rates and operating temperatures changes have been analyzed. The fuel design analyses that could be impacted are the fuel assembly lift forces and hold-down force margin. Analyses have been performed to demonstrate that the fuel assembly lift force design requirement is met for the SPU without any modifications to the current fuel assembly design. The hold-down force calculation conservatively assumed high burnup fuel assembly growth and hold-down spring relaxation due to irradiation effects. The analysis accounted for the opposing forces that act on the fuel assemblies due to fuel assembly weight, buoyancy, spring force, and lift force.

Seismic/LOCA

The results of the combined LOCA and seismic analysis were obtained using the time-history numerical integration technique. The maximum grid impact forces obtained from both transients were combined using the square-root-sum-of-the-squares (SRSS) method. The maximum loads were compared with the allowable grid crush strength. In the grid load analysis, the time-history motions of the barrel at the upper core plate elevation and the upper and lower core plates were applied simultaneously to the reactor core model. The time histories representing the seismic motion and the pipe rupture transients were obtained from the time-history analyses of the reactor vessel and internals finite element model.

Fuel Rod Performance

All the design criteria were analyzed for the representative uprated core design for all fuel regions in this core. These analyses used the SPU specific operating conditions and fuel duty (rod radial powers and axial power shapes).

Fuel rod design evaluations for the fuel were performed using NRC-approved models (References 1 and 3) and NRC-approved design criteria methods (References 4 and 5) to demonstrate that all fuel rod design criteria are satisfied. The fuel rod design criteria listed in Licensing Report (LR) subsection 2.8.1.2.2 are verified by evaluating the predicted performance of the limiting fuel rod, defined as the rod that gives the minimum margin to the design limit.

2.8.1.2.4 Fuel System Design Results

Mechanical Performance

With respect to the mechanical design of the fuel, the analyses performed confirm that the fuel design is structurally and mechanically acceptable for the SPU. Use of re-inserted previously irradiated assemblies is also acceptable for the SPU.

Seismic/LOCA

The effects of the SPU on the fuel mechanical design are limited to changes in the uprate parameters (LR Section 1.1). The seismic and LOCA core plate motions and the fuel assembly nozzle impact loads used in the current analysis of record remain applicable for the SPU. The acceptance criteria for the mechanical evaluation is that the fuel remain structurally acceptable.

A seismic and LOCA evaluation for CPNPP was performed at the uprated conditions. The results of the evaluation show that the maximum combined impact forces on the mid grid and intermediate flow mixer (IFM) grids are below the grid crush limits. The results indicated adequate margin for both fuel rod and thimble tube exists. Fragmentation of the fuel rods and thimble tubes will not occur.

The results of the combined seismic and LOCA analyses indicate that the maximum impact forces are less than the respective allowable grid strengths. The allowable grid strengths are established at the 95-percent confidence level on the true mean from the distribution of experimentally determined grid crush data at operating temperature. Based on the results of the combined seismic and LOCA loads, the fuel design is structurally acceptable for the SPU and the core coolable geometry requirements are met. Re-insertion of previously irradiated assemblies with or without IFM assemblies is also acceptable under seismic and LOCA loads for the SPU.

Fuel Rod Design

The results of these analyses have shown that all design criteria have been met for this representative uprated core. These results provide reasonable assurance that all design criteria will be met when cycle-specific analyses are performed for CPNPP actual uprated cycles in the future.

Fuel rod design analyses are performed on a cycle-specific basis considering the plant conditions of the specific cycle as well as the fuel duty of each of the fuel regions in the core during that specific cycle. These cycle-specific analyses are performed using NRC-approved design criteria and methods to demonstrate that all design criteria are satisfied. The results of these analyses are reported in cycle-specific documentation as part of the normal reload design process.

2.8.1.3 Conclusions

Luminant Power has reviewed the analyses related to the effects of the proposed SPU on the fuel system design of the fuel assemblies, control systems, and reactor core. Luminant Power concludes that the analyses have adequately accounted for the effects of the proposed SPU on the fuel system and demonstrated that: (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, Luminant Power concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC-10, GDC-27, and GDC-35 following implementation of the proposed SPU.

2.8.1.4 References

1. WCAP-12610, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.
2. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
3. WCAP-15063, Rev. 1 with Errata, WCAP-15064, Rev. 1, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000.
4. WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel," December 1985.
5. WCAP-13589, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," March 1995.
6. WCAP-12488, Addendum 1-A, Revision 1, "Addendum 1 to WCAP-12488, Revision to Design Criteria," January 2002.

2.8.2 Nuclear Design

2.8.2.1 Regulatory Evaluation

Luminant Power reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary (RCPB) or impair the capability to cool the core. The Luminant Power review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation.

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor core be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).
- GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.
- GDC-12, insofar as it requires that the reactor core be designed to ensure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed.
- GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges.
- GDC-20, insofar as it requires that the protection system be designed to automatically initiate the reactivity control systems to ensure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important-to-safety under accident conditions.
- GDC-25, insofar as it requires that the protection system be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.
- GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes.

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- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained.
 - GDC-28, insofar as it requires that the reactivity control systems be designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to conformance to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control and protection systems are designed with adequate margins to do the following:

- To preclude significant fuel damage during normal core operation and operational transients (Condition I) or transient conditions arising from occurrences of moderate frequency (Condition II).
- To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude immediate resumption of operation without considerable outage time.
- To ensure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components including the fuel, reactor vessel internals and reactivity control systems. Details of the control and protections systems instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the accident analysis of FSAR Chapter 15, which shows that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

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- GDC-11, Reactor Inherent Protection, is described in FSAR Section 3.1.2.2.

When the reactor is critical, prompt compensatory reactivity feedback effects are ensured by the negative fuel temperature effect (Doppler effect) and by the operational limit on the moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is ensured by the inherent design using low-enrichment fuel. The limit on moderator temperature coefficient of reactivity is ensured by administratively controlling the dissolved absorber concentration or by burnable poison.

These reactivity coefficients are discussed in FSAR Section 4.3.2.3.

- GDC-12, Suppression of Reactor Power Oscillations, is described in FSAR Section 3.1.2.3.

Power oscillations of the fundamental mode are inherently stable by the negative power coefficient of reactivity. In addition, oscillations, due to xenon spatial effects:

- In the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative power coefficient of reactivity.
- In the axial first overtone mode may occur. Assurance that the fuel design limits are not exceeded by xenon axial oscillations is provided by reactor trip functions using the measured axial power imbalance as an input.
- In the axial modes higher than the first overtones are heavily damped because of the inherent design and because of the negative Doppler coefficient of reactivity.

Xenon stability control is discussed in FSAR Section 4.3.2.7.

- GDC-13, Instrumentation and Control, is described in FSAR Section 3.1.2.4.

To ensure adequate safety, instrumentation and control systems are provided to monitor and control significant variables over their anticipated range for all conditions in the reactor core, RCS, steam and power conversion system, radioactive waste systems, engineered safety features (ESF) systems, and the Containment Building. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the Control Room in proximity to the controls that maintain the indicated parameters in the proper range.

The quantity and types of processing instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. These systems are described in FSAR Chapters 6, 7, 8, 9, 10, 11, and 12.

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- GDC-20, Protection System Functions, is described in FSAR Section 3.1.3.1.

A fully automatic protection system, with appropriate redundant channels, is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of Institute of Electrical and Electronics Engineers (IEEE) 279-1971 and IEEE 379-1972. The reactor protection system automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the control rod drive mechanisms (CRDMs) of all full-length rod cluster control assemblies (RCCAs). This causes the rods to insert by the force of gravity, rapidly reducing the reactor power output. The response and adequacy of the protection system has been verified by analysis of anticipated transients.

Refer to FSAR Chapter 7, Instrumentation and Controls, for additional information regarding actuating devices to the protection system.

The ESF actuation system automatically initiates emergency core cooling and other safeguards functions by sensing accident conditions using redundant analog channels measuring diverse variables. In addition, manual action of safeguards can be performed where ample time is available for operator action. In either case, the ESF actuation system automatically trips the reactor on manual or automatic safety injection signal generation.

- GDC-25, Protection System Requirements for Reactivity Control Malfunctions, is described in FSAR Section 3.1.3.6.

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by full-length rod insertion is completely independent of the normal control function since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. In the postulated accidental withdrawal (assumed to be initiated by a control malfunction), flux, temperature, pressure, level and flow signals are independently generated. Any of these signals (trip demands) can operate the breakers to trip the reactor.

Analyses of the effects of possible malfunctions are discussed in FSAR Chapter 15. These analyses show that for postulated dilution during refueling, startup or manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution, and to initiate boration before the shutdown margin is lost. The analyses also show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

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- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control systems are provided. These are RCCAs and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation the shutdown rod banks are fully withdrawn. The fuel-length rod control system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the full-length control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and AOOs, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boration system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCA are presented in FSAR Chapter 4. The operation is discussed in FSAR Chapter 7. The means of controlling the boric acid concentration are described in FSAR Chapter 9. Performance analyses under accident conditions are included in FSAR Chapter 15.

- GDC-27, Combined Reactivity Control Systems Capability, is described in FSAR Section 3.1.3.8.

The facility is provided with means of making the core subcritical and maintaining it at that level under any anticipated conditions and with an appropriate margin for contingencies. These means are discussed in detail in FSAR Chapters 4 and 9. Combined use of the RCCAs and the chemical shim permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. Upon trip for this determination, the single highest worth control cluster is assumed to be stuck full-out upon trip.

- GDC-28, Reactivity Limits, is described in FSAR Section 3.1.3.9.

The maximum reactivity worth of control rods and the maximum rate of reactivity insertion employing control rods are limited to values that prevent rupture of the RCS boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The maximum positive reactivity insertion rates for the withdrawal of RCCA and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCA and of the chemical and volume control system (CVCS). Technical Specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated

accidents are no more severe than those presented in the analyses of FSAR Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in FSAR Section 4.3. The capability of the CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in FSAR Chapter 15.

Assurance of core cooling capability following Condition IV accidents, such as rod ejections, steam line break, and similar accidents, is given by keeping the RCPB stresses within faulted condition limits as specified by applicable ASME codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of necessary safety features.

FSAR Section 4.3 describes the design bases and functional requirements used in the nuclear design of the fuel and reactivity control system.

2.8.2.2 Technical Evaluation

2.8.2.2.1 Introduction

The licensing basis for the reload core nuclear design is defined in FSAR Section 4.3. The purpose of the core analysis is to develop key safety parameters applicable to the stretch power uprate (SPU). This allows the majority of any safety analysis re-evaluations/re-analyses to be completed prior to the cycle specific design analysis. The effects of the SPU conditions on the nuclear design bases and methodologies for CPNPP are evaluated in this section.

2.8.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The specific values of core safety parameters, (such as power distributions, peaking factors, rod worths, and reactivity parameters) are loading pattern dependent. The variations in loading pattern dependent safety parameters are expected to be similar to the cycle-to-cycle variations for typical fuel reloads.

The reload design methodology includes the evaluation of the reload core key safety parameters which comprise the nuclear design-dependent input to the FSAR safety evaluation for each reload cycle (Reference 1). These key safety parameters are evaluated for each CPNPP reload cycle. If one or more of the parameters fall outside the bounds assumed in the reference safety analysis, the affected transients are re-evaluated/re-analyzed using standard methods and the results documented in the reload evaluation for that cycle.

Table 2.8.2-1 provides the key safety parameter ranges used in this evaluation compared to the current limits.

2.8.2.2.3 Description of Analyses and Evaluations

Core loading patterns for three cycles were established using PHOENIX-P and ANC (References 2 and 3) to model the CPNPP Units 1 and 2 SPU. Typical loading patterns were developed based on projected energy requirements of approximately 515 effective full-power days (EFPDs) for CPNPP. These models are not intended to represent limiting loading patterns, but were instead developed with the intent to show that enough margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores. The range of the values in Table 2.8.2-1, developed using these models, were used for comparison to evaluate the continued adequacy of margins between typical safety parameter values and the corresponding limits.

2.8.2.2.4 Results

The key safety parameters evaluated for CPNPP as it transitions to the SPU show little change relative to the current design. The changes in values of the key safety parameters are typical of the normal cycle-to-cycle variations experienced as loading patterns change. The specific reload cycle core design is such that adequate margin will exist for all key nuclear design criteria at the SPU conditions.

2.8.2.3 Conclusions

Luminant Power has reviewed the analyses related to the effect of the proposed SPU on the nuclear design of the fuel assemblies, control systems, and reactor core. Luminant Power concludes that the analyses have adequately accounted for the effects of the proposed SPU on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, Luminant Power concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs-10, -11, -12, -13, -20, -25, -26, -27, and -28. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the nuclear design.

2.8.2.4 References

1. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
2. WCAP-11596, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.
3. WCAP-10965, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.

Table 2.8.2-1
Range of Key Safety Parameters

	Current Design Values	SPU Values
Reactor Core Power (MWt)	3,458	3,612
Vessel Average Coolant Temp. Hot Full Power (HFP) (°F)	589.2	574.2 to 589.2 ⁽¹⁾
Coolant System Pressure (psia)	2,250	2,250
Core Average Linear Heat Rate (kW/ft)	5.52	5.77
Most Positive Moderator Temperature Coefficient (MTC) (pcm/°F)		
Power < 70%	+ 5.0	+ 5.0
Power > 70% ⁽²⁾	+ 0.0	+ 0.0
Most Positive Moderator Density Coefficient (MDC) (ΔK/g/cm ³)	0.38	0.50 0.38 ⁽³⁾
Doppler Temperature Coefficient (pcm/°F)	-2.241 to -0.785	-2.90 to -0.91
Doppler Only Power Coefficient (pcm/%Power)		
Least negative, 120% rated thermal power (RTP) to hot zero power (HZP)	-	-9.55 to -5.35
Most negative, 120% RTP to HZP	-	-19.40 to -11.24
Beta-Effective	0.0044 to 0.0070	0.0044 to 0.0070
Normal Operation F _N ΔH	1.55	1.60
Normal Operation F _Q (Z)	2.42	2.50
Shutdown Margin (%Δρ)	1.30	1.30
Notes:		
1. Constant temperature program assumed during nominal depletion.		
2. Linear ramp from 70% to 100% power.		
3. An MDC of 0.38 will bound the MDC response following a reactor trip with N-1 rods inserted. If the steam line break mass and energy event is sensitive to the MDC before the reactor trip occurs, a value of 0.50 should be used during this portion of the event.		

2.8.3 Thermal and Hydraulic Design

2.8.3.1 Regulatory Evaluation

The Luminant Power review covered the thermal and hydraulic design of the core and the reactor coolant system (RCS) to confirm that the design:

- Has been accomplished using acceptable analytical methods
- Is equivalent to or a justified extrapolation from proven designs
- Provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and anticipated operational occurrences (AOOs)
- Is not susceptible to thermal-hydraulic (T/H) instability

The review of the subject design analyses also covered hydraulic loads on the core and RCS components during normal operation and design basis accident (DBA) conditions, and core T/H stability under normal operation and Condition 2 events.

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor core be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs.
- GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to:

- GDC-10, Reactor Design, is described in Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I) or transient conditions arising from occurrences of moderate frequency (Condition II).
2. To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude immediate resumption of operation without considerable outage time.
3. To ensure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components including the fuel, reactor vessel internals, and reactivity control systems. Details of the control and protections systems instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the accident analysis of FSAR Chapter 15, which shows that acceptable fuel design limits are not exceeded for Conditions I and II occurrences.

- GDC-12, Suppression of Reactor Power Oscillations, is described in FSAR Section 3.1.2.3.

Power oscillations of the fundamental mode are inherently stable by the negative power coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative power coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the axial first overtone mode may occur. Assurance that the fuel design limits are not exceeded by xenon axial oscillations is provided by reactor trip functions using the measured axial power imbalance as an input.

Oscillations, due to xenon spatial effects, in the axial modes higher than the first overtones are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity. FSAR Section 4.3 discusses xenon and samarium stability control.

In addition, FSAR Section 4.4 states the following relative to thermal and hydraulic design:

- There will be at least a 95-percent probability that departure from nucleate boiling ratio (DNBR) will not occur on the limiting fuel rods during normal operation and operational transients and any transient arising from faults of moderate frequency (Conditions I and II events) at a 95-percent confidence level. Further information on CPNPP DNBR design is shown in FSAR Section 4.4.1.1.
- During modes of operation associated with Conditions I and II events, there is at least a 95-percent probability that the peak kW/ft fuel rods will not exceed the UO₂ melting temperature at the 95-percent confidence level. Further information on CPNPP fuel temperature design is shown in FSAR Section 4.4.1.2.
- Most of the thermal flow rate will pass through the fuel regions of the core and be effective for fuel rod cooling. Coolant flow through the thimble tubes, as well as the leakage from the core barrel-baffle region, into the core, are not considered effective for heat removal. Further information on the core flow design is shown in FSAR Section 4.4.1.3
- As indicated in FSAR Section 4.4.1.4, modes of operation associated with Conditions I and II events shall not lead to hydrodynamic instability.
- Other conditions for thermal and hydraulic design are discussed in FSAR Sections 4.4.1.5 and 4.4.2.

2.8.3.2 Technical Evaluation

Introduction

This section describes the T/H analysis supporting the CPNPP stretch power uprate (SPU) containing a full core of 17x17 VANTAGE+ fuel assemblies. The current licensing basis for T/H design for CPNPP includes the prevention of DNB on the limiting fuel rod with a 95-percent probability at a 95-percent confidence level and criteria to ensure fuel cladding integrity, and is documented in Section 4.4 of the FSAR. The SPU analysis is based on this licensing basis analysis incorporating the increased core power. The analysis addresses the departure from nucleate boiling (DNB) performance, including the effects of fuel rod bow and bypass flow.

Also considered in this section are:

- The calculation of fuel temperature/pressure data used in various safety analyses
- Core stored energy

Input Parameters, Assumptions, and Acceptance Criteria

VIPRE-01 is the core T/H subchannel analysis code that was used for the SPU analysis. NRC approval of the Westinghouse VIPRE-01 methodology was issued in the SER attached to Reference 1.

For the purposes of the SPU analysis, bounding fuel-related safety and design parameters have been chosen. These bounding parameters have been used in the safety and design analyses discussed in this section and in other relevant sections of this report.

Table 2.8.3-1 lists the T/H parameters for the current design at 3,458 MWt, as well as for the SPU design at 3,612 MWt. Some of the parameters listed in Table 2.8.3-1 are used in the analysis basis as VIPRE-01 input parameters while others are provided for information. This section identifies those parameters that are used as input parameters to the VIPRE-01 model and also identifies the limiting direction of each parameter, which is shown in Table 2.8.3-2. In addition, the average linear power (kW/ft) is used in the PAD analyses for the fuel temperature and the rod internal pressure calculations. The following parameters from Table 2.8.3-1 are used in the VIPRE-01 model:

- Reactor core heat output (MWt)
- Heat generated in fuel (%)
- Nominal vessel/core inlet temperature (°F)
- $F_{\Delta H}^N$, nuclear enthalpy rise hot-channel factor (radial power distribution)
- Pressurizer/core pressure (psia)
- Thermal design flow (gpm)

The T/H analysis of the SPU in CPNPP is based on the Revised Thermal Design Procedure (RTDP) (Reference 2), the WRB-2 DNB correlation (Reference 3), and the VIPRE-01 code (Reference 1). The W-3 correlation and the Standard Thermal Design Procedure (STDP) are used when any one of the conditions is outside the range of the WRB-2 correlation (that is, pressure, local mass velocity, local quality, heated length, grid spacing, equivalent hydraulic diameter, equivalent heated hydraulic diameter, and distance from last grid to critical heat flux (CHF) site) and RTDP (that is, the statistical variance is exceeded on power, T_{IN} , pressure, flow, bypass, $F_{\Delta H}^N$, $F_{\Delta H,1}^E$, and F_Q^E).

The WRB-2 DNB correlation is based entirely on rod bundle data and takes credit for the significant improvements in DNB performance due to the mixing vane grid effects. NRC acceptance of a 95/95 correlation limit DNBR of 1.17 is documented in Reference 3.

With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor. This factor is used to determine the plant-specific design limit DNBR that satisfies the DNB design criterion. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

The uncertainties included in the overall DNB uncertainty factor are:

- The nuclear enthalpy rise hot channel factor, ($F_{\Delta H}^N$)
- The enthalpy rise engineering hot channel factor, ($F_{\Delta H,1}^E$)
- Uncertainties in the VIPRE-01 and transient codes
- Vessel coolant flow
- Effective core flow fraction
- Core thermal power
- Coolant temperature
- System pressure

Because the uncertainties are incorporated in the DNBR limit, nominal values of the peaking and hot channel factors are used as input to the DNB safety analyses.

Uncertainties in core thermal power, RCS flow, pressure and temperature used for the SPU analyses, are listed in Table 2.8.3-3. The measurement uncertainties were used in determining the DNBR design limits.

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

- There is at least a 95-percent probability that DNB will not occur on the limiting fuel rods during Modes 1 and 2 operational transients, or any condition of moderate frequency at a 95-percent confidence level.
- There is no fuel melting during any anticipated normal operating condition, operational transients, or any conditions of moderate frequency.
- Mode of operation under Condition I and II events will not lead to thermal-hydrodynamic instabilities.

The ratio of the heat flux causing DNB at a particular core location, as predicted by a DNB correlation, to the actual heat flux at the same core location is the DNBR. Analytical assurance that DNB will not occur is provided by showing the calculated DNBR to be higher than the 95/95 limit DNBR for all conditions of normal operation, operational transients, and transient conditions of moderate frequency. The design limit DNBR is calculated by using the RTDP methodology, which includes appropriate margin to DNB for all operating conditions sufficient to assure compliance with the DNBR criteria above.

The safety analysis limit (SAL), which is higher than the Design Limit DNBR, is conservatively utilized in the DNB safety analyses to provide DNBR margin to offset the effect of rod bow and any other DNBR penalties that may occur, and to provide flexibility in design and operation of the plant. To account for various penalties and potential operational issues, the plant-specific margins are retained between the design limit DNBR and the SAL DNBR.

Description of Analyses and Evaluations

For the SPU analysis, the design limit DNBR value is []^{a,c} for typical and thimble cells. After accounting for the plant-specific margin, the SAL DNBR is []^{a,c} (typical/thimble). These SALs are employed in the DNB analyses.

With the SAL DNBR set, the core limits, axial offset limits, and dropped rod limits are generated. Based on these limits, the maximum $F_{\Delta H}^N$ limit that can be supported is 1.60. This limit incorporates all applicable uncertainties, including a measurement uncertainty of []^{a,c} percent (Reference 4), and is adjusted for power level using the following equation:

$$F_{\Delta H}^N = 1.60 \times (1 + 0.3(1-P))$$

where P is the fraction of full power.

Rod bow can occur between mid-grids, reducing the spacing between adjacent fuel rods and reducing the margin to DNB. Rod bow must be accounted for in the DNBR safety analysis of Conditions I and II events. Westinghouse has conducted tests to determine the impact of rod bow on DNB performance, the testing and subsequent analyses were documented in Reference 5.

Currently, the maximum rod bow penalty for the 17x17 VANTAGE+ fuel assembly is []^{a,c} at an assembly average burnup of 24,000 MWD/MTU (References 5 and 6). No additional rod bow penalty is required for burnups greater than 24,000 MWD/MTU since credit is taken for the effect of $F_{\Delta H}^N$ burndown due to the decrease in fissionable isotopes and the buildup of fission products (Reference 7).

Two different bypass flow rates are used in the T/H design analysis. The thermal design bypass flow (TDBF) is the conservatively high core bypass flow used with the thermal design flow (TDF) in power capability analyses that use standard (non-statistical) methods, and is also used to calculate fuel assembly pressure drops. The best-estimate bypass flow (BEBF) is the core bypass flow that would be expected using nominal values for dimensions and operating parameters that affect bypass flow without applying uncertainty factors. The BEBF is used in conjunction with the vessel minimum measured flow (MMF) for power capability analyses using the RTDP (statistical) design procedure. The BEBF is also used to calculate fuel assembly lift forces. For example, the TDBF limit is 5.8 percent and the BEBF limit is []^{a,c} based on thimble plug insertion.

Fuel temperatures and associated rod internal pressures have been generated using the NRC-approved PAD code (Reference 8). The maximum fuel rod average and surface temperatures are needed for the accident analyses. In addition, minimum fuel average and fuel surface temperatures are required by non-LOCA analysis. Fuel centerline temperatures were also generated. These will be used for future verification, during reload design validation, to ensure that fuel melt will not occur.

In addition to the fuel temperatures and pressures, the core stored energy has been determined for use in containment analysis (refer to Section 2.6 of this report). Core stored energy is defined as the amount of energy in the fuel rods in the core above the local coolant temperature. The local core stored energy is normalized to the local linear power level. A value of []^{a,c} full-power seconds has been determined.

Loss of Flow

This section supplements the methodology discussed in subsection 2.8.5.3.1 of this report.

The DNB analysis of the loss-of-flow accident was performed for SPU conditions. Several cases, including partial loss of flow (PLOF), complete loss of flow-undervoltage (CLOF-UV), and CLOF-frequency decay (CLOF-FD) statepoints were analyzed to ensure the limiting scenario was identified. The minimum DNBRs calculated for these cases were greater than the safety analysis DNBR limit, thereby demonstrating compliance to the DNB design criterion for this event.

Locked Rotor

This section supplements the methodology discussed in subsection 2.8.5.3.2 of this report.

The analysis of the locked rotor accident was performed for SPU conditions. The locked rotor accident is classified as an ANS Condition IV event. To estimate the radiation release possible as a consequence of the accident, DNB calculations were performed to quantify the inventory of rods that would experience DNB and be conservatively presumed to fail. For CPNPP, the analysis indicates that there would be []^{a,c}-percent rods-in-DNB due to the locked rotor accident. The radiological consequences analysis conservatively assumed 10 percent of the fuel rods as failed rods and showed that the site dose limits were met.

The locked rotor peak cladding temperature (PCT) was calculated with the VIPRE-01 code using STDP methodology. The following assumptions were used:

- DNB is assumed to occur at the beginning of the transient.
- The Bishop-Sandberg-Tong film boiling heat transfer coefficient was used (Reference 9).
- The fuel-clad gap heat transfer coefficient is assumed to increase to 10,000 Btu/hr-ft² at the beginning of the transient.
- The Baker-Just correlation (Reference 10) was used to predict the heat addition due to the zirconium-water reaction.

The results showed that the PCT limit of 2,700°F was met with a large margin.

RCCA Drop/Misoperation

This section supplements the methodology discussion of Licensing Report (LR) subsection 2.8.5.4.3 for this non-LOCA event.

The NRC-approved Westinghouse analysis methods in Reference 11 were used for analyzing the RCCA drop event. The dropped rod limit lines (DRLLs) defines DNB-based limits on peaking factors as functions of core inlet temperature, core power, and pressure. Based on the DRLL and transient statepoints covering a range of reactivity insertion mechanisms, nuclear design calculations determined pre-drop $F_{\Delta H}$ values corresponding to the post-drop peaking factors at the SAL DNBR. The maximum pre-drop $F_{\Delta H}$ for each reload is specified in the Core Operating Limit Report (COLR). The cycle-specific RCCA drop analysis confirms that all allowed pre-drop $F_{\Delta H}$ values do not violate the COLR limit, and the DNB design basis is met for SPU. In addition, the maximum linear heat rate from the RCCA drop analysis is lower than the fuel centerline melt limit. Therefore, the peak fuel centerline melt temperature criterion is also met for this event.

Uncontrolled Rod Cluster Control Assembly Withdrawal from Subcritical

The analysis for the uncontrolled rod cluster control assembly withdrawal from subcritical (RWFS) event is based on the STDP methodology since the event was initiated from hot zero power (HZP) conditions. The W-3 correlation was used below the first mixing vane grid and the WRB-2 correlation was used above the first mixing vane grid. The minimum DNBRs were greater than the limit of 1.30 for the W-3 correlation and 1.17 for the WRB-2 correlation. Additional information is contained in LR subsection 2.8.5.4.1.

Steam Line Break Accident

The event description is provided in LR subsection 2.8.5.1.2. Cases were analyzed for both HZP and HFP preconditions. For each of these cases, an appropriate methodology was applied. For the HFP cases, the RTDP methodology was used. For acceptability, calculated DNBRs must be above the RTDP design limit DNBR values. For the HZP cases, the RTDP methodology was not appropriate, so the mechanistic STDP was applied. For the STDP application, the DNBR limit was the approved W-3 correlation DNBR limit of 1.45, which has been acknowledged by the NRC as sufficiently high to ensure DNB criterion acceptance. Both HFP and HZP steam line break are typically reanalyzed for each reload. Limiting statepoints are used for confirmation of the DNBR criteria. The calculated minimum DNBR for HFP and HZP cases were above the DNBR limits for the SPU analysis.

Results

Core thermal-hydraulic analyses were performed in support of CPNPP operation at the SPU core power level of 3,612 MWt. Table 2.8.3-4 summarizes the available DNBR margin for CPNPP. It should be noted that the DNBR margin summaries are cycle-dependent and may vary from cycle-to-cycle in future reload designs. The continued satisfaction of the DNBR

criterion for reload cycles is confirmed via the approved reload methodology of WCAP-9272 (Reference 12).

For the SPU analysis, the design limit DNBR value for the 17x17 VANTAGE+ fuel is []^{a,c} for typical and thimble cells. After accounting for the plant-specific margin, the SAL DNBR is []^{a,c}. These SALs are employed in the DNB analyses.

With the SAL DNBR set, the core limits, axial offset limits, and dropped rod limits are generated. Based on these limits, the maximum $F_{\Delta H}^N$ limit that can be supported is 1.60. This limit incorporates all applicable uncertainties, including a measurement uncertainty of []^{a,c}, and is adjusted for power level using the following equation:

$$F_{\Delta H}^N = 1.60 \times (1 + 0.3(1-P))$$

where P is the fraction of full power.

The maximum rod bow penalty for the 17x17 VANTAGE+ fuel assembly is []^{a,c} at an assembly average burnup of 24,000 MWD/MTU. No additional rod bow penalty is required for burnups greater than 24,000 MWD/MTU since credit is taken for the effect of $F_{\Delta H}^N$ burndown due to the decrease in fissionable isotopes and the buildup of fission products.

For the two different bypass flow rates used in the thermal-hydraulic design analysis, the TDBF limit is 5.8 percent and the BEBF limit is []^{a,c} based on thimble plug removal.

For loss of flow studies, the minimum DNBRs calculated for PLOF, CLOF-UV, and CLOF-FD statepoints cases were greater than the safety analysis DNBR limit, thereby demonstrating compliance to the DNB design criterion for this event.

For locked rotor studies, the analysis indicates that there would be []^{a,c}-percent rods-in-DNB due to the locked rotor accident. The radiological consequences analysis conservatively assumed 10 percent of the fuel rods as failed rods and showed that the site dose limits were met. The locked rotor PCT was calculated with the VIPRE-01 code using STDP methodology. The results showed that the PCT limit of 2,700°F was met with a large margin.

The cycle-specific RCCA drop analysis confirms that all allowed pre-drop $F_{\Delta H}$ values do not violate the COLR limit, and the DNB design basis is met for SPU. In addition, the maximum linear heat rate from the RCCA drop analysis is lower than the fuel centerline melt limit. Therefore, the peak fuel centerline melt temperature criterion is also met for this event.

The analysis for the uncontrolled rod cluster control assembly withdrawal from subcritical (RWFS) event showed that the minimum DNBRs were greater than the limit of 1.30 for the W-3 correlation and 1.17 for the WRB-2 correlation.

The analyses for SLB showed the calculated minimum DNBR for HFP and HZP cases were above the DNBR limits for the SPU analysis.

The total DNBR penalty is []^{a,c}. The available DNBR margin is []^{a,c}.

The SPU analysis demonstrates that the combined DNBR margin gain is enough to accommodate the SPU to 3,612 MWt core power.

The fuel assembly lift forces were calculated for the uprated power and corresponding best-estimate flow with thimble plugs inserted. These lift forces were used to evaluate the hold-down spring forces, as described in subsection 2.8.1 of this report.

2.8.3.3 Conclusion

Luminant Power has reviewed the analyses related to the effects of the proposed SPU on the thermal and hydraulic design of the core and the RCS. Luminant Power concludes that the analyses have adequately accounted for the effects of the proposed SPU on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to the proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to T/H instability. Luminant Power further concludes that the analyses have adequately accounted for the effects of the proposed SPU on the hydraulic loads on the core and RCS components. Based on this, Luminant Power concludes that the thermal and hydraulic design will continue to meet the requirements of GDCs-10 and -12 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to thermal and hydraulic design.

2.8.3.4 References

1. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.
2. WCAP-11397, "Revised Thermal Design Procedure," April 1989.
3. WCAP-10444, "Reference Core Report VANTAGE+ Fuel Assembly," September 1985.
4. WCAP-7308, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
5. WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979.
6. Letter from Rahe, E. P., Jr. (Westinghouse) to Miller, J. R. (NRC), "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," NS-EPR-2515, October 9, 1981; and Letter from Rahe, E. P., Jr. (Westinghouse) to Miller, J. R. (NRC), "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," NS-EPR-2572, March 16, 1982.

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7. Letter from Berlinger, C. (NRC) to Rahe, E. P., Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1986.
 8. WCAP-15063, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," Rev. 1 with Errata, July 2000.
 9. ASME-65-HT-31, "Forced Convection Heat Transfer at High Pressure after the Critical Heat Flux," ASME-65-HT-31, 1965.
 10. L. Baker, Jr. L.C. Just, "Studies of Metal-Water Reactors at High Temperature," ANL-6548, Argonne National Laboratories, May 1962.
 11. WCAP-11394, "Methodology for the Analysis of the Dropped Rod Event," January 1990.
 12. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

Table 2.8.3-1		
Thermal-Hydraulic Design Parameter Comparisons		
Thermal-Hydraulic Design Parameters	Current Design Value⁽¹⁾	SPU Analysis Value
Reactor Core Heat Output, MWt ⁽²⁾	3,458	3,612
Reactor Core Heat Output, 10 ⁶ BTU/Hr ⁽²⁾	11,799	12,325
Heat Generated in Fuel, %	97.4	97.4
Pressurizer Pressure, Nominal, psia	2,250	2,250
Radial Power Distribution ⁽³⁾	1.55(1+0.3(1-P))	1.60(1+0.3(1-P))
HFP Nominal Coolant Conditions		
Vessel Thermal Design Flow Rate (including bypass) ⁽¹⁾ 10 ⁶ lbm/hr GPM	142.04 382,800	142.30 382,800
Core Flow Rate (excluding Bypass based on TDF) ⁽⁴⁾ 10 ⁶ lbm/hr GPM	133.80 360,598	134.04 360,598
Core Flow Area, ft ²	54.1	54.1
Core Inlet Mass Velocity (based on TDF), 10 ⁶ lbm/hr-ft ²	2.472	2.476
Nominal Vessel/Core Inlet Temperature, °F	559.2	558.0
Vessel Average Temperature, °F	589.2	589.2
Core Average Temperature, °F	592.6	592.8
Vessel Outlet Temperature, °F	619.2	620.4
Core Outlet Temperature, °F	622.5	623.8
Average Temperature Rise in Vessel, °F	60.0	62.4
Average Temperature Rise in Core, °F	63.3	65.8
Heat Transfer		
Active Heat Transfer Surface Area, ft ²	57,505	57,505
Average Heat Flux, BTU/hr-ft ²	199,900	208,802
Average Linear Power, kW/ft	5.520	5.766
Peak Linear Power for Normal Operation ⁽⁵⁾ kW/ft	13.36	14.41
Peak Linear Power for Prevention of Centerline Melt, kW/ft	22.4	22.4
Pressure Drop Across Core ⁽⁶⁾ , psi	29.1	28.9
Notes:		
1. Unit 1 flow rates are used for comparison. The lower Unit 1 flow rates are more limiting with respect to DNB. Also, these flows are to be used for both units for uprate condition.		
2. The proposed power level of 3,612 MWt has been used for all thermal-hydraulic design analyses.		
3. $P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$		
4. Based on thimble plugs inserted.		
5. Based on maximum F ₀ of 2.42 for current design and 2.5 for uprate design.		
6. Based on best-estimate flow and design bypass flow rates.		

Table 2.8.3-2	
Limiting Parameter Direction for DNB	
Parameter	Limiting Direction for DNB
$F_{\Delta H}^N$, nuclear enthalpy rise hot-channel factor	maximum
Heat generated in fuel (%)	maximum
Reactor core heat output (MWt)	maximum
Average heat flux (BTU/hr-ft ²)	maximum
Nominal vessel/core inlet temperature (°F)	maximum
Core pressure (psia)	minimum
Pressurizer pressure (psia)	minimum
Thermal design flow for non-RTDP analyses (gpm)	minimum
Minimum measured flow for RTDP analyses (gpm)	minimum

Table 2.8.3-3	
RTDP Uncertainties	
Parameter	Uncertainty Used in SPU Safety Analysis
Power	[] ^{a,c}
RSC Flow	[] ^{a,c}
Pressure	[] ^{a,c}
Inlet Temperature	[] ^{a,c}

**Table 2.8.3-4
DNBR Margin Summary⁽¹⁾**

DNB Correlation		WRB-2
DNBR Correlation Limit		1.17
DNBR Design Limit	(TYP/THM) ⁽²⁾	[] ^{a,c}
DNBR SAL	(TYP/THM)	[] ^{a,c}
DNBR Retained Margin ⁽³⁾	(TYP/THM)	[] ^{a,c}
Rod Bow DNBR Penalty ⁽⁴⁾	(TYP/THM)	[] ^{a,c}
Lower Plenum Flow Anomaly	(TYP/THM)	[] ^{a,c}
Temperature Bias (1.49°F)	(TYP/THM)	[] ^{a,c}
Total Penalty	(TYP/THM)	[] ^{a,c}
Available DNBR Margin	(TYP/THM)	[] ^{a,c}

Notes:

- The values below correspond to RTDP. HZP steam line break and RWFS are based on STDP. The DNBR limit for HZP steam line break is 1.45. The minimum DNBR for SPU analysis was above this limit. The HZP steam line break is normally analyzed each cycle.
The DNBR limitation for RWFS are 1.30 with W-3 correlation below the first mixing vane grid and 1.17 above this grid. The minimum DNBRs were above these limits for SPU analysis.
- TYP = Typical cell.
THM = Thimble cell.
- DNBR margin is the difference between the SAL and the design limit DNBRs.
- The rod bow penalty is []^{a,c}.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of the Control Rod Drive System

2.8.4.1.1 Regulatory Evaluation

Luminant Power's review covered the functional performance of the control rod drive system (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during anticipated operational occurrences (AOOs), and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements.

The acceptance criteria are based on:

- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC-23, insofar as it requires that the protection system be designed to fail into a safe state.
- GDC-25, insofar as it requires that the protection system be designed to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded for any single malfunction of the reactivity control systems.
- GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes.
- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained.
- GDC-28, insofar as it requires that the reactivity control systems be designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.
- GDC-29, insofar as it requires that the protection and reactivity control systems be designed to ensure an extremely high probability of accomplishing their safety functions in event of AOOs.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of Units 1 and 2 design relative to:

- GDC-4, Environmental and Dynamic Effects Design Bases, is described in FSAR Section 3.1.1.4.

SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents including LOCAs. These items are either protected from accident conditions or designed to withstand, without failure, exposure to the combination of temperature, pressure, humidity, radiation, and dynamic effects expected during the required operational period.

Physical separation, physical protection, pipe restraints, and redundancy are included in the design of safety-related systems to ensure that each such system performs its intended safety function.

SSCs important to safety are classified and designed in accordance with the codes and classifications indicated in FSAR Section 3.2.

- GDC-23, Protection System Failure Modes, is described in FSAR Section 3.1.3.4.

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. FSAR Sections 7.2 and 7.3 discuss this protection system.

- GDC-25, Protection System Requirements for Reactivity Control Malfunctions, is described in FSAR Section 3.1.3.6.

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by full-length rod insertion is completely independent of the normal control function, since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Therefore, in the postulated accidental withdrawal (assumed to be initiated by a control malfunction), flux, temperature, pressure, level and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

FSAR Chapter 15 discusses analyses of the effects of possible malfunctions. These analyses show that for postulated dilution during refueling, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of the

dilution, terminate the source of the dilution, and initiate boron before the shutdown margin is lost. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control systems are provided. They are the rod cluster control assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The rod control system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and AOOs, thereby ensuring that specific fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses, and the most reactive rod cluster is assumed to be in the fully withdrawn position.

FSAR Chapter 4 presents details of the construction for the RCCAs. FSAR Chapter 7 discusses their operation. FSAR Chapter 15 includes performance analyses under accident conditions.

- GDC-27, Combined Reactivity Control Systems Capability, is described in FSAR Section 3.1.3.8.

CPNPP Units 1 and 2 are provided with a means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. FSAR Chapters 4 and 9 discuss these means in detail. Combined use of the rod cluster control system and the chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full-out upon trip for this determination. FSAR Chapter 15 describes accident assumptions in detail.

- GDC-28, Reactivity Limits, is described in FSAR Section 3.1.3.9.

The maximum reactivity worth of control rods and the maximum rate of reactivity insertion employing control rods are limited to values that prevent rupture of the RCS boundary or disruption of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The maximum positive reactivity insertion rates for the withdrawal of RCCAs and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCAs and of the CVCS. Technical Specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional

assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of FSAR Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in FSAR Section 4.3. The capability of the CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in FSAR Section 15.

- GDC-29, Protection Against Anticipated Operational Occurrences, is described in FSAR Section 3.1.3.10.

The protection and reactivity control systems are designed to assure an extremely high probability of accomplishing their safety functions in any operational occurrences. Equipment used in these systems is designed, constructed, operated, and maintained with a high level of reliability. FSAR Chapter 7 provides details of system design.

As stated in FSAR Section 3.9N.4.1, the CRDMs are located on the dome of the reactor vessel head. They are coupled to RCCAs that have neutron absorber material over the entire length of the control rods and derive their name from this feature. The primary function of the CRDM is to insert, withdraw, or hold stationary, RCCAs within the core to control core average temperature and to shutdown the reactor. The CRDM is a magnetically operated jack. A magnetic jack is an arrangement of three electromagnets that are energized in a controlled sequence by a power cycler to insert or withdraw RCCAs in the reactor core in discrete steps. Rapid insertion of the RCCAs occurs when electric power is interrupted. The CRDM can be tripped during any part of the power cycler sequence if electric power to the coils is interrupted, thereby releasing the drive rod assembly and inserting the RCCA.

As stated in FSAR Section 4.6, the CRDS includes the CRDMs (discussed in FSAR Section 3.9N.4.1), the rod control system (discussed in FSAR Section 7.7.1.2) and the reactor trip switchgear (discussed in FSAR Section 7.2.1.1).

FSAR Section 4.6.2 also states in part that the CRDS has been analyzed in detail in a failure modes and effects analysis (FMEA) in WCAP-8976 (Reference 1). These studies and the analyses presented in FSAR Section 15.0 demonstrate that the CRDS performs its intended safety function - reactor trip, by putting the reactor in a subcritical condition when a safety system setting is approached, with any assumed credible failure of a single active component. The essential elements of the CRDS are isolated from the nonessential portions of the CRDS (the rod control system).

The design of the CRDM is such that failure of the CRDM cooling system will, in the worst case, result in an individual control rod drop or a full reactor trip.

Other FSAR sections that address the design features and function of the CRDM and chemical reactivity control systems are as follows:

- FSAR Section 7.2, Reactor and Trip System, which provides a description of the reactor trip system interface with CRDS
- FSAR Section 7.7.1.2, Rod Control System, which provides a description of the operation of the CRDS
- FSAR Section 7.7.1.3.2, Rod Position Monitoring of Full Length Rods, which provides a description of the digital rod position indication system and the demand position system
- FSAR Section 9.4.A.1.2, CRDM Ventilation System, which provides a description of the design of the CRDM cooling system
- FSAR Section 15.4, Reactivity and Power Distribution Anomalies, which provides a description of the transient and accident analyses associated with the malfunctions of the CRDS and CVCS

2.8.4.1.2 Technical Evaluation

2.8.4.1.2.1 Introduction

The potential impact of the stretch power uprate (SPU) on the CRDS results from the temperature effects associated with increasing reactor core thermal power to 3,612 MWt.

CRDMs use electro-magnetic coils to position the RCCAs within the reactor core. The insulation and potting materials used in the construction of the coils are subject to thermal aging. In order to reduce the thermal aging, CRDM cooling systems were designed to remove heat supplied by conduction and convection from the reactor head and reactor coolant.

2.8.4.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The temperature of the CPNPP Units 1 and 2 reactor vessel head is the same as the reactor vessel inlet temperature. Licensing Report (LR) Section 1.1 indicates that the full power reactor vessel inlet temperature is effectively unchanged at 558°F., i.e., the maximum reactor vessel inlet temperature for both the current condition and all cases evaluated for the SPU.

The specific acceptance criteria are to demonstrate that the temperatures associated with the SPU on the components and coils of the CRDM remain acceptable.

As a result of the SPU, there are no physical changes required to the CRDS, operating coil stacks, power supplies, solid-state electronic control cabinets, or the control rod drive cooling system.

2.8.4.1.2.3 Description of Analyses and Evaluations

The changes to the CRDM operating temperatures as a result of the SPU are small, and therefore, have no effect upon CRDM functionality and operability.

Accident and transient analyses for the events listed in the FSAR Chapter 15 were performed for the SPU conditions listed in LR Section 1.1, Nuclear Steam Supply System Parameters. These analyses are described in LR Section 2.8.5, Accident and Transient Analyses, of this report. All events associated with the CRDS provided acceptable results and maintained departure from nucleate boiling (DNB), the RCS pressure, and main steam system pressure within the acceptable limits.

Analyses and evaluations of the impact of the SPU on the structural integrity of the CRDS during normal, transient, and accident conditions were also performed using the SPU conditions listed in LR Section 1.1 of this report. These analyses and evaluations are discussed in LR subsection 2.2.2.4, Control Rod Drive Mechanisms. The results of the analyses and evaluations determined the structural integrity of the CRDS remained within acceptable limits at the SPU conditions.

With respect to CRDM cooling, as indicated above, the temperature of the CPNPP Units 1 and 2 reactor vessel head is the same as the reactor vessel inlet temperature. Since the primary source of heat to the CRDM cooling system is conduction and convection from the reactor vessel head, the temperature of the CRDMs and amount of heat rejected to the Containment Building are also a function of the reactor vessel inlet temperature. To evaluate the effects of the SPU on the CRDM cooling system, the maximum current condition reactor inlet temperature is compared to the maximum reactor inlet temperature following implementation of the SPU. LR Section 1.1 indicates that the full-power reactor vessel inlet temperature of 558°F is the maximum reactor vessel inlet temperature for both the current condition and all cases evaluated for the SPU. Given that the maximum reactor vessel head temperature remains unchanged for the SPU, the performance of the CRDM cooling system and maximum heat load on containment from this system are not affected by the SPU.

2.8.4.1.2.4 Results

Luminant Power has reviewed the functional design of the CRDS and the CRDM cooling system for the effects of the SPU. Plant operability described in LR subsection 2.4.2, Plant Operability/Component Sizing, and LR Section 2.8.5, Accident and Transient Analyses, have demonstrated that at SPU conditions the rod control system continues to satisfy the design basis for reactivity control and ensure specified acceptable fuel design limits are met for any single malfunction of the reactivity control systems.

The impact of the SPU on the structural integrity of the CRDMs discussed in LR subsection 2.2.2.4 of this report, Control Rod Drive Mechanism. No modifications have been made to the hardware, logic or operation of the system that affect the system's current ability to fail into a safe state. The impact of the SPU on the control rod drive cooling system was evaluated and there is no impact to the cooling system.

2.8.4.1.3 Conclusions

Luminant Power has reviewed the analyses related to the effects of the proposed SPU on the functional design of the CRDS and concluded that the evaluation has adequately accounted for the effects of the proposed SPU on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed SPU. It is further concluded that the evaluation has demonstrated that sufficient cooling exists to ensure the system's design bases will be followed upon implementation of the proposed SPU. Based on this, Luminant Power concludes that the CRDS and associated analyses will meet the requirements of GDCs-4, -23, -25, -26, -27, -28, and -29 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the functional design of the CRDS.

2.8.4.2 Overpressure Protection During Power Operations

2.8.4.2.1 Regulatory Evaluation

Overpressure protection for the reactor coolant pressure boundary (RCPB) during power operation is provided by relief and safety valves and the reactor protection system (RPS). The review covered pressurizer relief and safety valves and the piping from these valves to the quench tank.

The acceptance criteria are based on:

- General Design Criterion (GDC)-15, insofar as it requires that the reactor coolant system (RCS) and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).
- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the CPNPP design relative to:

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the reactor coolant and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions. FSAR Chapter 5 discusses the RCS design.

- GDC-31, Fracture Prevention of Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.4.2.

Close control is maintained over material selection and fabrication for the RCS to assure that the boundary behaves in a non-brittle manner. The RCS materials exposed to the coolant are corrosion-resistant stainless steel or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests, in accordance with 10 CFR Part 50, Appendix G.

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G, Protection Against Non-Ductile Failure. This approach specifies that allowed stress intensity factors for all vessel operating conditions shall not exceed the reference stress intensity factor (KIR) for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material (RT_{NDT}) due to irradiation.

CPNPP FSAR Section 5.2.2 states that overpressure protection is provided for the RCS by the pressurizer safety valves along with the RPS and associated equipment. This protection is afforded for the following events:

- Loss of electrical load and/or turbine trip
- Uncontrolled rod withdrawal at power
- Loss of reactor coolant flow
- Loss of normal feedwater
- Loss of offsite power to the station auxiliaries

These events bound those credible events that could lead to overpressure of the RCS if adequate overpressure protection were not provided.

Pressurizer safety valve sizing is sufficient to prevent exceeding 110 percent of system design pressure for the events listed in this section. As indicated in FSAR Section 5.2.2, the total relief capacity of the pressurizer safety valves installed at CPNPP was originally established based on the method described in WCAP-7769 Revision 1 (Reference 3). This method, which ensures compliance with overpressure protection requirements of the ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, involved the analysis of a complete loss of steam flow to the

turbine with credit taken for the actuation of the main steam safety valves. In this analysis, main feedwater flow was assumed to be maintained and no credit was taken for the following:

- Reactor Trip,
- Pressurizer Power-Operated Relief Valve Operation,
- Steam Line Power-Operated Relief Valve Operation,
- Steam Dump System Operation,
- Reactor Control System Operation,
- Pressurizer Level Control System Operation,
- Pressurizer Spray Valve

A description of the pressurizer safety valves, including a design basis discussion, is provided in FSAR Section 5.4.13.

2.8.4.2.2 Technical Evaluation

2.8.4.2.2.1 Introduction

The limiting credible event with respect to primary and secondary system overpressurization is the loss of external electrical load/turbine trip (LOL/TT) event. This section briefly summarizes the LOL/TT analysis performed for CPNPP Units 1 and 2, which demonstrates that the overpressure criteria continue to be met for the SPU program. Details of that analysis are given in Licensing Report (LR) subsection 2.8.5.2.1, Loss of External Electrical Load, Turbine Trip, Steam Pressure Regulator Failure, and Loss of Condenser Vacuum.

The technical evaluations of the RCS and components are described in LR subsection 2.2.2, Pressure-Retaining Components and Component Supports. The technical evaluation of the piping from the safety valves to the pressurizer relief tank (PRT) is described in LR subsection 2.5.2, Pressurizer Relief Tank.

Note that overpressure protection during low temperature operation is discussed in LR subsection 2.8.4.3, Overpressure Protection During Low-Temperature Operation.

2.8.4.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The LOL/TT cases for maximizing the RCS and main steam system (MSS) peak pressures were analyzed using the standard thermal design procedure (STDP). Initial uncertainties on reactor power and reactor coolant flow, temperature, and pressure were applied in the conservative direction to obtain the initial plant conditions for the transient analyses in which overpressurization of the RCS or MSS is the primary concern (for the departure from nucleate boiling (DNB) analysis, the uncertainties associated with power, pressure, temperature, and flow are statistically accounted for in the safety analysis DNB ratio (DNBR) correlation limit). Further details of the input parameters and assumptions for the LOL/TT analyses at the uprated power are discussed in LR subsection 2.8.5.2.1.

For this event, primary and secondary system pressures must remain below 110 percent of their respective design pressures (an RCS pressure limit of 2,748.2 psia and secondary side pressure limit of 1,318.2 psia) at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfies the requirements of GDC-15 and -31.

The NRC Standard Review Plan (SRP) 5.2.2 (Overpressure Protection) Section II identifies the following applicable NRC regulations that must be satisfied with respect to overpressure protection: GDC-15, GDC-31, 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi), 10 CFR 52.47(a)(8), 10 CFR 52.79(a)(17), 10 CFR 52.47(b)(1), and 10 CFR 52.80(a). For the SPU, demonstrating compliance with GDC-15 and GDC-31 satisfies the intent of SRP 5.2.2. The other regulations, which do not need to be addressed for SPU, include requirements for: (1) relief and safety valve performance testing and valve position indication (not changing for the proposed uprate), (2) design certification applications (not applicable), and (3) combined license (COL) applications (not applicable).

2.8.4.2.2.3 Description of Analyses and Evaluations

For the LOL/TT event, the behavior of CPNPP Units 1 and 2 was analyzed for a complete loss of steam load from full power without a direct reactor trip. A detailed analysis was performed, as described in LR subsection 2.8.5.2.1, to determine the plant transient conditions following a total loss of load.

In addition, the allowable power levels with inoperable main steam safety valves have been determined and specified in Technical Specification 3.7.1.1. This table is being revised as described in License Amendment Request (LAR) 07-006 submitted in letter TXX-07108. This Technical Specification allows CPNPP Units 1 and 2 to operate with a reduced number of operable main steam safety valves (MSSVs) at a reduced power level, as determined by resetting the power range high neutron flux setpoint. In order to preclude secondary side overpressurization in the event of a LOL/TT event, the maximum power level allowed for operation with inoperable MSSVs must be below the heat removing capability of the operable MSSVs. Table 3.7.1-1 of the CPNPP Technical Specifications defines the power range high neutron flux setpoint corresponding to one, two, or three inoperable MSSVs.

2.8.4.2.2.4 Results

The results of the LOL/TT analysis documented in LR subsection 2.8.5.2.1 demonstrate that the primary and secondary pressure limits are met at the proposed SPU conditions. No changes were needed to the main steam safety valves in order to meet the applicable pressure limits.

Operation at the SPU conditions will have no impact on the reliability of the reactor protection system or the safety valves. Therefore, conclusions of the Overpressure Protection Report referenced in the FSAR remain valid.

Table 2.8.4.2-1 provides the maximum allowable power range neutron flux high setpoints with inoperable MSSVs described in the LAR 07-006 transmitted in TXX-07108.

2.8.4.2.3 Conclusions

Luminant Power has reviewed the analyses related to the effects of the proposed SPU on the overpressure protection capability of the plant during power operation. It is concluded that the analyses have (1) adequately accounted for the effects of the proposed SPU on pressurization events and overpressure protection features and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, it is concluded that the overpressure protection features will continue to provide adequate protection to meet GDC-15 and GDC-31 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to overpressure protection during power operation.

Table 2.8.4.2-1

Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable MSSVs

Number of Operable Safety Valves on Any Operating SG	TXX-07108 Technical Specification Change Request Setpoint (% of RTP)	Current Technical Specification Setpoint (% of RTP)
4	≤ 61	≤ 87
3	≤ 43	≤ 65
2	≤ 26	≤ 43

2.8.4.3 Overpressure Protection During Low-Temperature Operation

2.8.4.3.1 Regulatory Evaluation

Overpressure protection for the reactor coolant pressure boundary (RCPB) during low-temperature operation of the plant is provided by pressure-relieving systems that function during the low-temperature operation. The power-operated relief valves (PORVs) discharge into the pressurizer discharge system (Final Safety Analysis Report (FSAR) Section 5.4.11).

Luminant Power's acceptance criteria are based on:

- General Design Criterion (GDC)-15, insofar as it requires that the reactor coolant system (RCS) and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.
- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

Current Licensing Basis

The Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 FSAR Section 3.1 briefly discusses the extent to which the design criteria for the plant structures, systems, and components (SSCs) important to safety comply with Title 10, Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP Units 1 and 2 designs relative to the GDC is discussed in FSAR Sections 3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.1.5, and 3.1.6. Specifically, the adequacy of the CPNPP Units 1 and 2 RCPB overpressure protection during low-temperature operation was assessed by reviewing conformance to:

- GDC-15 is described in FSAR Section 3.1.2.6, General Design Criterion 15 – Reactor Coolant System Design. The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions. In addition, RCPB components achieve a large margin of safety by the use of proven ASME materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and of integrated hydrostatic testing of assembled components. FSAR Chapter 5 discusses the RCS design.
- GDC-31 is described in FSAR Section 3.1.4.2, General Design Criterion 31 – Fracture Prevention of RCPB. Close control is maintained over material selection and fabrication for the RCS to ensure that the boundary behaves in a nonbrittle manner. Those RCS materials which are exposed to the coolant are corrosion resistant, stainless steel, or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is

established by Charpy V-notch and drop weight tests in accordance with 10 CFR Part 50, Appendix G.

As part of the reactor vessel specification, the following requirements that are not specified by the applicable ASME codes were performed: ultrasonic testing, radiation surveillance, and control of the material chemistry of the reactor vessel core region.

The fabrication and quality control techniques used in the fabrication of the RCS are equivalent to those used for the reactor vessel. The inspections of the reactor vessel, pressurizer, piping, pumps, and steam generator are governed by the requirements of ASME Codes (see FSAR Chapter 5).

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the ASME B&PV Code, Section III, Appendix G, Protection Against Non-Ductile Failure. The approach specifies that allowed stress intensity factors for all vessel operating conditions shall not exceed the reference stress intensity factor (KIR) for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material (RT_{NDT}) due to irradiation.

The CPNPP Units 1 and 2 low-temperature overpressure protection (LTOP) system is applicable in Modes 4, 5, and 6 as specified in CPNPP Units 1 and 2 Technical Specification Section 3.4.12. The LTOP system shall be operable with a maximum of zero safety injection pumps and two charging pumps capable of injecting into the RCS and the accumulators isolated and one of the following pressure relief capabilities:

- Two PORVs with lift settings within the limits specified in the Pressure and Temperature Limits Report (PTLR)
- Two RHR suction relief valves with setpoints as specified in Technical Specification Section 3.4.12
- One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint as specified in Technical Specification Section 3.4.12
- The RCS depressurized and an RCS vent as specified in Technical Specification Section 3.4.12

2.8.4.3.2 Technical Evaluation

Existing Design Basis Requirements

The LTOP System acts as a backup to the reactor operators to mitigate RCS pressurization transients at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G.. The reactor vessel is the limiting RCPB component for demonstrating such

protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Appendix G requirements during the LTOP MODES.

The reactor operators control the RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits. The LTOP System for pressure relief consists of two PORVs with reduced lift settings, or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurizing the RCS for the required coolant input capability.

Whenever the RCS cold leg temperature is below the temperature setpoint specified in the PTLR for the LTOP system, the low-pressure PORV setpoint is automatically enabled. Pressure transients caused by mass injection (MI) or heat injection (HI) are terminated below the limits of 10 CFR 50, Appendix G, as amended by ASME Code Case N-641, by automatic operation of the pressurizer PORVs.

2.8.4.3.2.1 Input Parameters, Assumptions, and Acceptance Criteria

The critical parameters for the LTOP system PORV setpoint determination are the 1) design basis MI and HI transients, 2) RCS volumes, 3) MI flow rates, 4) differential pressures between the reactor vessel and the hot leg pressure transmitter, 5) wide range pressure and temperature uncertainties, 6) pressurizer PORV characteristics, and 7) pressure-temperature limits of 10 CFR 50, Appendix G. The impact of the SPU program on these parameters is discussed in LR subsection 2.8.4.3.2.2.

The acceptance criterion for the LTOP system analysis is that the LTOP system PORV setpoints should prevent the RCS pressure from exceeding the P-T limits of 10 CFR 50, Appendix G for the design basis MI and HI transients.

2.8.4.3.2.2 Description of Analyses and Evaluations

There are no changes in the design basis MI and HI transients from the current analyses of record for CPNPP as a result of the SPU program.

There are no changes to the RCS volumes, as there are no physical changes to CPNPP Units 1 and 2 components as part of the SPU program.

There are no changes to the MI flow rates and differential pressures between the reactor vessel and the hot leg pressure transmitter as a result of the SPU program. The RCS wide range pressure uncertainty and RCS hot and cold leg temperature uncertainty will not change. In addition, the SPU does not result in any other plant changes (such as pressurizer PORV characteristics) from the current analyses of record.

The existing P-T limits for 36 EFPY are not impacted by the SPU and therefore remain applicable.

The only input parameter changes to the LTOP system analysis due to the SPU are to the nominal full-power conditions as presented in LR Section 1.1. The LTOP system PORV setpoints analyses for both units are performed at reactor shutdown and RCS cold conditions. Therefore, the SPU does not affect the LTOP system PORV setpoint determination.

Based on this evaluation, the existing LTOP system setpoints based on the P-T limits at 36 EFPY remain valid for the SPU program.

2.8.4.3.2.3 Results

The current analyses of record, which were based on the methodology presented in Reference 2, showed that the LTOP system PORV setpoints associated with the heatup and cooldown curves for Units 1 and 2 meet the acceptance criterion. It is further concluded that the existing setpoints are not impacted by the uprate and remain applicable for the CPNPP.

2.8.4.3.3 Conclusions

Luminant Power has reviewed the analyses related to the effects of the proposed SPU on the overpressure protection capability of the plant during low-temperature operation. Luminant Power concludes that:

- The analyses adequately accounted for the effects of the proposed SPU on pressurization events and overpressure protection features
- The plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded.

Based on this, Luminant Power concludes that the LTOP system features will continue to provide adequate protection to meet the CPNPP Units 1 and 2 current licensing basis requirements with respect to GDC-15 and -31 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU is acceptable with respect to overpressure protection during low temperature operation.

2.8.4.3.4 References

1. WCAP-8976, "Failure Modes and Effects Analysis (FMEA) of the Solid State Full Length Rod Control System."
2. WCAP-14040, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.
3. WCAP-7769, "Topical Report Overpressure Protection for Westinghouse PWR," October 1971.

2.8.4.4 Residual Heat Removal System

2.8.4.4.1 Regulatory Evaluation

The residual heat removal system (RHRS) cools down the reactor coolant system (RCS) following shutdown. The RHRS is a low-pressure system that takes over the shutdown cooling function when the RCS temperature is reduced. Westinghouse evaluated the effect of the proposed SPU on the functional capability of the RHRS to cool the RCS following shutdown and provide decay heat removal.

The acceptance criteria are based on:

- General Design Criterion (GDC) -4, insofar as it requires that structures, systems, and components (SSCs) important to safety be protected against dynamic effects.
- GDC-5, insofar as it requires that SSCs important to safety SSCs not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.
- GDC-34, which specifies requirements for a residual heat removal system.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP RHRS relative to conformance to:

- GDC-4, Environmental and Dynamic Effects Design Bases, is described in FSAR Section 3.1.1.4.

The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible, with the environmental conditions associated with normal operating maintenance, testing, and postulated accidents including a loss-of-coolant accident (LOCA). Environmental conditions are described in FSAR Section 3.11.

These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power plant.

Details of the design, environmental testing, and construction of these SSCs are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

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- GDC-5, Sharing of Structures, Systems, and Components, is described in FSAR Section 3.1.1.5.

The facilities that have shared systems or components are tabulated in FSAR Section 1.2 along with references to the appropriate sections containing specific design details. The residual heat removal system (RHRS) is not a shared system.

- GDC-34, Residual Heat Removal, is described in FSAR Section 3.1.4.5.

The RHRS, in conjunction with the steam and power conversion system, is designed to transfer the fission production decay heat and other residual heat from the reactor core within acceptable limits. The auxiliary feedwater system provides backup for the steam and power conversion system in this function. The auxiliary feedwater system is described in FSAR Section 10.4.9.

The cross-over from the steam and power conversion system to the RHRS occurs at approximately 350°F and 425 psig.

Suitable redundancy at temperatures below approximately 350°F is accomplished with the two RHR pumps (located in separate compartments with means available for draining and monitoring of leakage), the two heat exchangers, and the associated piping, cabling, and electric power sources. The RHRS is capable of operating on either onsite or offsite electrical power systems.

2.8.4.4.2 Technical Evaluation

2.8.4.4.2.1 Introduction

The RHRS is described in FSAR Section 5.4.7. The RHRS transfers heat from the RCS to the component cooling water system (CCS) to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal plant cooldown and maintains this temperature until the plant is started up again.

Parts of the RHRS also serve as parts of the ECCS during the injection and recirculation phases of a LOCA. This is described in FSAR Section 6.3.

The RHRS also is used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of the refueling operations.

2.8.4.4.2.2 Description of Analyses and Evaluations

The SPU increases the residual heat generated in the core during normal cooldown, refueling operations, and accident conditions. This provides a higher total heat load on the RHR heat exchangers during cooldown and also during refueling outages. The removal of core decay heat for accident conditions is addressed in Licensing Report (LR) subsection 2.6.5, Containment Heat Removal. The increased heat loads will be transferred to the CCS and

ultimately to the service water system (SW). Evaluation of the SPU performance of the RHRS, in conjunction with the CCS and SW, with the increased heat loads is addressed in LR subsection 2.5.4.3, Reactor Auxiliary Cooling Water System, LR subsection 2.5.4.2, Service Water System, and Section 2.5.4.4, Ultimate Heat Sink.

The SPU affects the plant cooldown time(s) since core power, and therefore the decay heat, increases. The plant cooldown calculation was performed at decay heat levels based on the updated core power level. The RCS heat capacity and the other CCS heat loads were explicitly considered in this analysis. The analysis was performed to demonstrate that the RHRS and CCS continue to comply with their design basis functional requirements and performance criteria for plant cooldown under the SPU conditions.

This analysis considered the following FSAR cases (FSAR 5.4.7.1) for the RHRS cooldown design basis:

- Normal plant cooldown
- Forced plant cooldown
- Fire safe shutdown
- Reactor System Branch Technical Position 5-1 (RSB 5-1) safe shutdown

The following considerations were applied to these cooldown analyses:

- The reactor coolant pump (RCP) heat load was used in the Normal Plant Cooldown and the Forced Plant Cooldown runs. Only one RCP is required in Modes 3 and 4.
- The maximum RHR tube side flow rate is approximately 3,800 gpm. The RHR flow is throttled through the tube side of the RHR heat exchanger in the early stage of cooldown to maintain CCS temperature below a normal maximum value of 122°F.
- The initial SW temperature was assumed to be 102°F.
- The normal plant cooldown time to 140°F for refueling (Mode 6) or cold shutdown maintenance (Mode 5) with both trains of CCS and RHR available (that is, two RHR pumps and heat exchangers and two CCS pumps and heat exchangers) may increase from 28 hours to approximately 33 hours for the SPU assuming a normal cooldown start time of 4 hours after reactor shutdown. Since there is no design criterion for normal plant cooldown time, these increases in values, based on design conditions, are acceptable.
- The forced plant cooldown analysis was performed to demonstrate continued compliance with Technical Specification actions. The analysis assumes a single train of RHR is placed in service 12 hours after shutdown. The results show that with a single train of RHR in service, the plant can be brought to less than 200°F in less than 24 hours after RHRS initiation.

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- The fire safe shutdown requires that cold shutdown (Mode 5 ($\leq 200^{\circ}\text{F}$)) be achieved in 72 hours after reactor shutdown. Continued compliance with this time limit was demonstrated at the SPU conditions. The worst-case cooldown scenario assumes loss-of-offsite power and only one train of RHR and CCS equipment available. At SPU conditions, one train of RHR and CCS equipment can cool the RCS to cold shutdown conditions (200°F) within approximately 8 hours after RHR initiation. It should be noted that the uncertainty in strap-on resistance temperature detectors (RTDs) used for RCS temperature indication from the remote shutdown panel (for example, during design basis fire) could add up to about 6 hours prior to RHR initiation. By increasing the cooldown time by 6 hours, it is evident that the time to reach 200°F remains significantly less than the required 72 hours.
 - Reactor Systems Branch Technical Position 5-1 (RSB 5-1) requires RHR initiation at 9 hours based on the condensate storage tank capacity. The design of the RHR system is such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems in a reasonable period of time. The plant is capable of being cooled via natural circulation and reaching cold shutdown conditions within 36 hours, including the time required to perform any manual actions.

The SPU does not impact the design temperature and pressure of the RHRS piping and associated components. Refer to LR subsection 2.5.1.3, Pipe Failures for the RHRS piping evaluation and the environmental and dynamic effects evaluation relative to meeting the CPNPP current licensing basis requirements with respect to GDC-4.

The SPU has no affect on the ability of the RHRS to remove residual heat at reduced RCS inventory. Therefore, CPNPP will continue to meet the current licensing basis requirements with respect to NRC Generic Letter (GL) 88-17. Additional discussion of GL 88-17 is provided in LR subsection 2.8.7.3, Loss of Residual Heat Removal at Mid-Loop.

2.8.4.4.2.3 Results

Continued compliance with the RHRS cooldown performance requirements was demonstrated at the SPU conditions with no plant hardware changes being necessary. The SPU cooldown analysis results show that although cooldowns may be extended, they will remain within acceptable limits.

Evaluations described in LR subsection 2.5.1.3, Pipe Failures, show the response of the RHRS piping to the SPU environmental and dynamics effects remain acceptable relative to meeting the CPNPP current licensing basis requirements with respect to GDC-4.

The SPU has no affect on the ability of the RHRS to comply with GDC-34. The SPU operating conditions have no adverse impact on the following:

- The design and operating characteristics of the RHRS with respect to its shutdown and long-term cooling function (Refer to the above evaluation relative to the RHRS cooldown performance at SPU conditions).
- The isolation provisions provided between the high pressure RCS and the lower pressure RHRS and the RHRS overpressure protection features.

2.8.4.4.3 Conclusion

Luminant Power has reviewed the effects of the SPU on the RHRS. It has been concluded, based on the analyses discussed in this section, that the effects of the SPU on the system are adequately accounted for and it has been demonstrated that the RHRS will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the RHRS will continue to meet CPNPP current licensing requirements with respect to GDC-4, GDC-5, and GDC-34 following implementation of the SPU. Therefore, the SPU is acceptable with respect to the RHRS.

2.8.5 Accident and Transient Analyses

2.8.5.0 Non-LOCA Analyses Introduction

This section summarizes the non-loss-of-coolant accident (non-LOCA) transient analyses and evaluations performed to support the stretch power uprate (SPU) program for the Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2.

2.8.5.0.1 Fuel Design Mechanical Features

The fuel type currently in use at CPNPP Units 1 and 2 is the Westinghouse 17×17 VANTAGE+ fuel design which contains intermediate flow mixing (IFM) grids that are designed to improve fuel performance. The fuel rod cladding material is ZIRLO™, as is the material for the mid-grids, IFM grids, guide tubes, and instrument tubes. The burnable absorber types in use at CPNPP are the integral fuel burnable absorber (IFBA) pellets and the wet annular burnable absorber (WABA) rodlets. Information on the fuel design is provided in Section 2.8.1 of this Licensing Report (LR). With respect to the non-LOCA transient analyses, the effects of fuel design mechanical features were accounted for in fuel-related input assumptions, such as fuel and cladding dimensions, cladding material, fuel temperatures, and core bypass flow.

2.8.5.0.2 Peaking Factors, Kinetics Parameters

Relative to the fuel, the power distribution is characterized by a nuclear enthalpy rise hot channel factor (radial peaking factor, $F_{\Delta H}^N$) of 1.538 for analyses employing the Revised Thermal Design Procedure (RTDP) (Reference 1), and 1.600 for non-RTDP analyses, and a full-power heat flux hot channel factor (total peaking factor, F_Q) of 2.50. $F_{\Delta H}^N$ is important for transients that are analyzed for departure from nucleate boiling (DNB) concerns (Table 2.8.5.0-1 identifies which events are analyzed for DNB concerns, as well as the DNB methodology used (RTDP or non-RTDP)). As $F_{\Delta H}^N$ increases with decreasing power level, due to rod insertion, all transients analyzed for DNB concerns are assumed to begin with an $F_{\Delta H}^N$ consistent with the $F_{\Delta H}^N$ defined in the Core Operating Limits Report (COLR) identified in Technical Specifications 5.6.5 for the assumed nominal power level. The F_Q , for which the limits are specified in the COLR, is important for transients that are analyzed for overpower concerns, e.g., rod cluster control assembly (RCCA) ejection.

The minimum shutdown margin at hot zero power (HZP) conditions, with the most reactive RCCA fully withdrawn, is assumed to be 1.3-percent $\Delta k/k$. This was assumed in the HZP steam line break and HZP feedwater malfunction analyses.

2.8.5.0.3 SPU Program Features

Key features of the SPU program that were considered in the non-LOCA transient analyses are as follows:

- A nuclear steam supply system (NSSS) power level of 3,628 MWt (includes a net reactor coolant pump (RCP) heat of 16 MWt),

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- 17×17 VANTAGE+ fuel with a fuel rod outer diameter of 0.360 inches,
 - A nominal, full-power reactor coolant vessel average temperature (T_{avg}) window between 574.2° and 589.2°F was supported for most analyses. However, the minimum T_{avg} value is limited to 585.4°F for both units based on the analysis of the Inadvertent Actuation of the ECCS event discussed in Section 2.8.5.5.
 - A reactor coolant system (RCS) thermal design flow (TDF) of 382,800 gpm (95,700 gpm/loop), and a Technical Specifications minimum measured flow (MMF) of 396,400 gpm (99,100 gpm/loop). Although the Unit 2 Technical Specifications MMF is 408,000 gpm (102,000 gpm/loop), the more conservative Unit 1 value (396,400 gpm or 99,100 gpm/loop) was applied to both Units.
 - Westinghouse Model Δ 76 steam generators in Unit 1 and Westinghouse Model D-5 steam generators in Unit 2,
 - Uniform steam generator tube plugging (SGTP) levels of 0 and 10 percent,
 - A nominal operating pressurizer pressure of 2,250 psia,
 - A design core bypass flow of 5.8 percent (non-RTDP analyses) and a statistical core bypass flow of 4.3 percent (RTDP analyses), with core thimble plugs installed,
 - A nominal, full-power main feedwater temperature window between 390° and 450.3°F.

For most transients that were analyzed for departure from nucleate boiling (DNB) concerns, the RTDP methodology (Reference 1) was employed. With this methodology, nominal values are assumed for the initial reactor coolant system (RCS) conditions of power, temperature, pressure, and flow, and the corresponding uncertainty allowances are accounted for statistically in defining the departure from nucleate boiling ratio (DNBR) safety analysis limit. Note that the nominal RCS flow modeled in RTDP transient analyses is the minimum measured flow of 396,400 gpm.

As discussed in Licensing Report Section 2.8.3 uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions were combined statistically to obtain the overall DNB uncertainty factor, which was used to define the design-limit DNBR (1.23 for typical cell and 1.22 for thimble cell). In other words, the design limit DNBR is a DNBR value that is greater than the WRB-2 DNB correlation limit (1.17) by an amount that accounts for the RTDP uncertainties. To provide DNBR margin to offset various penalties such as those due to rod bow and instrument bias, and to provide flexibility in design and operation of the plant, the design limit DNBR was conservatively increased to a value designated as the safety analysis limit DNBR, to which transient-specific DNBR values were compared. The DNBR safety analysis limit selected for CPNPP is 1.61 for both typical and thimble cells.

For transient analyses that are not DNB-limited, or for which RTDP is not employed, the initial conditions were obtained by applying the maximum, steady-state uncertainties to the nominal values in the most conservative direction; this is known as Standard Thermal Design Procedure (STDP), or non-RTDP. In these analyses, the RCS flow was assumed to be equal to the TDF, and the following steady-state initial condition uncertainties were applied:

- ± 0.6 -percent NSSS power allowance for calorimetric measurement uncertainty,
- $\pm 6^\circ\text{F}$ T_{avg} allowance for deadband and system measurement uncertainties,
- ± 30 psi pressurizer pressure allowance for steady-state fluctuations and measurement uncertainties.

2.8.5.0.4 Other Major Assumptions

Table 2.8.5.0-2 lists the non-LOCA initial condition assumptions used. Other major assumptions considered in the non-LOCA transient analyses are discussed below:

- Staggered lift setpoints were modeled for the main steam safety valves (MSSVs) using plant-specific Technical Specification setpoints. Each MSSV was modeled with a +3-percent setpoint tolerance and a 5-psi ramp from closed to full-open, which accounts for accumulation. A 3-percent setpoint tolerance is also supported, but because none of the non-LOCA transients are limiting with minimum setpoints, it has not been explicitly modeled.
- +1%/-3% setpoint tolerance was considered in the modeling of the pressurizer safety valves (PSVs), which have a required nominal setpoint pressure of 2,460 psig. (See License Amendment Request 06-006, submitted by TXX-07108.) Additionally, when it was conservative to do so (that is, for peak RCS pressure concerns), the effects of the PSV water-filled loop seals, as discussed in Reference 2, were explicitly modeled.
- Consistent with the CPNPP Technical Specifications (Figure 3.1.3-1), for minimum reactivity feedback, a maximum moderator temperature coefficient (MTC) of $+0.5 \times 10^{-4} \Delta\text{k/k}/^\circ\text{F}$ (+5 pcm/ $^\circ\text{F}$) is applicable at power levels up to 70 percent. Between 70 percent and 100 percent power, the maximum MTC ramps linearly to 0 $\Delta\text{k/k}/^\circ\text{F}$ (0 pcm/ $^\circ\text{F}$). For maximum reactivity feedback, a maximum moderator density coefficient (MDC) of 0.50 $\Delta\text{k/g/cc}$ was assumed.
- The fission product contribution to decay heat assumed in the non-LOCA analyses is consistent with the American National Standards Institute/American Nuclear Society standard ANSI/ANS-5.1-1979 for decay heat power in light water reactors (Reference 3), including two standard deviations of uncertainty.

2.8.5.0.5 Overtemperature and Overpower Nitrogen-16 (N-16) Reactor Trip Setpoints

The overtemperature and overpower N-16 (OTN-16/OPN-16) reactor trip setpoints were recalculated based on the methodology described in WCAP-8745 (Reference 4). Conservative core thermal limits developed using the RTDP methodology were used to calculate the OTN-16 and OPN-16 reactor trip setpoints. The assumed core thermal limits are presented in Figure 2.8.5.0-1. The OTN-16 and OPN-16 trip setpoints for two sample pressures are illustrated in Figure 2.8.5.0-2 and presented in Table 2.8.5.0-3.

The adequacy of these setpoints was confirmed by showing that the DNB design basis is met in the analyses of those events that credit these functions for accident mitigation. The revised safety analysis setpoints are based upon the assumption that the reference cold leg temperature (T_c^0) used in the OTN-16 and OPN-16 setpoint equations is equal to the cold leg temperature (T_c) corresponding to nominal full power conditions.

The boundaries of operation defined by the OTN-16 and OPN-16 trips are represented as "protection lines" in Figure 2.8.5.0-2. The protection lines were drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, a trip would occur well within the area bounded by these lines. These protection lines are based upon the safety analysis limit OTN-16 and OPN-16 setpoint values, which are essentially the Technical Specification nominal values with allowances for instrumentation errors and acceptable drift between instrument calibrations. The diagram is useful in the fact that the limit imposed by any given DNBR can be represented as a line (T_{inlet} versus Power). The DNB lines represent the locus of conditions for which the DNBR equals the limit value (1.61 for both typical and thimble cells). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit DNBR value.

The area of permissible operation (power, temperature, and pressure) is bounded by the combination of the high neutron flux (fixed setpoint) reactor trip, the high- and low-pressurizer pressure reactor trips (fixed setpoints), the OTN-16 (variable setpoint) and OPN-16 (fixed setpoint) reactor trips, and the opening of the MSSVs, which limits the maximum RCS average temperature. The adequacy of the OTN-16 and OPN-16 setpoints was confirmed by demonstrating that the DNB design basis was met for those transients that credit these protection functions.

The resistance temperature detectors (RTDs) that measure the cold leg temperatures used to determine the T_c input of the OTN-16 setpoint equation are scaled with a range of 510° to 630°F. It was confirmed that this range bounds the T_c range that is required to be protected by the OTN-16 reactor protection function.

2.8.5.0.6 RTS and ESFAS Functions Assumed in Analyses

Table 2.8.5.0-4 contains a list of the different reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) functions credited in the non-LOCA transient analyses. The safety analysis setpoints and associated time delays of each function are also presented in Table 2.8.5.0-4.

2.8.5.0.7 RCCA Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to the non-LOCA transient analyses, the critical parameter is the time from the start of RCCA insertion to when the RCCAs reach the dashpot region, which is located at an insertion point corresponding to approximately 86 percent of the total RCCA travel distance. For the non-LOCA analyses, the RCCA insertion time from fully withdrawn to dashpot entry was modeled as 2.7 seconds. The assumed negative reactivity insertion following reactor trip is based on having the most reactive RCCA stuck in the fully withdrawn position:

Three figures relating to RCCA drop time and reactivity worth are presented in this report. The RCCA position (fraction of full insertion) versus the time from release is presented in Figure 2.8.5.0-3. The normalized reactivity worth assumed in the safety analyses is shown in Figure 2.8.5.0-4 as a function of rod insertion fraction and in Figure 2.8.5.0-5 as a function of time.

2.8.5.0.8 Reactivity Coefficients

The transient response of the reactor core is dependent on reactivity feedback effects, in particular the MTC and the Doppler power coefficient (DPC). Depending upon event-specific characteristics, conservatism dictates the use of either maximum or minimum reactivity coefficient values. Justification for the use of the reactivity coefficient values was treated on an event-specific basis. Table 2.8.5.0-5 presents the core kinetics parameters and reactivity feedback coefficients assumed in the non-LOCA analyses.

The maximum and minimum integrated DPCs assumed in the safety analyses are provided in Figure 2.8.5.0-6. Note that the HZP steam line break core response analysis used a different DPC, which was based on an RCCA being stuck out of the core (not shown in Figure 2.8.5.0-6).

2.8.5.0.9 Computer Codes Utilized

Summary descriptions of the principal computer codes used in the non-LOCA transient analyses are provided below. Table 2.8.5.0-6 lists the computer codes used in each of the non-LOCA analyses.

FACTRAN

FACTRAN 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 of pressure, flow, temperature, and density. The code uses a fuel model with the following features:

- A sufficiently large number of radial space increments to handle fast transients such as an RCCA ejection accident,

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- Material properties that are functions of temperature,
 - A sophisticated fuel-to-cladding gap heat transfer calculation, and
 - Calculations to address post-DNB conditions (film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel).

The FACTRAN licensing topical report, WCAP-7908 (Reference 5), was approved by the NRC via a Safety Evaluation Report (SER) from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse), dated September 30, 1986. This SER issued for FACTRAN identifies seven conditions of acceptance, which are summarized below along with justifications for application to CPNPP.

1. "The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."

Justification

The FACTRAN code was used in the analyses of the following transients for CPNPP: Uncontrolled RCCA Withdrawal from a Subcritical Condition (FSAR 15.4.1) and RCCA Ejection (FSAR 15.4.8). Initial fuel temperatures used as FACTRAN input in the RCCA Ejection analysis were calculated using the NRC-approved PAD 4.0 computer code as described in WCAP-15063 Revision 1 (Reference 6). As indicated in WCAP-15063 Revision 1, the NRC has approved the method of determining uncertainties for PAD 4.0 fuel temperatures.

2. "Table 2 presents the guidelines used to select initial temperatures."

Justification

In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: Loss of Flow, Locked Rotor, and Rod Ejection. As discussed above, fuel temperatures were used as input to the FACTRAN code in the RCCA Ejection analysis for CPNPP. The assumed fuel temperatures, which were calculated using the PAD 4.0 computer code (Reference 6), include uncertainties and are conservatively high. FACTRAN was not used in the Loss of Flow and Locked Rotor analyses.

3. "The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."

Justification

The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the RCCA Withdrawal from a Subcritical Condition transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for DNB concerns)

and a low constant value is assumed to maximize fuel temperatures. For the RCCA Ejection transient, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate clad collapse onto the fuel pellet.

4. "...the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."

Justification

Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed for CPNPP (RCCA Withdrawal from a Subcritical Condition (FSAR 15.4.1) and RCCA Ejection (FSAR 15.4.8)). Therefore, additional justification is not required.

5. "The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."

Justification

At least 6 concentric rings were assumed in FACTRAN for each transient analyzed for CPNPP (RCCA Withdrawal from a Subcritical Condition (FSAR 15.4.1) and RCCA Ejection (FSAR 15.4.8)).

6. "Although time-independent mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-dependent mechanical behavior (e.g., plastic deformation) is not considered in the code. ...for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet."

Justification

The two transients that were analyzed with FACTRAN for CPNPP (RCCA Withdrawal from a Subcritical Condition (FSAR 15.4.1) and RCCA Ejection (FSAR 15.4.8)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the RCCA Withdrawal from a Subcritical Condition transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the RCCA Ejection transient, a high gap heat

transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.

7. “The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When $(T(\text{centerline}) - T(\text{surface}))$ is on the order of 3,000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable.”

Justification

The condition of concern $(T(\text{centerline}) - T(\text{surface}))$ on the order of 3,000°F is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA ejection transient for CPNPP.

RETRAN

RETRAN is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. This code simulates a multi-loop system by a lumped parameter model containing the reactor vessel, hot- and cold-leg piping, RCPs, steam generators (tube and shell sides), main steam lines, and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves can also be modeled. RETRAN includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a detailed nodalization for the thermal transients. The RTS simulated in the code includes reactor trips on high neutron flux, high neutron flux rate, OTN-16, OPN-16, low reactor coolant flow, high- and low-pressurizer pressure, high pressurizer level, and low-low steam generator water level. Control systems are also simulated including rod control and pressurizer pressure control. Parts of the safety injection system (SIS), including the accumulators, are also modeled. Also, a conservative approximation of the transient DNBR, based on the core thermal limits, is calculated via RETRAN.

The RETRAN licensing topical report, WCAP-14882 (Reference 7), was approved by the NRC via an SER from F. Akstulewicz (NRC) to H. Sepp (Westinghouse), dated February 11, 1999. This SER issued for RETRAN identifies three conditions of acceptance, which are summarized below along with justifications for application to CPNPP.

1. "The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."

Justification

The transients listed in Table 1 of the SER are:

- Feedwater system malfunctions,
- Excessive increase in steam flow,
- Inadvertent opening of a steam generator relief or safety valve,
- Steam line break,
- Loss of external load/turbine trip,
- Loss of offsite power,
- Loss of normal feedwater flow,
- Feedwater line rupture,
- Loss of forced reactor coolant flow,
- Locked reactor coolant pump rotor/sheared shaft,
- Control rod cluster withdrawal at power,
- Dropped control rod cluster/dropped control bank,
- Inadvertent increase in coolant inventory,
- Inadvertent opening of a pressurizer relief or safety valve,
- Steam generator tube rupture.

The transients explicitly analyzed for CPNPP using RETRAN are:

- Feedwater system malfunctions (FSAR 15.1.1 and 15.1.2),
- Steam line break (FSAR 15.1.5),
- Loss of external load/turbine trip (FSAR 15.2.2, 15.2.3),
- Loss of non-emergency AC power (loss of offsite power) (FSAR 15.2.6),
- Loss of normal feedwater flow (FSAR 15.2.7),
- Feedwater system pipe break (feedwater line rupture) (FSAR 15.2.8),
- Loss of forced reactor coolant flow (FSAR 15.3.1, 15.3.2),
- Locked reactor coolant pump rotor/shaft break (FSAR 15.3.3, 15.3.4),
- Uncontrolled RCCA withdrawal at power (FSAR 15.4.2),
- Inadvertent operation of the ECCS (increase in coolant inventory) (FSAR 15.5.1),
- Inadvertent opening of a pressurizer safety or relief valve (FSAR 15.6.1),

As each transient analyzed for CPNPP using RETRAN matches one of the transients listed in Table 1 of the SER, additional justification is not required.

-
2. "WCAP-14882 describes modeling of Westinghouse designed 4-, 3-, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."

Justification

The CPNPP consists of two 4-loop Westinghouse-designed units that were "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required.

3. "Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 (WCAP-9272) (Reference 8 in this document). Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."

Justification

The input data used in the RETRAN analyses performed by Westinghouse came from both Luminant Power and Westinghouse sources. Assurance that the RETRAN input data is conservative for CPNPP is provided via Westinghouse's use of transient-specific analysis guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results. Based on the analysis guidance documents, conservative plant-specific input values were requested and collected from the responsible Luminant Power and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272 (Reference 8), the safety analysis input values used in the CPNPP analyses were selected to conservatively bound the values expected in subsequent operating cycles.

Note that since CPNPP has OTN-16 and OPN-16 reactor trips rather than overtemperature and overpower ΔT , an N-16 model, which is equivalent to that currently being used by Luminant Power in support of CPNPP, was applied.

The RETRAN nodalization modeling used in the CPNPP analyses is consistent with the Westinghouse 4-loop plant nodalization model of WCAP-14882, except for the preheat steam generator model used for CPNPP Unit 2 (see discussion below) and the nodalization of the hot legs. Since the approval of WCAP-14882, the hot leg modeling was enhanced to minimize code instabilities attributed to pressurizer insurge and outsurge. This hot leg model enhancement, which has been applied in other RETRAN analyses performed by Westinghouse, consisted of dividing each hot leg control volume into three equal control volumes. Although it was needed only for the hot leg connected to the pressurizer, all loops were divided in the same manner.

Preheat Steam Generator Model for CPNPP Unit 2

As noted in WCAP-14882, multi-node RETRAN models were developed for Westinghouse feeding and preheat steam generators. The nodalization of the feeding and preheat steam generators explicitly models the circulation loop and allows for the calculation of the indicated water level. The multi-node secondary model also allows for a detailed transient response during the steam line break and other secondary side transients.

Figure 3.6-1 of WCAP-14882 (Reference 7) presents the nodalization of a preheat steam generator model that was based on the Westinghouse Model E steam generator design. The Comanche Peak Unit 2 Westinghouse Model D-5 steam generator design includes variations in the preheater and feedwater designs when compared to the Model E steam generator, and thus the RETRAN model nodalization was revised. Figure 2.8.5.0-7 provides the revised nodalization used for the Model D-5 steam generator. The Model D-5 steam generator features a split feedwater flow injection design, which, at full power, has approximately 80 percent of the flow entering the preheater region and 20 percent of the flow entering the downcomer region below the water level via the auxiliary feedwater nozzle when operating at full power.

- The feedwater entering the preheater enters a region of the cold side of the tube bundle (Volume x60) and then splits into two streams. Most of this feedwater flow passes up through the upper sections of the preheater (Volumes x61 and x62) and joins the flow rising in the hot tube side of the tube bundle in Volume x72. The remainder of the feedwater entering the preheater travels down to the lower section of the preheater and joins with recirculated water before passing to the hot side of the tube bundle (Volume x71).
- The feedwater entering the feeding in the upper downcomer region (Volume x77) enters and mixes with the recirculated liquid from Volume x76 and flows to the lower downcomer (Volume x78) where it enters the tube bundle on the cold side (Volume x69) or hot side (Volume x71).

The CPNPP Model D-5 tube bundle region above the preheater is modeled as 3 sequential secondary side nodes (Volumes x72, x73, and x74) and combines the flows from both the hot tube side (Volumes x70 and x71) and the flow exiting from the top of the preheater. This is similar to the Model E steam generator design documented in WCAP-14882. Two-phase fluid exits the tube bundle and flows through the riser (Volume x75) and enters Volume x76 where phase separation is modeled using the RETRAN bubble rise model. The phase separation in Volume x76 simulates the steam generator moisture separators, i.e., the swirl vane and demister vane separators. The steam fraction rises to the top of Volume x76 and exits the steam generator through the steam nozzle while the liquid fraction mixes with feedwater and continues the cycle by mixing with the feedwater nozzle flow in Volume x77.

On the primary side of the Unit 2 Model D-5 steam generator, primary coolant passes from the hot leg (Volume x03) to the steam generator inlet plenum (Volume x20) through the U-tubes (Volumes x21 through x32), into the steam generator outlet plenum (Volume x40), and exits the steam generator into the cold leg (Volume x09). Twelve conducting heat exchangers transfer heat from the primary to the secondary during normal operation and transient conditions.

This D-5 model has been benchmarked against the Westinghouse preheat steam generator design code, including the primary and secondary side volumes, primary side pressure drops for 0 percent and maximum tube plugging levels, secondary side pressure drops, heat transfer characteristics for 0 percent and maximum tube plugging levels, and steam generator masses versus both power and water levels.

LOFTRAN

Transient response studies of a PWR to specified perturbations in process parameters use the LOFTRAN computer code. This code simulates a multi-loop system by a model containing the reactor vessel, hot- and cold-leg piping, steam generators (tube and shell sides), the pressurizer and the pressurizer heaters, spray, relief valves, and safety valves. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients. The code simulates the RPS, which includes reactor trips on high neutron flux, OTN-16 and OPN-16, high- and low-pressurizer pressure, low RCS flow, low-low steam generator water level, and high pressurizer level. Control systems are also simulated including rod control, steam dump, and pressurizer pressure control. The SIS, including the accumulators, is also modeled. LOFTRAN can also approximate the transient value of DNBR based on input from the core thermal safety limits.

The LOFTRAN licensing topical report, WCAP-7907 (Reference 9), was approved by the NRC via an SER from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), dated July 29, 1983. This SER for LOFTRAN identifies one condition of acceptance, which is summarized below along with justification for application to CPNPP.

1. "LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are:
 - Feedwater System Malfunction
 - Excessive Increase in Steam Flow
 - Inadvertent Opening of a Steam Generator Relief or Safety Valve
 - Steamline Break
 - Loss of External Load
 - Loss of Offsite Power
 - Loss of Normal Feedwater
 - Feedwater Line Rupture

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- Loss of Forced Reactor Coolant Flow
 - Locked Pump Rotor
 - Rod Withdrawal at Power
 - Rod Drop
 - Startup of an Inactive Pump
 - Inadvertent ECCS Actuation
 - Inadvertent Opening of a Pressurizer Relief or Safety Valve

This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture...”

Justification

For CPNPP, the LOFTRAN code was only used in the analysis of the dropped rod transient (FSAR 15.4.3) and in the analysis of the anticipated transients without scram (FSAR 15.8). As these transients match one of the transients listed in the SER, additional justification is not required.

TWINKLE

TWINKLE is a multi-dimensional spatial neutron kinetics code. 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 multi-region fuel-cladding-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 8,000 spatial points and performs 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. The code provides various outputs, such as channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

The TWINKLE licensing topical report, WCAP-7979 (Reference 10), was approved by the U.S. Atomic Energy Commission (AEC) via an SER from D. B. Vassallo (AEC) to R. Salvatori (Westinghouse), dated July 29, 1974. This SER for TWINKLE does not identify any conditions, restrictions, or limitations that need to be addressed for application to CPNPP.

Advanced Nodal Code (ANC)

ANC is an advanced nodal code capable of two-dimensional (2-D) and three-dimensional (3-D) neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, 3-D ANC validates 1-D and 2-D results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

The ANC licensing topical report, WCAP-10965 (Reference 11), was approved by the NRC via an SER from C. Berlinger (NRC) to E. P. Rahe (Westinghouse), dated June 23, 1986. This SER for ANC does not identify any conditions, restrictions, or limitations that need to be addressed for application to CPNPP.

VIPRE

The VIPRE computer program performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core.

The VIPRE licensing topical report, WCAP-14565 (Reference 12), was approved by the NRC via an SER from T. H. Essig (NRC) to H. Sepp (Westinghouse), dated January 19, 1999. This SER for VIPRE identifies four conditions of acceptance, which are summarized below along with justification for application to CPNPP.

1. "Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."

Justification

The WRB-2 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the CPNPP 17x17 VANTAGE+ fuel type. The use of the WRB-2 DNB correlation for VANTAGE+ fuel was approved September 1985 (Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-10444, VANTAGE+ Fuel Assembly," Reference 19). WCAP-12610 extended the use of the WRB-2 correlation to VANTAGE+ fuel and was approved July 1, 1991 (Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610 VANTAGE+ Fuel Assembly Reference Core Report," Reference 20).

The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the CPNPP VANTAGE+ fuel were justified using the same methodologies as for previously approved safety evaluations of other Westinghouse four-loop plants using the same fuel design.

2. "Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."

Justification

The core boundary conditions for the VIPRE calculations for the CPNPP fuel are all generated from NRC-approved codes and analysis methodologies. Conservative

reactor core boundary conditions were justified for use as input to VIPRE. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272 (Reference 8).

3. "The NRC Staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."

Justification

As discussed in response to Condition 1, the WRB-2 correlation with a limit of 1.17 was used for the DNB analyses of the CPNPP fuel. For conditions where WRB-2 is not applicable, the W-3 DNB correlation was used with a limit of 1.30 (1.45, for pressures between 500 psia and 1,000 psia).

4. "Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."

Justification

For application to CPNPP safety analysis, the usage of VIPRE in the post-critical heat flux region is limited to the peak clad temperature calculation for the locked rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565 and included the following conservative assumptions:

- DNB was assumed to occur at the beginning of the transient
- Film boiling was calculated using the Bishop-Sandberg-Tong correlation
- The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction

Conservative results were further ensured with the following input:

- Fuel rod input based on the maximum fuel temperature at the given power
- The hot spot power factor was equal to or greater than the design linear heat rate

Uncertainties were applied to the initial operating conditions in the limiting direction.

2.8.5.0.10 Classification of Events

Each of the non-LOCA events listed in Table 2.8.5.0-7 is presented in Section 15 of the FSAR (Reference 13). Each non-LOCA event is categorized with respect to its potential consequences. Since 1970, the classification of plant conditions in American Nuclear Society Standard ANSI N18.2-1973 (Reference 14) has often been used to facilitate the evaluation of nuclear plant safety and the functional requirements for structures, systems, and components. The plant conditions are divided into four categories in accordance with the anticipated frequencies of occurrence and potential radiological consequences. The four categories (or conditions) are:

- Condition I – Normal Operation
- Condition II – Faults of Moderate Frequency
- Condition III – Infrequent Faults
- Condition IV – Limiting Faults

The basic principle applied in relating requirements to each of the conditions is that the more probable occurrences must result in little or no risk to the public, and those extreme situations having the potential for greater risk should be those situations least likely to occur. Where applicable, the reactor trip system and/or engineered safety features are assumed in fulfilling this principle. Each condition is described in more detail as follows.

Condition I – Normal Operation

Condition I occurrences are those that are expected frequently or regularly during power operation, refueling, maintenance, or maneuvering of the plant. Condition I occurrences are accommodated with margin between any plant parameter and the value of the parameter that would require either automatic or manual protective action. In this regard, analysis of the fault condition is typically based on a conservative set of initial conditions corresponding to the most adverse set of conditions occurring during Condition I operation.

Condition II – Faults of Moderate Frequency

These faults occur with moderate frequency during the life of the plant, any one of which may occur during a calendar year (i.e., between 1/year and 1×10^{-1} /year). These faults, at worst, result in a reactor trip with the plant being capable of returning to operation after corrective action. Any release of radioactive materials in effluents to unrestricted areas should be in conformance with Title 10 Part 20 of the Code of Federal Regulations (10 CFR 20). A Condition II fault (or event), by itself, does not propagate to a more serious incident of the

Condition III or Condition IV type without the occurrence of other independent incidents. A single Condition II incident should not cause the loss of any barrier to the escape of radioactive products.

Condition III – Infrequent Faults

Condition III faults occur very infrequently during the life of the plant, any one of which may occur during the plant's lifetime (i.e., between 1×10^{-1} /year and 1×10^{-2} /year). Condition III faults can be accommodated with the failure of only a small fraction of the fuel rods, although sufficient fuel damage might occur to preclude resumption of operation for a considerable outage time. The release of radioactivity due to Condition III faults may exceed the guidelines of 10 CFR 20, but is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary. A Condition III fault does not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or containment barriers.

Condition IV – Limiting Faults

Condition IV occurrences are faults that are not expected to occur (i.e., $< 1 \times 10^{-2}$ /year), but are postulated because their consequences have the potential for the release of significant amounts of radioactive material. Condition IV faults are the most drastic occurrences that must be designed against, and represent the limiting design cases. Condition IV faults should not cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of the guideline values in Title 10 Part 100 of the Code of Federal Regulations (10 CFR 100). A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the RCS and the reactor containment.

2.8.5.0.11 Events Evaluated or Analyzed

Each of the FSAR transients listed in Table 2.8.5.0-1 were evaluated or analyzed as shown in Table 2.8.5.0-7 in support of the CPNPP Uprate Program. These transient evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria are satisfied for CPNPP. Table 2.8.5.0-1 summarizes the results obtained for each of the non-LOCA transient analyses.

2.8.5.0.12 Analysis Methodology

The transient-specific analysis methodologies that were applied to CPNPP have been reviewed and approved by the NRC via transient-specific topical reports (e.g., WCAPs) and/or through the review and approval of plant-specific safety analysis reports. There are four non-LOCA transients analyzed for CPNPP that have an approved transient-specific topical report: steam line break (FSAR Section 15.1.5), dropped rod (FSAR Section 15.4.3), boron dilution (FSAR Section 15.4.6), and RCCA ejection (FSAR Section 15.4.8).

Steam Line Break Methodology

The steam line break licensing topical report, WCAP-9226 Revision 1 (Reference 21), was approved by the NRC via an SER from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), dated January 31, 1989. The steam line break SER identifies two conditions of acceptance, which are summarized below along with justification for application to CPNPP.

1. "Only those codes which have been accepted by the NRC should be used for licensing application."

Justification

As identified in Table 2.8.5.0-6, the computer codes used in the analysis of the steam line break event are RETRAN, ANC, and VIPRE. Per Section 2.8.5.0.9, these codes have been accepted by the NRC, and therefore this condition of acceptance is satisfied for CPNPP.

2. "For the pressure between 500 and 1,000 psia, the 95/95 DNBR limit for the W-3 correlation is 1.45."

Justification

As shown in Table 2.8.5.0-1, 1.45 was applied as the DNBR limit in the steam line break analysis that used the W-3 DNB correlation. Thus, no further justification is required for CPNPP.

Dropped Rod Methodology

The dropped rod licensing topical report, WCAP-11394 (Reference 15), was approved by the NRC via an SER from A. C. Thadani (NRC) to R. A. Newton (Westinghouse Owners Group), dated October 23, 1989. The dropped rod SER identifies one condition of acceptance, which is summarized below along with justification for application to CPNPP.

1. "The Westinghouse analysis, results and comparisons are reactor and cycle specific. No credit is taken for any direct reactor trip due to dropped RCCA(s). Also, the analysis assumes no automatic power reduction features are actuated by the dropped RCCA(s). A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results."

Justification

For the reference cycle assumed in the CPNPP Uprate Program, the methodology described in WCAP-11394 was applied and the required comparisons have been made with acceptable results (DNB limits were not exceeded). Future cycles will be assessed as part of the reload safety evaluation process described in Reference 8.

Boron Dilution Methodology

The boron dilution in modes 3, 4, and 5 licensing topical report, RXE-94-001-A (Reference 22), was approved by the NRC specifically for Comanche Peak via an SER from T. A. Bergman (NRC) to W. J. Cahill, Jr. (TU Electric), dated November 3, 1993. The SER does not identify any conditions, restrictions, or limitations that need to be addressed for application to CPNPP; note that the method of RXE-94-001-A was applied in support of the current licensing basis for CPNPP.

RCCA Ejection Methodology

The RCCA ejection licensing topical report, WCAP-7588 Rev. 1-A (Reference 16), was approved by the AEC via an SER from D. B. Vassallo (AEC) to R. Salvatori (Westinghouse), dated August 28, 1973. The RCCA ejection SER identifies two conditions of acceptance, which are summarized below along with justification for application to CPNPP.

1. "The staff position, as well as that of the reactor vendors over the last several years, has been to limit the average fuel pellet enthalpy at the hot spot following a rod ejection accident to 280 cal/gm. This was based primarily on the results of the SPERT tests which showed that, in general, fuel failure consequences for UO₂ have been insignificant below 300 cal/gm for both irradiated and unirradiated fuel rods as far as rapid fragmentation and dispersal of fuel and cladding into the coolant are concerned. In this report, Westinghouse has decreased their limiting fuel failure criterion from 280 cal/gm (somewhat less than the threshold of significant conversion of the fuel thermal energy to mechanical energy) to 225 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods. Since this is a conservative revision on the side of safety, the staff concludes that it is an acceptable fuel failure criterion."

Justification

The maximum fuel pellet enthalpy at the hot spot calculated for each CPNPP-specific RCCA ejection case was less than 200 cal/gm. These results satisfy the fuel failure criterion accepted by the NRC staff.

2. "Westinghouse proposes a clad temperature limitation of 2,700°F as the temperature above which clad embrittlement may be expected. Although this is several hundred degrees above the maximum clad temperature limitation imposed in the AEC ECCS Interim Acceptance Criteria, this is felt to be adequate in view of the relatively short time at temperature and the highly localized effect of a reactivity transient."

Justification

As discussed in Westinghouse letter NS-NRC-89-3466 written to the NRC (Reference 17), the 2,700°F clad temperature limit was historically applied by Westinghouse to demonstrate that the core remains in a coolable geometry during an RCCA ejection transient. This limit was never used to demonstrate compliance with fuel

failure limits and is no longer used to demonstrate core coolability. The RCCA ejection acceptance criteria applied by Westinghouse to demonstrate long-term core coolability and compliance with applicable offsite dose requirements are identified in Licensing Report Section 2.8.5.4.6.

2.8.5.0.13 Operator Actions

The feedwater system pipe break and inadvertent operation of the ECCS events are the only events for which operator action is credited in the analyses. Licensing Report Sections 2.8.5.2.4 and 2.8.5.5 discuss the details of the feedwater system pipe break and inadvertent operation of the ECCS analyses.

2.8.5.0.14 References

1. WCAP-11397, "Revised Thermal Design Procedure," April 1989.
2. WCAP-12910 Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift," May 1993.
3. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power In Light Water Reactors," August 29, 1979.
4. WCAP-8745, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.
5. WCAP-7908, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO₂ Fuel Rod," December 1989.
6. WCAP-15063 Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000.
7. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
8. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
9. WCAP-7907, "LOFTRAN Code Description," April 1984.
10. WCAP-7979, "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," January 1975.
11. WCAP-10965, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.
12. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

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13. CPNPP FSAR, Amendment No. 101, February 1, 2007.
 14. ANS N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary PWRs, American Nuclear Society."
 15. WCAP-11394, "Methodology for the Analysis of the Dropped Rod Event," January 1990.
 16. WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.
 17. NS-NRC-89-3466, Letter from W. J. Johnson (Westinghouse) to R. C. Jones (NRC), dated October 23, 1989, "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents."
 18. NS-TMA-2182, Letter from T. M. Anderson (Westinghouse) to Dr. S. H. Hanauer (NRC), dated December 30, 1979, "Anticipated Transients Without Scram for Westinghouse Plants."
 19. WCAP-10444, "Reference Core Report VANTAGE+ Fuel Assembly," September 1985.
 20. WCAP-12610, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.
 21. WCAP-9226 Revision 1, "Reactor Core Response to Excessive Secondary Steam Releases," February 1998.
 22. RXE-94-001-A, "Safety Analysis of the Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5," February 1994.

Table 2.8.5.0-1				
Non-LOCA Analysis Limits and Analysis Results				
FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.1.1	Decrease in Feedwater Temperature	Minimum DNBR (RTDP, WRB-2)	1.61	1.90
15.1.2	Increase in Feedwater Flow	Minimum DNBR (RTDP, WRB-2) (HFP) Minimum DNBR (non-RTDP, W-3) (HZP)	1.61 (HFP) 1.45 (HZP)	2.10 (HFP) ⁽¹⁾ (HZP)
15.1.3	Excessive Increase in Secondary Steam Flow	Minimum DNBR (RTDP, WRB-2)	1.61	> 1.61
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Bounded by Steam Line Break (FSAR Section 15.1.5)	N/A	N/A
15.1.5	Steam System Piping Failure – Zero Power (Core response only)	Minimum DNBR (non-RTDP, W-3) (typical/thimble)	1.45/1.45	3.067/2.861
	Steam System Piping Failure – Full Power (Core response only)	Minimum DNBR (RTDP, WRB-2 correlation) (typical/thimble)	1.61/1.61	2.015/1.963
		Peak Linear Heat Generation (kW/ft)	22.4 ⁽²⁾	21.6
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	There are no steam pressure regulators at CPNPP whose failure or malfunction could cause a steam flow transient (FSAR Section 15.2.1)	N/A	N/A
15.2.2	Loss of External Electrical Load	Bounded by Turbine Trip (FSAR Section 15.2.3)	N/A	N/A
15.2.3	Turbine Trip	Minimum DNBR (RTDP, WRB-2)	1.61	1.98
		Peak RCS Pressure, psia	2,748.2	2,746.0
		Peak MSS Pressure, psia	1,318.2	1,298.4
15.2.4	Inadvertent Closure of Main Steam Isolation Valves	Bounded by Turbine Trip (FSAR Section 15.2.3)	N/A	N/A

Table 2.8.5.0-1 (cont.)

Non-LOCA Analysis Limits and Analysis Results

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.2.5	Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	Bounded by Turbine Trip (FSAR Section 15.2.3)	N/A	N/A
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	Maximum pressurizer mixture volume, ft ³	1,800	1,600.4
15.2.7	Loss of Normal Feedwater	Maximum Pressurizer Mixture Volume, ft ³	1,800	1,747.9
15.2.8	Feedwater System Pipe Break	Minimum Margin to Hot Leg Saturation, °F	>0.0	10
15.3.1	Partial Loss of Forced Reactor Coolant Flow	Minimum DNBR (RTDP, WRB-2) (typical/thimble)	1.61/1.61	2.253/2.173
15.3.2	Complete Loss of Forced Reactor Coolant Flow	Minimum DNBR (RTDP, WRB-2) (typical/thimble)	1.61/1.61	1.940/1.901
15.3.3/ 15.3.4	Reactor Coolant Pump Shaft Seizure (Locked Rotor)/Shaft Break	Peak RCS Pressure, psia	2,748.2	2,574.5
		Peak Cladding Temperature, °F	2,700	1,723.6
		Maximum Zirconium-Water Reaction, %	16	0.22
		Maximum Percentage of Rods-in-DNB, %	10	<10
15.4.1	Uncontrolled RCCA Withdrawal from a Subcritical or Low Power Condition	Minimum DNBR Below First Mixing Vane Grid (non-RTDP, W-3 correlation) (typical/thimble)	1.30/1.30	1.824/1.616
		Minimum DNBR Above First Mixing Vane Grid (non-RTDP, WRB-2 correlation) (typical/thimble)	1.17/1.17	2.018/1.997
		Maximum Fuel Centerline Temperature, °F	4,800 ⁽³⁾	2,304
15.4.2	Uncontrolled RCCA Withdrawal at Power	Minimum DNBR (RTDP, WRB-2)	1.61	1.689
		Peak MSS Pressure, psia	1,318.2	1,275.8

Table 2.8.5.0-1 (cont.)
Non-LOCA Analysis Limits and Analysis Results

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.4.3	RCCA Misalignment (Dropped Rod)	Minimum DNBR (RTDP, WRB-2)	1.61	> 1.61
		Peak Linear Heat Generation (kW/ft)	22.4 ⁽²⁾	< 22.4
		Peak Uniform Cladding Strain (%)	1.0	< 1.0
15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	No Analysis Performed (See Licensing Report Section 2.8.5.4.4)	N/A	N/A
15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	This event is not applicable to CPNPP.	N/A	N/A
15.4.6	Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)	Minimum Time from Alarm to Operator Action to prevent a Complete Loss of Shutdown Margin, Minutes	15	48.0(Mode 1 manual)
				49.8 (Mode 1 auto)
				52.5 (Mode 2)
				The maximum critical boron concentration is controlled as a function of the plant initial boron concentration to meet a minimum operator action time of 15 minutes. (Modes 3, 4 and 5)

Table 2.8.5.0-1 (cont.)
Non-LOCA Analysis Limits and Analysis Results

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.4.8	Spectrum of RCCA Ejection Accidents	Maximum Fuel Pellet Average Enthalpy, cal/g	200	114.3 (BOC-HZP) ⁽⁴⁾ 161.6 (BOC-HFP) ⁽⁵⁾ 138.9 (EOC-HZP) ⁽⁶⁾ 157.5 (EOC-HFP) ⁽⁷⁾
		Maximum Fuel Melt, %	10 ⁽⁸⁾	0.00 (BOC-HZP) ⁽⁴⁾ 0.04 (BOC-HFP) ⁽⁵⁾ 0.00 (EOC-HZP) ⁽⁶⁾ 0.23 (EOC-HFP) ⁽⁷⁾
		Peak RCS Pressure, psia	Generically addressed in Reference 16	
15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	Maximum pressurizer mixture volume, ft ³	1,800	1,780.0
15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Event is covered by the analyses of the Boron Dilution event (FSAR Section 15.4.6) and the Inadvertent Operation of the Emergency Core Cooling System During Power Operation event (FSAR Section 15.5.1).	N/A	N/A
15.5.3	A Number of BWR Transients	These events are not applicable to CPNPP.	N/A	N/A
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Minimum DNBR (RTDP, WRB-2)	1.61	1.9
15.8	ATWS	Peak RCS Pressure, psig	3,200	<3,200

Table 2.8.5.0-1 (cont.)

Non-LOCA Analysis Limits and Analysis Results

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
Notes:				
1. Bounded by zero power steam system piping failure.				
2. Corresponds to a conservative UO ₂ fuel melting temperature of 4,700°F.				
3. 4,800°F is the fuel melting temperature corresponding to a maximum UO ₂ burnup at the hot spot of ~48,276 MWd/MTU.				
4. BOC-HZP ≡ Beginning of cycle HZP.				
5. BOC-HFP ≡ Beginning of cycle HFP.				
6. EOC-HZP ≡ End of cycle HZP.				
7. EOC-HFP ≡ End of cycle HFP.				
8. BOC and EOC fuel melting temperatures are 4,900 and 4,800°F, respectively. These temperatures correspond to hot spot burnups of approximately 31,034 MWD/MTU (BOC) and 48,276 MWD/MTU (EOC).				

Table 2.8.5.0-2

Non-LOCA Plant Initial Condition Assumptions

Parameter	RTDP	Non-RTDP	Notes
NSSS Power (MWt)	3,628.0	3,628.0 * 1.006	1
Nominal Total Net RCP Heat (MWt)	16.0	16.0	1, 2, 3
Maximum Full-Power Vessel T _{avg} (°F)	589.2	589.2 ± 6.0	1, 4
Minimum Full-Power Vessel T _{avg} (°F)	574.2	574.2 ± 6.0	1, 4, 8
No-Load RCS Temperature (°F)	557.0	557.0	1, 4
Pressurizer Pressure (psia)	2,250	2,250 ± 30	1
Steam Flow (lbm/hr)	see Note 5	see Note 5	--
Steam Pressure (psia)	see Note 5	see Note 5	--
Full-Power Feedwater Temperature Range (°F)	390-450.3	390-450.3	1
Pressurizer Water Level (% span)	see Note 6	see Note 6	--
Steam Generator Water Level (% NRS)	see Note 7	see Note 7	--

Notes:

1. See Section 1.1 of Licensing Report.
2. Total RCP heat input minus RCS thermal losses.
3. A maximum net RCP heat of 20 MWt was conservatively assumed in some non-RTDP analyses, e.g., loss of normal feedwater and feedline break events.
4. All analyses assumed a programmed no-load T_{avg} of 557°F. For the events initiated from a no-load condition (rod withdrawal from subcritical, steam line break, rod ejection, boron dilution), the use of the no-load temperature as the initial temperature bounded the case of startup operations at Comanche Peak with a temperature less than 557°F.
5. The nominal steam flow rate and steam pressure are dependent on other nominal conditions.
6. The nominal/programmed pressurizer water level varies linearly from 25% of span at the no-load T_{avg} of 557°F to either 43.4% of span at the minimum full-power T_{avg} of 574.2°F or 60% of span at full-power T_{avg} values greater than or equal to 584.7°F. The programmed level is constant at the full-power T_{avg} level for T_{avg} values greater than the full-power T_{avg}. An uncertainty of ±5% of span was applied when conservative.
7. The programmed steam generator water level modeled in the Unit 1 analyses was a constant 67% narrow range span (NRS) for all power levels; uncertainty of ±10% NRS was applied when conservative. The programmed steam generator water level modeled in the Unit 2 analyses was a constant 64% NRS for all power levels; uncertainty of +18%/-7% NRS was applied when conservative. The "+" uncertainty means that the actual level is higher than indicated, and the "-" uncertainty means that the actual level is lower than indicated.
8. The minimum nominal full-power vessel average temperature is limited to 585.4°F for Unit 1 and 586.0°F for Unit 2, based on the analysis of the Inadvertent Actuation of the ECCS event discussed in Section 2.8.5.5.

Table 2.8.5.0-3	
Overtemperature and Overpower N-16 Setpoints	
Allowable Full-Power T_{avg} Range	574.2°F ⁽¹⁾ to 589.2°F
K_1 (safety analysis value)	1.31
K_2	0.0139/°F
K_3	0.00071/psi
K_4 (safety analysis value)	1.185
T_c^0	(2)
P^0	2,250 psia
f(Δq) Deadband	-18% Δq ⁽³⁾ to +10% Δq
f(Δq) Negative Gain	-2.78%/° Δq ⁽³⁾
f(Δq) Positive Gain	+2.34%/° Δq
High Pressurizer Pressure Reactor Trip Setpoint (safety analysis value)	2,460 psia (Unit 1) 2,437 psia (Unit 2)
Low Pressurizer Pressure Reactor Trip Setpoint (safety analysis value)	1,860 psia
Notes:	
1. Bounding value supported with respect to OTN-16/OPN-16, although the minimum T_{avg} value is limited to 585.4°F for Unit 1 and 586.0°F for Unit 2 based on the analysis of the Inadvertent Actuation of the ECCS event discussed in Section 2.8.5.5.	
2. Value to be set equal to or less than the cold leg temperature corresponding to the chosen full power operating T_{avg} .	
3. Value supported by non-LOCA transient analysis; it may change based on the fuel rod design analysis.	

Table 2.8.5.0-4				
Summary of RTS and ESFAS Functions Actuated				
FSAR Section	Event Description	RTS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
15.1.1	Decrease in Feedwater Temperature	Overpower N-16 Reactor Trip	See Table 2.8.5.0-3	2.0
15.1.2	Increase in Feedwater Flow	High-High Steam Generator Water Level Feedwater Isolation Valve Closure	100% NRS	11.0
15.1.3	Excessive Increase in Secondary Steam Flow	N/A	N/A	N/A
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	See Note 1		
15.1.5	Steam System Piping Failure – Zero Power (Core response only)	Low Steam Pressure Safety Injection (SI) and Steam Line Isolation Valve Closure	395 psia (lead/lag = 10/5)	2.0
		Steam Line Isolation Valve Closure Delay Following Low Steam Pressure Signal	N/A	5.0
		Feedwater Isolation Valve Closure Delay Following SI Signal	N/A	5.0
		SI Pumps at Full Flow Following SI Signal (with/without offsite power)	N/A	25/35
	Steam System Piping Failure – Full Power (Core response only)	Overpower N-16 Reactor Trip	See Table 2.8.5.0-3	2.0
15.2.1	Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow	There are no steam pressure regulators at CPNPP whose failure or malfunction could cause a steam flow transient (FSAR Section 15.2.1)		
15.2.2	Loss of External Electrical Load	See Note 2		
15.2.3	Turbine Trip	High Pressurizer Pressure Reactor Trip	2.460 psia (Unit 1) 2.437 psia (Unit 2)	1.25
		Overtemperature N-16 Reactor Trip	See Table 2.8.5.0-3	7.0 ⁽³⁾

Table 2.8.5.0-4 (cont.)

Summary of RTS and ESFAS Functions Actuated

FSAR Section	Event Description	RTS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
15.2.4	Inadvertent Closure of Main Steam Isolation Valves	See Note 2		
15.2.5	Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	See Note 2		
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	Low-Low Steam Generator Water Level Reactor Trip	0% NRS	2.0
		Low-Low Steam Generator Water Level Auxiliary Feedwater (AFW) Pump Start	0% NRS	60.0
15.2.7	Loss of Normal Feedwater	Low-Low Steam Generator Water Level Reactor Trip	0% NRS (Unit 1) 10% NRS (Unit 2)	2.0
		Low-Low Steam Generator Water Level Motor-Driven AFW Pumps Start	0% NRS (Unit 1) 10% NRS (Unit 2)	60.0
15.2.8	Feedwater System Pipe Break	Low-Low Steam Generator Water Level in Any One Loop Reactor Trip	10% NRS (Unit 1) 7.5% NRS (Unit 2)	2.0
		Low-Low Steam Generator Water Level in Any One Loop Motor-Driven AFW Pump Start	10% NRS (Unit 1) 7.5% NRS (Unit 2)	60
		Low-Low Steam Generator Water Level in More than One Loop Turbine-Driven AFW Pump Start	10% NRS (Unit 1) 7.5% NRS (Unit 2)	85
15.3.1	Partial Loss of Forced Reactor Coolant Flow	Reactor Coolant Low Flow Reactor Trip	87%	1.0
15.3.2	Complete Loss of Forced Reactor Coolant Flow	Reactor Coolant Pump Undervoltage Reactor Trip	See Note 4	1.5
		Reactor Coolant Pump Underfrequency Reactor Trip	57.2 Hz	0.6

Table 2.8.5.0-4 (cont.)
Summary of RTS and ESFAS Functions Actuated

FSAR Section	Event Description	RTS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
15.3.3/15.3.4	Reactor Coolant Pump Shaft Seizure (Locked Rotor)/Shaft Break	Reactor Coolant Low Flow Reactor Trip	87%	1.0
15.4.1	Uncontrolled RCCA Withdrawal from a Subcritical or Low Power Condition	Power-Range High Neutron Flux Reactor Trip (Low Setting)	35%	0.5
15.4.2	Uncontrolled RCCA Withdrawal at Power	Power-Range High Neutron Flux Reactor Trip (High Setting)	118%	0.5
		Overtemperature N-16 Reactor Trip	See Table 2.8.5.0-3	7.0 ⁽³⁾
15.4.3	RCCA Misalignment (Dropped Rod)	Low Pressurizer Pressure Reactor Trip	1,860 psia	2.0
15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	N/A	N/A	N/A
15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	This event is not applicable to CPNPP.	N/A	N/A
15.4.6	CVCS Malfunction (Boron Dilution)	Overtemperature N-16 Reactor Trip ⁽⁵⁾	See Table 2.8.5.0-3	7.0 ⁽³⁾
15.4.8	RCCA Ejection	Power-Range High Neutron Flux Reactor Trip (Low and High Settings)	35% (low setting)	0.5
			118% (high setting)	0.5
15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	Event is terminated by operator action (see Section 2.8.5.5).	N/A	N/A
15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Event is covered by the analyses of the Boron Dilution event (FSAR Section 15.4.6) and the Inadvertent Operation of the Emergency Core Cooling System During Power Operation event (FSAR Section 15.5.1).	N/A	N/A

Table 2.8.5.0-4 (cont.)
Summary of RTS and ESFAS Functions Actuated

FSAR Section	Event Description	RTS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
15.5.3	A Number of BWR Transients	This event is not applicable to CPNPP.	N/A	N/A
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Low Pressurizer Pressure Reactor Trip	1,860 psia	2.0
15.8	ATWS	ATWS Mitigation System Actuation Circuitry (AMSAC) – Turbine Trip (TT), AFW Pump Start	N/A	30 (TT) 60 (AFW)

Notes:

1. Transient bounded by steam system piping failure (FSAR Section 15.1.5).
2. Transient bounded by turbine trip (FSAR Section 15.2.3).
3. Modeling the overtemperature N-16 reactor trip included a time constant (first order lag) of 5 or 6 seconds for the cold leg RTDs and an additional delay of 2 seconds or 1 second to account for electronic delays, reactor trip breakers opening, and RCCA gripper release.
4. Reactor coolant pump power supply undervoltage reactor trip was assumed to occur 1.5 seconds following the loss of bus voltage. For a Westinghouse-designed plant, 1.5 seconds is a typical, conservative value for this delay. A typical breakdown of the time delay is as follows.

Undervoltage trip circuitry including	
adjustable delay preventing spurious trip	0.95 second
EMF decay	0.25 second
Trip breaker opening	0.15 second
<u>RCCA release time</u>	<u>0.15 second</u>
TOTAL DELAY TIME	1.50 seconds

However, only the total time is used in the analysis and it is verified during plant surveillance tests. It is conservative because the surveillance tests ensure that the assumed value bounds the measured plant value. Also, note that the analysis conservatively assumes that the pump coastdown begins at time zero, even though in reality pump speed will not be reduced as the EMF decays to the undervoltage setpoint.

5. Although not an RTS or ESFAS function, the volume control tank water level high alarm is credited in the boron dilution analyses for Modes 3, 4, and 5, consistent with the methodology of Reference 22; a setpoint of 70% of span was applied.

Table 2.8.5.0-5

Core Kinetics Parameters and Reactivity Feedback Coefficients

Parameter	Beginning of Cycle (Minimum Feedback)	End of Cycle (Maximum Feedback)
MTC, pcm/°F	5.0 ($\leq 70\%$ RTP) ⁽¹⁾ linearly ramping to 0.0 at 100% RTP)	N/A
Moderator Density Coefficient, $\Delta k/(g/cc)$	N/A	0.50
Doppler Temperature Coefficient, pcm/°F	-0.91	-2.90
Doppler-Only Power Coefficient, pcm/%power (Q = power in %)	-9.55 + 0.035Q	-19.4 + 0.068Q
Delayed Neutron Fraction	0.0070 (maximum)	0.0044 (minimum)
Minimum Doppler Power Defect, pcm		
– RCCA Ejection	1,000	950
– RCCA Withdrawal from Subcritical	1,000	N/A
Note:		
1. RTP \equiv Rated Thermal Power		

Table 2.8.5.0-6

Summary of Initial Conditions and Computer Codes Used

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Vessel Coolant Flow, gpm	Vessel Average Coolant Temp, °F	RCS Pressure, psia
Decrease in Feedwater Temperature	RETRAN	WRB-2	Yes	100 (3,628 MWt - NSSS power)	396,400	589.2	2,250
Increase in Feedwater Flow	RETRAN VIPRE	WRB-2 (HFP) W-3 (HZP)	Yes (HFP) No (HZP)	100 0 (3,628 MWt - NSSS power)	396,400 (HFP) 382,800 (HZP)	589.2 (HFP) 557.0 (HZP)	2,250
Excessive Increase in Secondary Steam Flow	N/A	WRB-2	Yes	100 (3,612 MWt - Core Power)	396,400	589.2	2,250
Inadvertent Opening of a Steam Generator Relief or Safety Valve	Event bounded by the steam system piping failure event.						
Rupture of a Steam Pipe – Zero Power Core Response	RETRAN ANC VIPRE	W-3	No	0 (3,628 MWt - NSSS power)	382,800	557.0	2,250
Rupture of a Steam Pipe – Full Power Core Response	RETRAN VIPRE	WRB-2	Yes	100 (3,628 MWt - NSSS power)	396,400	589.2	2,250
Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	There are no steam pressure regulators at CPNPP whose failure or malfunction could cause a steam flow transient (FSAR Section 15.2.1)						
Loss of External Electrical Load	Event bounded by the turbine trip event.						
Turbine Trip	RETRAN	N/A (pressure) WRB-2 (DNB)	N/A (pressure) Yes (DNB)	100.6 (pressure) 100 (DNB) (3,628 MWt - NSSS power)	382,800 (pressure) 396,400 (DNB)	583.2 (Unit 1) & 589.2 (Unit 2) (RCS pressure) 595.2 (MSS pressure) 589.2 (DNB)	2,220 (pressure) 2,250 (DNB)

Table 2.8.5.0-6 (cont.)

Summary of Initial Conditions and Computer Codes Used

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Vessel Coolant Flow, gpm	Vessel Average Coolant Temp, °F	RCS Pressure, psia
Inadvertent Closure of Main Steam Isolation Valves	Event bounded by the turbine trip event.						
Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	Event bounded by the turbine trip event.						
Loss of Nonemergency AC Power to the Station Auxiliaries	RETRAN	N/A	N/A	100.6 (3,628 MWt - NSSS power)	382,800	568.2 (Unit 1) 595.2 (Unit 2)	2,280 (Unit 1) 2,220 (Unit 2)
Loss of Normal Feedwater	RETRAN	N/A	N/A	100.6 (3,628 MWt - NSSS power)	382,800	568.2 (Unit 1) 595.2 (Unit 2)	2,280 (Unit 1) 2,220 (Unit 2)
Feedwater System Pipe Break	RETRAN	N/A	N/A	100.6 (3,628 MWt - NSSS power)	382,800	595.2	2,220
Partial Loss of Forced Reactor Coolant Flow	RETRAN VIPRE	WRB-2	Yes	100 (3,628 MWt - NSSS power)	396,400	589.2	2,250
Complete Loss of Forced Reactor Coolant Flow	RETRAN VIPRE	WRB-2	Yes	100 (3,628 MWt - NSSS power)	396,400	589.2	2,250
Locked Rotor/Shaft Break	RETRAN VIPRE	N/A (pressure) WRB-2 (DNB)	N/A (pressure) Yes (DNB)	100.6 (pressure) 100 (DNB) (3,628 MWt - NSSS power)	382,800 (pressure) 396,400 (DNB)	595.2 (pressure) 589.2 (DNB)	2,280 (pressure) 2,250 (DNB)
Uncontrolled RCCA Withdrawal from a Subcritical or Low Power Condition	TWINKLE FACTRAN VIPRE	W-3 ⁽¹⁾ WRB-2 ⁽²⁾	No	0 (3,612 MWt - core power)	176,088 ⁽³⁾	557	2,220
Uncontrolled RCCA Withdrawal at Power	RETRAN	N/A (Pressure) WRB-2 (DNB)	N/A (Pressure) Yes (DNB)	100 60 10 (3,628 MWt - NSSS power)	396,400	589.2 (100%) 576.3 (60%) 560.2 (10%)	2,250

Table 2.8.5.0-6 (cont.)

Summary of Initial Conditions and Computer Codes Used

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Vessel Coolant Flow, gpm	Vessel Average Coolant Temp, °F	RCS Pressure, psia
RCCA Misalignment (Dropped Rod)	LOFTRAN ⁽⁴⁾ ANC VIPRE	WRB-2	Yes	100 (3,612 MWt - core power)	396,400	589.2	2,250
Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	See Licensing Report Section 2.8.5.4.4.						
CVCS System Malfunction (Boron Dilution)	N/A	N/A	N/A	100 (Mode 1) 5 (Mode 2) 0 (Mode 3) 0 (Mode 4) 0 (Mode 5) N/A (Mode 6)	N/A	595.2 (Mode 1) 564.7 (Mode 2) 557.0 and 350.0 (Mode 3) 350.0 and 200.0 (Mode 4) 200.0 and 68.0 (Mode 5) N/A (Mode 6)	2,250 (Modes 1 and 2) 14.7 (Modes 3, 4, 5) N/A (Mode 6)
Spectrum of RCCA Ejection Accidents	TWINKLE FACTRAN	N/A	N/A	100.6 (HFP) 0 (HZP) (3,612 MWt - core power)	382,800 (HFP) 176,088 ⁽³⁾ (HZP)	589.2 (HFP) 557.0 (HZP)	2,220
Inadvertent Operation of the Emergency Core Cooling System During Power Operation	RETRAN	N/A (filling) WRB-2 (DNB)	N/A (filling) Yes (DNB)	100.6 (filling) 100 (DNB) (3,628 MWt - NSSS power)	382,800 (filling) 396,400 (DNB)	579.4 (Unit 1 filling) 580.0 (Unit 2 filling) 589.2 (DNB)	2,220 (filling) 2,250 (DNB)
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Event is covered by the analyses of the Boron Dilution event and the Inadvertent Operation of the Emergency Core Cooling System During Power Operation event.						
Inadvertent Opening of a Pressurizer Safety or Relief	RETRAN	WRB-2	Yes	100 (3,628 MWt - NSSS power)	396,400	589.2	2,250

Table 2.8.5.0-6 (cont.)

Summary of Initial Conditions and Computer Codes Used

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Vessel Coolant Flow, gpm	Vessel Average Coolant Temp, °F	RCS Pressure, psia
Valve							
ATWS	LOFTRAN	N/A	N/A	100 (3,628 MWt - NSSS power)	382,800	589.2	2,250

Notes:

1. Below the first mixing vane grid.
2. Above the first mixing vane grid.
3. Flow from two loops = 0.46 * TDF.
4. The LOFTRAN portion of the analysis was generic; the DNB evaluation performed with VIPRE utilized the plant-specific values presented.

Table 2.8.5.0-7
Non-LOCA Transients Evaluated or Analyzed⁽³⁾

Transient	Report Section	FSAR Section	Notes
Decrease in Feedwater Temperature	2.8.5.1.1	15.1.1	1
Increase in Feedwater Flow	2.8.5.1.1	15.1.2	1
Excessive Increase in Secondary Steam Flow	2.8.5.1.1	15.1.3	2
Inadvertent Opening of a Steam Generator Relief or Safety Valve	2.8.5.1.1	15.1.4	2
Rupture of a Steam Pipe – Zero Power Core Response	2.8.5.1.2	15.1.5	1
Rupture of a Steam Pipe – Full Power Core Response	2.8.5.1.2	15.1.5	1
Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	2.8.5.2.1	15.2.1	4
Loss of External Electrical Load	2.8.5.2.1	15.2.2	2
Turbine Trip	2.8.5.2.1	15.2.3	1
Inadvertent Closure of Main Steam Isolation Valves	2.8.5.2.1	15.2.4	2
Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	2.8.5.2.1	15.2.5	2
Loss of Nonemergency AC Power to the Station Auxiliaries	2.8.5.2.2	15.2.6	1
Loss of Normal Feedwater	2.8.5.2.3	15.2.7	1
Feedwater System Pipe Break	2.8.5.2.4	15.2.8	1
Partial Loss of Forced Reactor Coolant Flow	2.8.5.3.1	15.3.1	1
Complete Loss of Forced Reactor Coolant Flow	2.8.5.3.1	15.3.2	1
Locked Rotor/Shaft Break	2.8.5.3.2	15.3.3, 15.3.4	1
Uncontrolled RCCA Withdrawal from a Subcritical or Low Power Condition	2.8.5.4.1	15.4.1	1
Uncontrolled RCCA Withdrawal at Power	2.8.5.4.2	15.4.2	1
RCCA Misalignment (Dropped Rod)	2.8.5.4.3	15.4.3	1
Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	2.8.5.4.4	15.4.4	2
A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	N/A	15.4.5	N/A
CVCS System Malfunction (Boron Dilution)	2.8.5.4.5	15.4.6	1
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	N/A	15.4.7	N/A

Table 2.8.5.0-7 (cont.)

Non-LOCA Transients Evaluated or Analyzed⁽³⁾

Transient	Report Section	FSAR Section	Notes
Spectrum of RCCA Ejection Accidents	2.8.5.4.6	15.4.8	1
Inadvertent Operation of the Emergency Core Cooling System During Power Operation	2.8.5.5	15.5.1	1
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	N/A	15.5.2	2
A Number of BWR Transients	N/A	15.5.3	N/A
Inadvertent Opening of a Pressurizer Safety or Relief Valve	2.8.5.6.1	15.6.1	1
ATWS	2.8.5.7	15.8	1
Notes:			
<ol style="list-style-type: none"> 1. Complete analysis. 2. Evaluation. 3. All analyses and evaluations cover Units 1 and 2. 4. There are no steam pressure regulators at CPNPP whose failure or malfunction could cause a steam flow transient (FSAR Section 15.2.1). 			

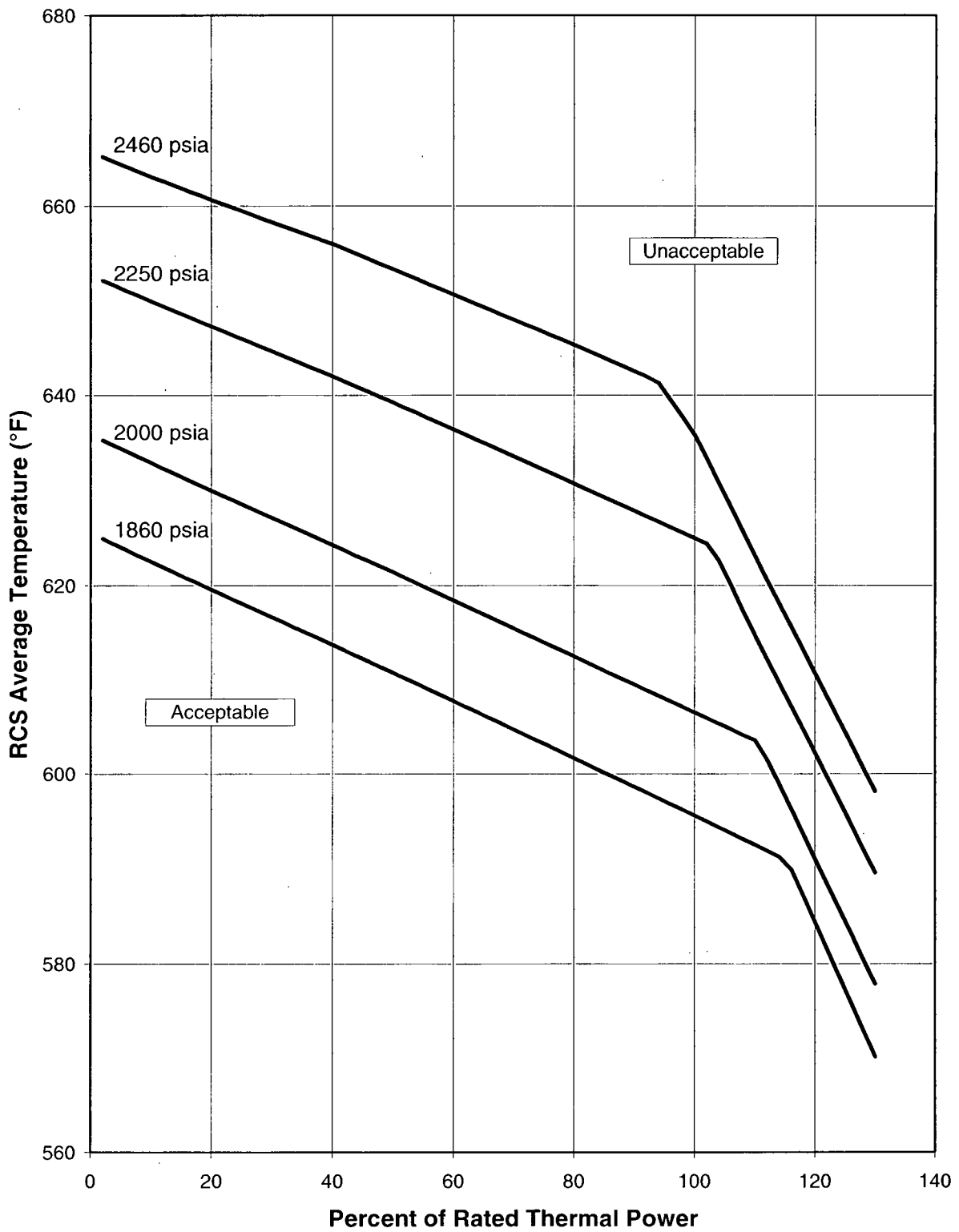


Figure 2.8.5.0-1 Reactor Core Safety Limits

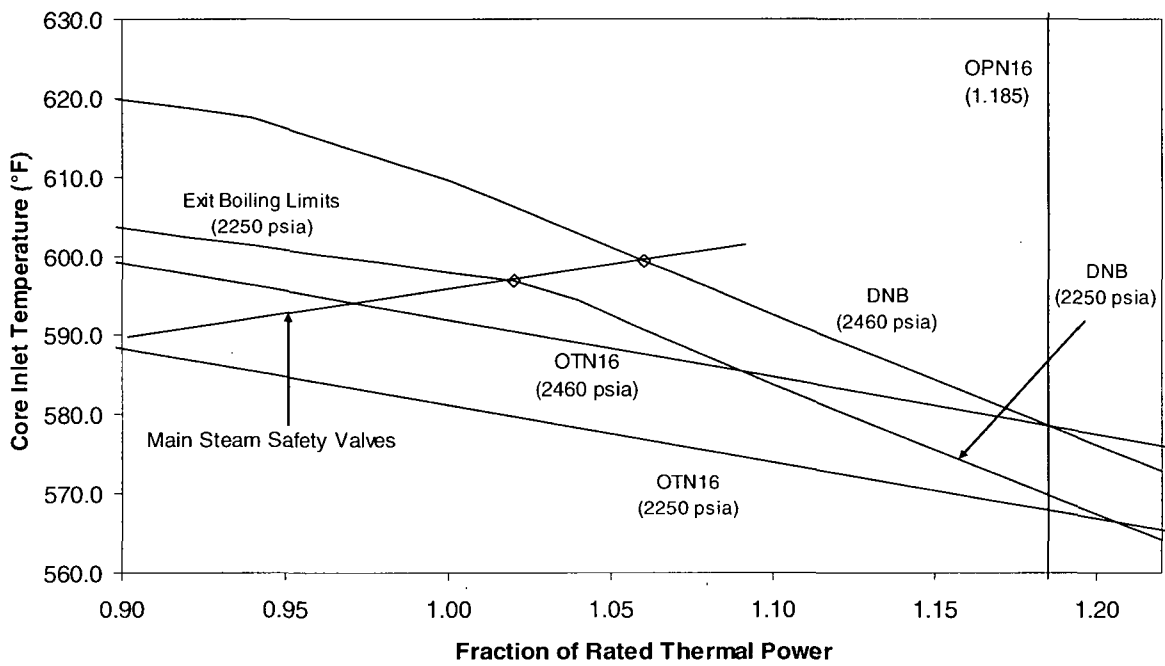


Figure 2.8.5.0-2 Illustration of OTN-16 and OPN-16 Protection

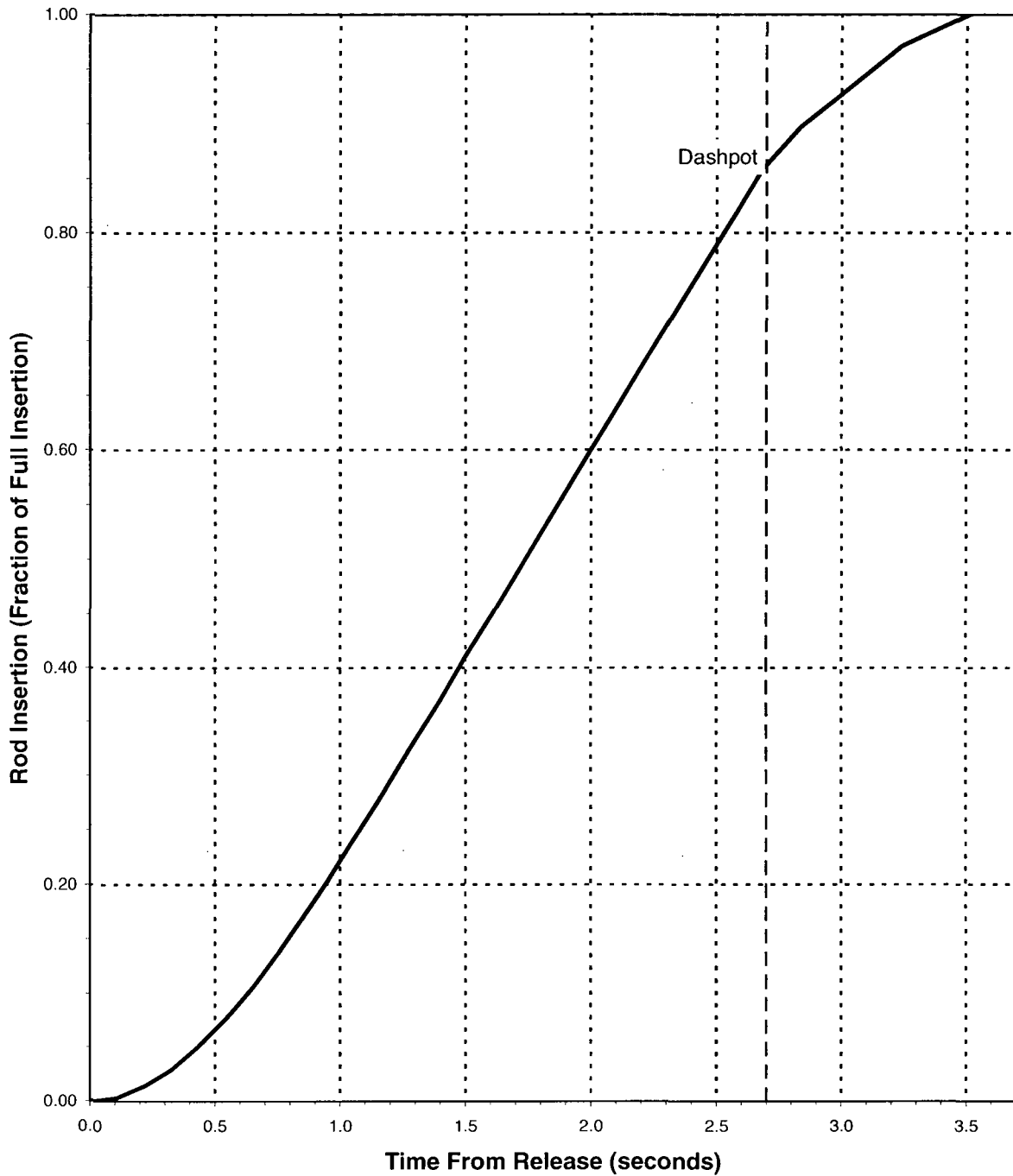


Figure 2.8.5.0-3 Fractional Rod Insertion Versus Time from Release

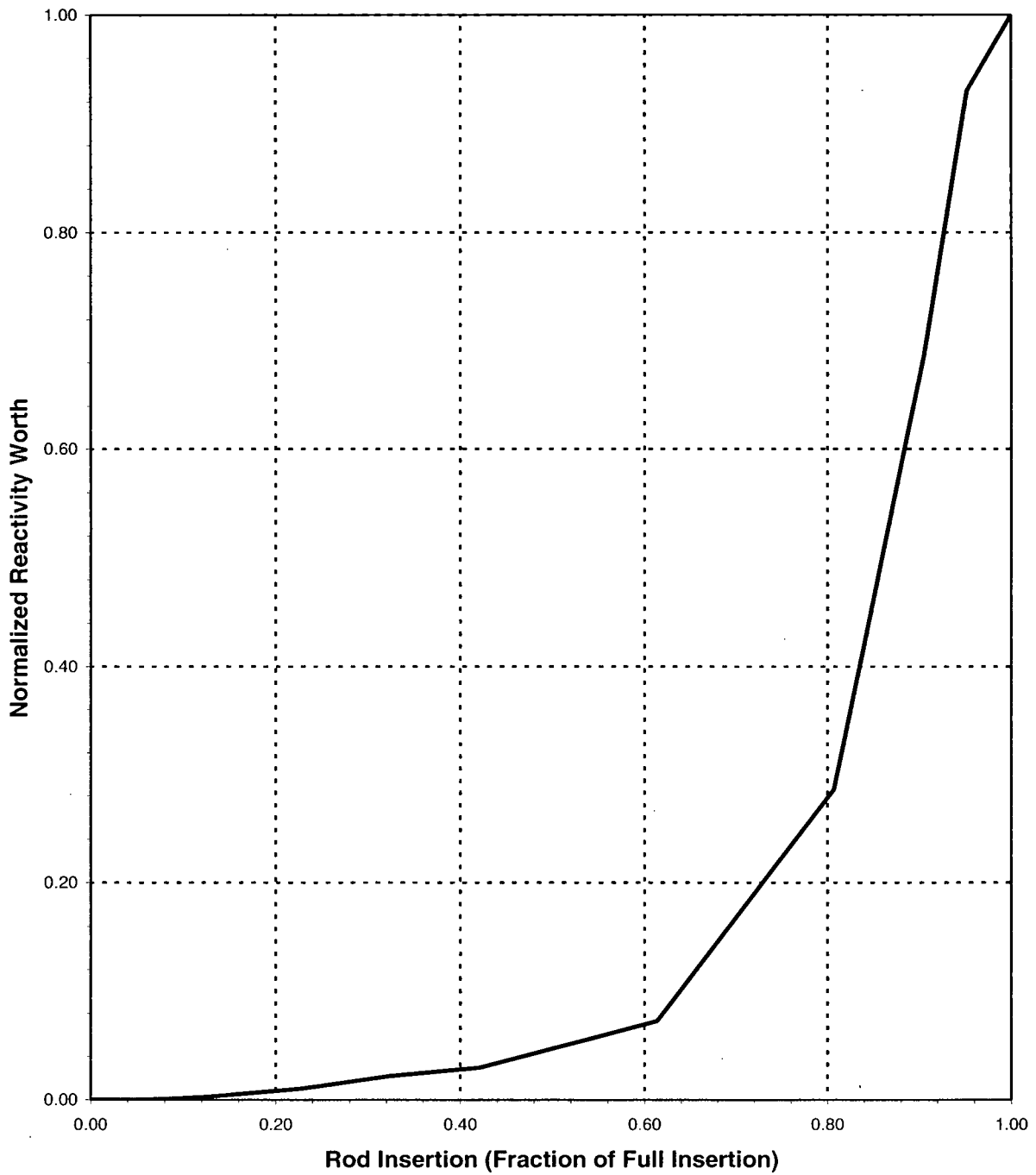


Figure 2.8.5.0-4 Normalized RCCA Reactivity Worth Versus Fractional Rod Insertion

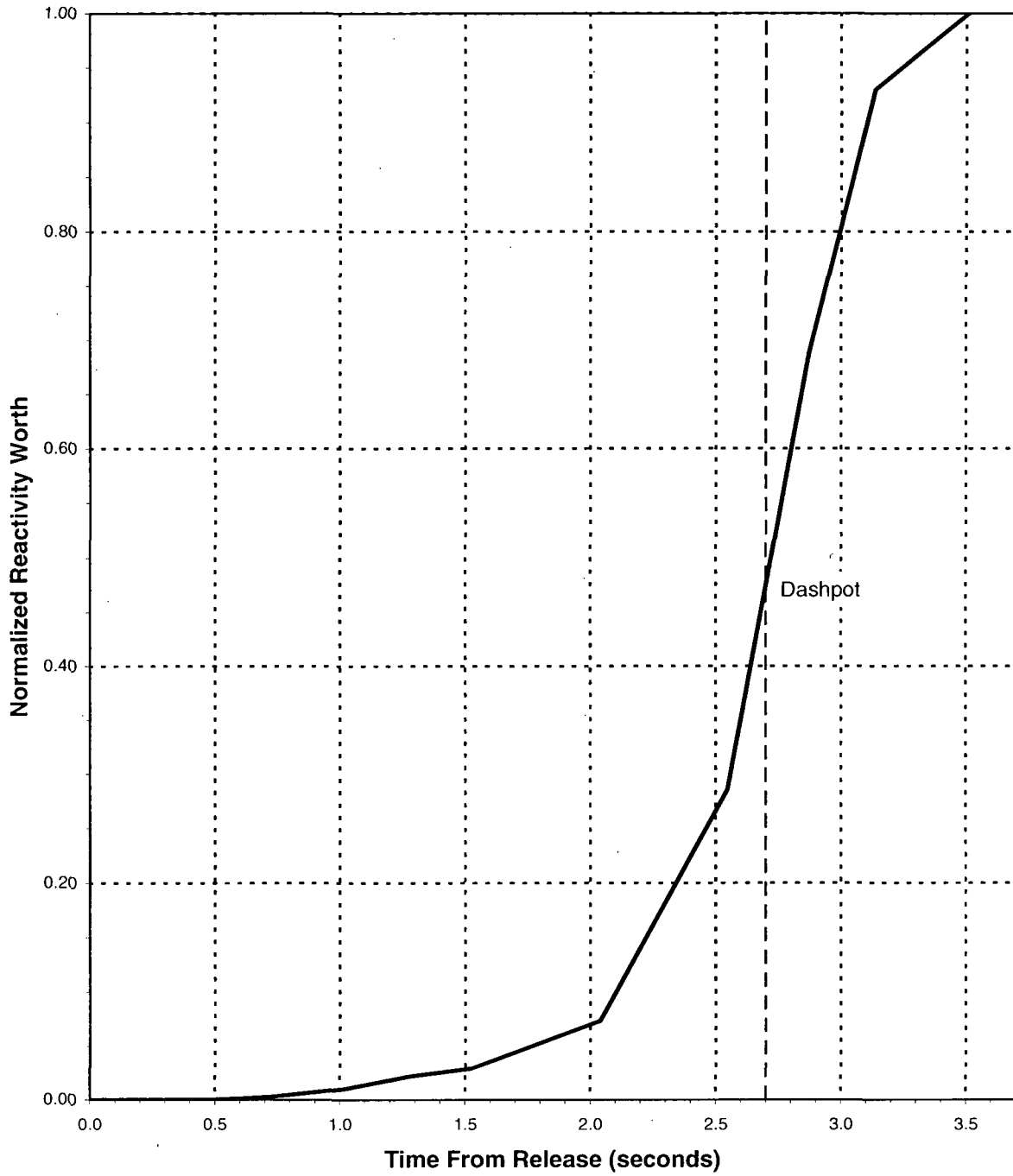


Figure 2.8.5.0-5 Normalized RCCA Reactivity Worth Versus Time from Release

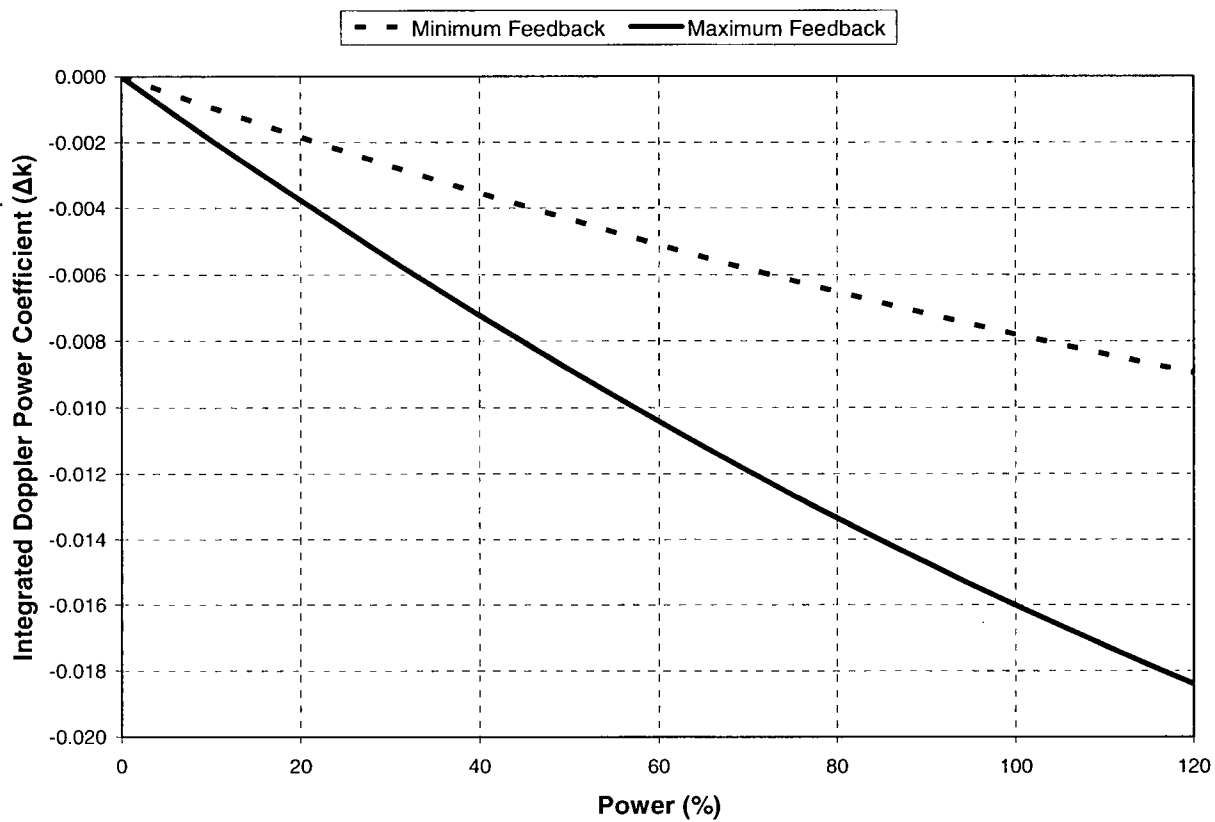


Figure 2.8.5.0-6 Integrated DPC Used in Non-LOCA Transient Analyses

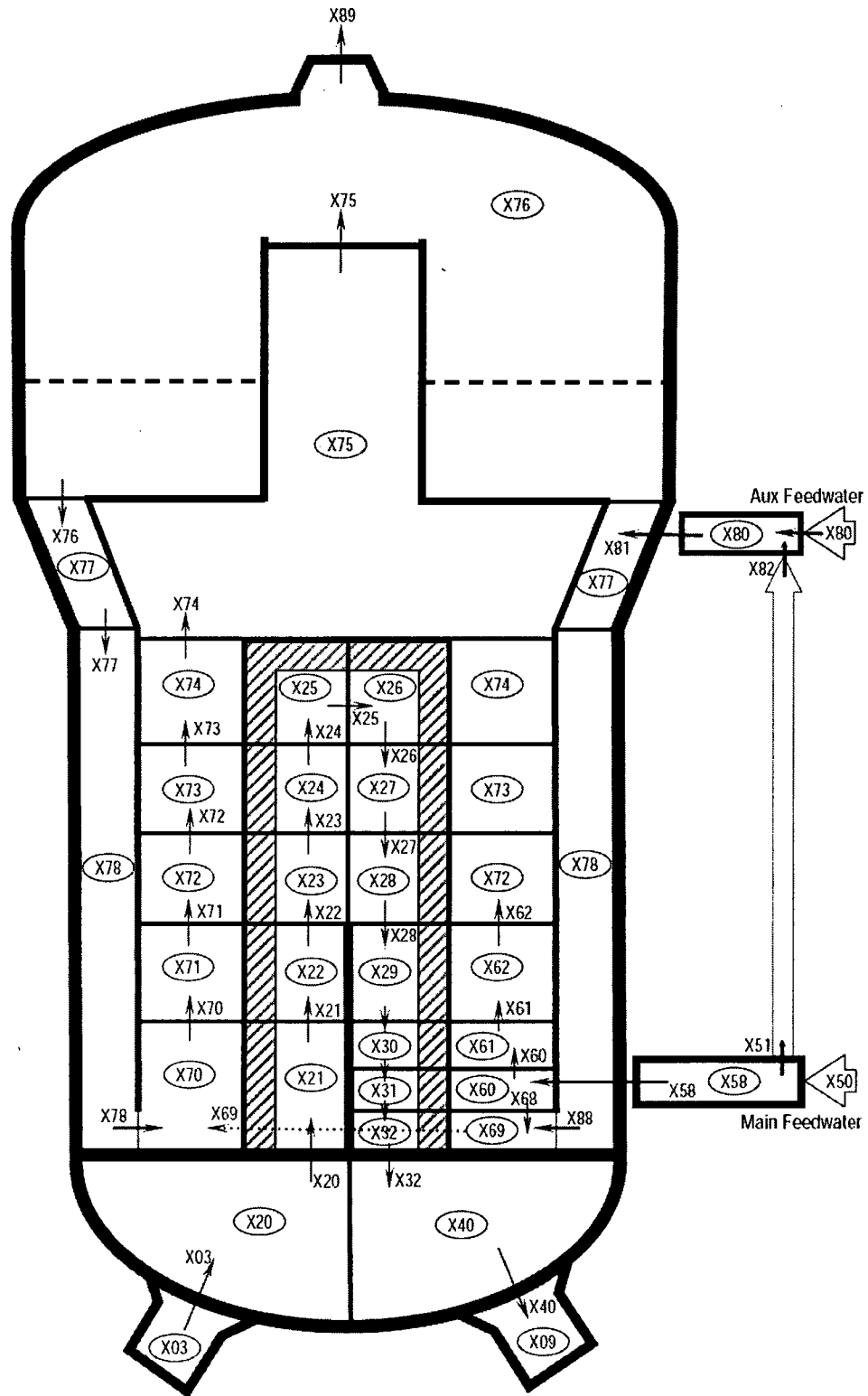


Figure 2.8.5.0-7 RETRAN Nodalization Diagram for Model D-5 Steam Generator

2.8.5.1 Increase in Heat Removal by the Secondary System

2.8.5.1.1 Decrease In Feedwater Temperature, Increase In Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

2.8.5.1.1.1 Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature that increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase can result in fuel damage or excessive reactor system pressure. Reactor trip and safety systems are actuated to mitigate the transient.

The review covered:

- The postulated initial core and reactor conditions
- The methods of thermal-hydraulic analyses
- The sequence of events
- The assumed reactions of reactor system components
- The functional and operational characteristics of the reactor trip system
- The results of the transient analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs)
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation
- GDC-20, insofar as it requires that the reactor trip system be designed to automatically initiate the operation of appropriate systems, including reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs
- GDC-26, insofar as it requires that reactivity control systems be provided and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II).
2. To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.
3. To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

Note that the term "fuel damage" as used in Item 1 above is defined as penetration of the fission product barrier (that is, the fuel rod cladding). Also note that American National Standards Institute (ANSI) N18.2-1973 expands the definitions of the four conditions enumerated in Items 1 through 3 above.

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components including the fuel, reactor vessel internals, and reactivity control systems. FSAR Chapter 7 provides the details of the control and protection systems instrumentation design and logic. This information supports the FSAR Chapter 15 accident analyses, which show that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of

proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components. FSAR Chapter 5 discusses the RCS design.

- GDC-20, Protection System Functions, is described in FSAR Section 3.1.3.1.

A fully automatic protection system with appropriate redundant channels is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of Institute of Electrical and Electronic Engineers (IEEE) 279-1971 and IEEE 379-1972. The reactor protection system automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the control rod drive mechanisms (CRDMs) of all full-length rod cluster control assemblies (RCCAs). This causes the rods to insert by the force of gravity, rapidly reducing the reactor power output. The response and adequacy of the protection system has been verified by analysis of anticipated transients.

The engineered safety features (ESF) actuation system automatically initiates emergency core cooling and other safeguards functions by sensing accident conditions using redundant analog channels measuring diverse variables. In addition, manual actuation of safeguards can be performed where ample time is available for operator action. In either case, the ESF actuation system automatically trips the reactor on manual or automatic safety injection signal generation.

- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control methods are provided. These are RCCAs and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation the shutdown rod banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the full-length control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCA are presented in FSAR Chapter 4. The operation is discussed in FSAR Chapter 7. The means of controlling the boric acid concentration are described in FSAR Chapter 9. Performance analyses under accident conditions are included in FSAR Chapter 15.

Decrease in Feedwater Temperature

FSAR Section 15.1.1.1 states that reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Opening of a low-pressure heater bypass valve, tripping of the heater drain pumps, and isolating all high pressure extraction steam causes a reduction in feedwater temperature which increases the thermal load on the primary system. For this event, there is a sudden reduction in feedwater temperature into the steam generators.

At power, the increased subcooling caused by the reduced feedwater temperature creates a greater load demand on the RCS. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature, and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, since the rate of energy change is reduced as the load and feedwater flow decrease, the no-load transient is less severe than the full-power case.

The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow, that is, the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT unless terminated by a reactor trip.

Based on results discussed in FSAR Section 15.1.1.3, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

Increase in Feedwater Flow

FSAR Section 15.1.2.1 states that the addition of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. An example of excessive feedwater flow would be the full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, the excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of excess feedwater may cause a decrease in RCS temperature, and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity.

FSAR Section 15.1.2.3 states that, for excessive feedwater addition events, the results show that the departure from nucleate boiling ratios (DNBRs) encountered are above the limiting safety analysis values at all times. Therefore, the departure from nucleate boiling (DNB) design basis as described in FSAR Section 4.4 is met.

Excessive Increase in Secondary Steam Flow

FSAR Section 15.1.3.1 states that an excessive increase in secondary system steam flow (excessive load increase incident) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor 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 15- to 100-percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor trip system. Steam flow increases greater than 10 percent are discussed in FSAR Sections 15.1.4 and 15.1.5. This incident, which is classified as a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973, 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 systems.

FSAR Section 15.1.3.3 concludes that the DNBR remains above the safety analysis limit value for a 10-percent step-load increase. Therefore, the design basis for DNBR as described in FSAR Section 4.4 is met. The analysis showed that the plant reached a stabilized condition rapidly following the load increase.

Inadvertent Opening of a Steam Generator Relief or Safety Valve

FSAR Section 15.1.4.1 states that the most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening, with failure to close, of the largest of any single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in FSAR Section 15.1.5.

The steam release as a consequence of this accident results in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS 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.

FSAR Section 15.1.4.3 concludes that all applicable acceptance criteria are satisfied for an accidental depressurization of the main steam system. An explicit analysis was performed that showed that it is bounded by the hypothetical steam line rupture analysis presented in FSAR Section 15.1.5.

2.8.5.1.1.2 Technical Evaluation

2.8.5.1.1.2.1 Decrease in Feedwater Temperature

2.8.5.1.1.2.1.1 Introduction

Opening of a low-pressure heater bypass valve, tripping of the heater drain pumps, and isolating all high-pressure extraction steam causes a reduction in feedwater temperature that

increases the thermal load on the primary system. For this event, there is a sudden reduction in feedwater temperature into the steam generators.

At power, the increased subcooling caused by the reduced feedwater temperature creates a greater load demand on the RCS. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature, and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, since the rate of energy change is reduced as the load and feedwater flow decrease, the no-load transient is less severe than the full power case.

Depending on the magnitude of the temperature reduction and the operation of the automatic rod control system, the net effect on the RCS can be similar to the effect of increasing secondary steam flow; that is, the reactor will reach a new equilibrium condition at a power level corresponding to the new temperature difference across the secondary-side of the steam generator. For large feedwater temperature reductions, the overpower N-16 reactor trip function will prevent a power increase that could lead to a DNBR that is lower than the safety analysis limit value.

2.8.5.1.1.2.1.2 Input Parameters, Assumptions and Acceptance Criteria

The decrease in feedwater temperature event was analyzed to confirm that the minimum DNBR and fuel centerline temperature design bases are met. The feedwater temperature reduction analysis was performed with the following assumptions to bound feedwater temperature reductions greater than 70°F:

- The Revised Thermal Design Procedure (RTDP) (Reference 1) was used for the cases initiated from full power. The initial reactor power, RCS pressure and RCS temperature were assumed to be at the nominal values consistent with steady-state full-power operation. Minimum measured flow (MMF) was also modeled. Uncertainties for these initial conditions were accounted for in the DNBR safety analysis limit as described in Reference 1.
- The analyses were performed at the updated nuclear steam supply system (NSSS) power of 3,628 MWt.
- The analyses model CPNPP Unit 1 steam generators (Westinghouse Model $\Delta 76$). This is applicable since the CPNPP Units 1 and 2 primary-side models are not significantly different and because of the larger heat transfer area associated with the Model $\Delta 76$ steam generators. A larger heat transfer area is conservative with respect to the decrease in feedwater temperature transient.
- A conservative feedwater temperature reduction of 246°F was analyzed for the nominal high and nominal low feedwater temperatures. The temperature reduction was modeled as a step decrease from the nominal high and low full-power values of 450.3° and 390.0°F, respectively.

-
- The heat capacities of the RCS and steam generator thick metal were not considered, thereby maximizing the potential temperature reduction of the reactor coolant.
 - The overpower N-16 reactor trip function was assumed to be available for event mitigation, as required. Depending on the magnitude of the feedwater temperature reduction and the availability of the rod control system, if core power increases, the reactor may reach a new equilibrium condition and not require actuation of the reactor trip system (RTS).
 - Both manual and automatic rod control cases were considered.
 - An initial water level of nominal-minus-uncertainty was modeled for all four steam generators.
 - Pressurizer sprays and power-operated relief valves (PORVs) were modeled to reduce RCS pressure, resulting in a conservative evaluation of the margin to the DNBR safety analysis limit.

Based on its frequency of occurrence, the decrease in feedwater temperature event is considered to be a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. As such, the applicable acceptance criteria for this incident are:

- Pressures in the RCS and main steam system (MSS) should be maintained below 110 percent of the respective design pressures.
- Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains greater than the 95/95 DNBR safety analysis limit in the limiting fuel rods and that the centerline temperature of the fuel rods with the peak linear heat rate (kW/ft) does not exceed the UO₂ melting temperature. Fuel melting is precluded by ensuring that the maximum transient core average thermal power does not exceed a value that would result in exceeding the kW/ft value corresponding to fuel centerline melting at the core hot spot.
- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. Demonstrating that the pressurizer does not become water-solid ensures a more serious plant condition is not generated. Since this event results in a cooldown of the RCS, the reactor coolant experiences a reduction in volume, and therefore pressurizer filling is not a concern.

2.8.5.1.1.2.1.3 Description of Analyses and Evaluation

The sudden decrease in feedwater temperature transient was analyzed using the RETRAN computer code (Reference 2). This code simulates a multi-loop RCS, core neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray and heaters, steam

generators and main steam safety valves (MSSVs). The code computes pertinent plant variables including temperatures, pressures, and power level.

A decrease in feedwater temperature can be caused by an accidental opening of a low-pressure feedwater heater bypass valve or a load reduction. Evaluations of CPNPP plant transients concluded that a bypass of a low-pressure feedwater heater leads to an immediate consequential isolation of the other feedwater heaters, thereby resulting in a maximum feedwater temperature decrease of 246°F. It has been determined that the excessive increase in steam flow event (a step-load increase of 10 percent from full load), which is discussed in Licensing Report (LR) subsection 2.8.5.1.1.2.3, is equivalent to a 70°F reduction in the feedwater temperature. Therefore, the consequences of a feedwater temperature reduction of up to 70°F are equivalent or bounded by the consequences of a 10-percent load increase from full power. The 246°F feedwater temperature reduction analysis was performed to bound temperature reductions greater than 70°F. Table 2.8.5.1.1.2.1-1 summarizes the analyzed cases.

2.8.5.1.1.2.1.4 Decrease in Feedwater Temperature Results

The sudden decrease in feedwater temperature transient was analyzed for CPNPP for the SPU program. Manual and automatic rod control, as well as high and low nominal feedwater temperatures were considered. The most limiting case was the Unit 1 temperature reduction from the nominal high feedwater temperature with manual rod control. This case resulted in the largest reactivity feedback and produced the lowest DNBR. The reactor was tripped by the overpower N-16 signal. If the reactor was in automatic rod control mode, the control rods would be inserted at the maximum rate and the resulting transient would not be as limiting in terms of the minimum DNBR as the case with manual rod control. Table 2.8.5.1.1.2.1-2 shows the time sequence of events for the decrease in feedwater temperature transient and Table 2.8.5.1.1.2.1-3 provides the results. Figures 2.8.5.1.1.2.1-1 through 2.8.5.1.1.2.1-4 show transient responses of various system parameters for the limiting (Unit 1) decrease in feedwater temperature case.

2.8.5.1.1.2.2 Increase in Feedwater Flow

2.8.5.1.1.2.2.1 Introduction

Addition of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. An example of excessive feedwater flow would be a full opening of a main feedwater flow control valve (FCV) due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of excess feedwater may cause a decrease in RCS temperature, and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity.

The transient is analyzed using the RETRAN (Reference 2) and VIPRE codes. RETRAN computes pertinent plant variables including temperatures, pressures, and power level. VIPRE is used to verify that the DNBR remains above the DNBR safety analysis limit.

2.8.5.1.1.2.2 Input Parameters, Assumptions and Acceptance Criteria

The feedwater flow increase event is analyzed to confirm that the minimum DNBR remains greater than the safety analysis limit. Therefore, the analysis uses the following key modeling assumptions:

- The RTDP (Reference 1) was used for the cases initiated from full power. The initial reactor power, RCS pressure, and RCS temperature were assumed to be at the nominal values consistent with steady-state full-power operation. MMF was also modeled. Uncertainties for these initial conditions were accounted for in the DNBR safety analysis limit as described in Reference 1.
- The analyses were performed at the uprated NSSS power of 3,628 MWt.
- The analyses model CPNPP Unit 1 steam generators (Westinghouse Model $\Delta 76$). This is applicable since the CPNPP Units 1 and 2 primary-side models are not significantly different and because of the larger heat transfer area associated with the Model $\Delta 76$ steam generators. A larger heat transfer area is conservative with respect to the increase in feedwater flow transient.
- For the single-loop feedwater flow increase event at full-power, one feedwater control valve was assumed to malfunction, resulting in a step increase to 207.5 percent of the nominal full-power feedwater flow to one steam generator.
- For the multiple-loop feedwater flow increase event at full-power, two feedwater control valves were assumed to malfunction, resulting in a step increase to 168.2 percent of the nominal full-power feedwater flow to two steam generators.
- The increase in feedwater flow rate results in a decrease in the feedwater temperature (enthalpy) due to the reduced efficiency of the feedwater heaters. For full-power, a 25 Btu/lbm decrease in the feedwater enthalpy was conservatively assumed to occur coincident with the feedwater flow increase.
- For the single-loop feedwater flow increase event at no-load conditions, one feedwater control valve was assumed to malfunction, resulting in a step increase to 254.3 percent of the full-power nominal flow to one steam generator.
- For the multiple-loop feedwater flow increase event at no-load conditions, two feedwater control valves were assumed to malfunction, resulting in a steam increase to 158.7 percent of the full-power nominal flow to two steam generators.
- For the cases initiated at zero power, initial reactor power, RCS pressure, and RCS temperature were assumed to be at levels corresponding to no-load conditions. Thermal design flow was also modeled. In addition, the reactor was assumed to be at the minimum shutdown margin condition of 1.3-percent $\Delta k/k$.

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- For the full-power cases, an initial water level corresponding to the nominal level minus uncertainties was modeled in all four steam generators, while an initial water level corresponding to the nominal level was modeled for the zero-power cases.
 - Pressurizer sprays and PORVs were modeled to reduce RCS pressure, resulting in a conservative evaluation of the margin to the DNBR safety analysis limit.
 - The full-power cases were analyzed with manual and automatic rod control.
 - For cases at zero-power conditions, the initial feedwater temperature was assumed to be 70°F.
 - The heat capacities of the RCS and steam generator thick metal were not considered, thereby maximizing the potential temperature reduction of the reactor coolant.
 - Reactor trip on turbine trip was assumed operable in the feedwater flow increase analyses. However, this trip is not required for core protection. Assuming reactor trip on turbine trip to be operable is consistent with the fact that an increase in feedwater flow is a cooldown transient. If this trip were not assumed, then following turbine trip and feedwater isolation on high-high steam generator level, the transient would resemble a loss of normal feedwater, an RCS heatup event, with level dropping until a reactor trip occurs on a low-low steam generator level.

Based on its frequency of occurrence, the increase in feedwater flow event is considered to be a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. As such, the applicable acceptance criteria for this incident are:

- Pressure in the RCS and MSS should be maintained below 110 percent of the design pressures.
- Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains greater than the 95/95 DNBR safety analysis limit in the limiting fuel rods and that the centerline temperature of the fuel rods with the peak linear heat rate (kW/ft) does not exceed the UO₂ melting temperature. Fuel melting is precluded by ensuring that the maximum transient core average thermal power does not exceed a value that would result in exceeding the kW/ft value corresponding to fuel centerline melting at the core hot spot.
- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. Demonstrating that the pressurizer does not become water-solid ensures a more serious plant condition is not generated. Since this event results in a cooldown of the RCS, the reactor coolant experiences a reduction in volume, and therefore pressurizer filling is not a concern.

The primary acceptance criterion used in this analysis is that the minimum DNBR remains greater than the safety analysis limit. The event does not challenge the primary- or secondary-side pressure limits since the increased heat removal results in an RCS cooldown.

2.8.5.1.1.2.2.3 Description of Analyses and Evaluations

The excessive heat removal due to a feedwater flow increase transient was analyzed with the RETRAN computer code (Reference 2). This code simulates a multi-loop RCS, core neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray and heaters, steam generators, and MSSVs. The code computes pertinent plant variables including temperatures, pressures, and power level.

The excessive feedwater flow event assumes an accidental opening of one or more feedwater control valves with the reactor at full- and zero-power conditions, and with automatic and manual rod control, where applicable. Both the automatic and manual rod control cases assume a conservatively large moderator density coefficient characteristic of end-of-life (EOL) conditions. Table 2.8.5.1.1.2.2-1 summarizes the analyzed cases.

2.8.5.1.1.2.2.4 Increase in Feedwater Flow Results

A comparison of the multiple-loop (failure of two feedwater control valves) and single-loop (failure of one feedwater control valve) cases demonstrates that the Unit 1 single-loop failure case with manual rod control is more limiting. The single-loop feedwater flow increase case with manual rod control produces the largest reactivity feedback, and therefore results in the greatest power increase. A turbine trip, which results in a reactor trip, is actuated when the steam generator water level in the affected steam generator(s) reaches the high-high water level setpoint.

The cases initiated at hot zero-power conditions are less limiting than the hot zero-power steam line break analysis. Therefore, the results of this case are not presented.

For all excessive feedwater flow cases, continuous addition of cold feedwater is prevented by automatic closure of all feedwater control and isolation valves, closure of all feedwater bypass valves, a trip of the feedwater pumps and a turbine trip on high-high steam generator water level. In addition, the feedwater pump discharge isolation valves will automatically close upon receipt of the feedwater pump trip signal.

Following turbine trip, the reactor will automatically be tripped, either directly due to the turbine trip or due to one of the reactor trip signals discussed in LR subsection 2.8.5.2.1 (Loss of External Electrical Load and/or Turbine Trip). With the rod control system in automatic mode, the control rods would be inserted at the maximum rate following the turbine trip, and the resulting transient would not be limiting in terms of peak RCS pressure.

Table 2.8.5.1.1.2.2-2 shows the time sequence of events for the limiting (Unit 1) single-loop, full-power feedwater flow increase transient with manual rod control and Table 2.8.5.1.1.2.2-3 provides the results. Figures 2.8.5.1.1.2.2-1 through 2.8.5.1.1.2.2-4 show the transient

responses of various system parameters for the limiting (Unit 1) single-loop feedwater flow increase initiated from full-power conditions with manual rod control.

2.8.5.1.1.2.2.5 Conclusions

The decrease in feedwater temperature transient due to the opening of a condensate bypass valve diverting flow around the low-pressure feedwater heaters shows that the DNBRs encountered are above the safety analysis limit value and that the core average thermal power does not exceed a value that results in exceeding the kW/ft limit corresponding to fuel centerline melting at the core hot spot. Therefore, no fuel damage is predicted and all applicable acceptance criteria are satisfied for CPNPP Units 1 and 2.

For the excessive increase in feedwater flow event, the results show that the DNBRs encountered are above the safety analysis limit value and that the core average thermal power does not exceed a value that results in exceeding the kW/ft limit corresponding to fuel centerline melting at the core hot spot. Therefore, no fuel damage is predicted and all applicable acceptance criteria are satisfied for CPNPP Units 1 and 2.

2.8.5.1.1.2.2.6 References

1. WCAP-11397, "Revised Thermal Design Procedure," April 1989.
2. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

2.8.5.1.1.2.3 Increase in Steam Flow

2.8.5.1.1.2.3.1 Introduction

An excessive load increase incident is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor 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 15- to 100-percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor trip system. If the load increase exceeds the capability of the reactor control system, the transient would be terminated in sufficient time to prevent the DNB design basis from being violated. This incident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam bypass control system, or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals, such as a high reactor coolant temperature indicates a need for steam dump. A single controller malfunction will not cause steam dump valves to open; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred. For all cases, the plant rapidly reaches a stabilized condition at a higher power level. Normal plant operating procedures would be followed to reduce power. The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip

may not occur for some cases, and the plant will reach a new equilibrium condition at a higher power level corresponding to the increase in steam flow. Protection against an excessive load increase incident, if necessary, is provided by the following reactor trip signals:

- Overpower N-16
- Overtemperature N-16
- Power range high neutron flux

2.8.5.1.1.2.3.2 Input Parameters, Assumptions, and Acceptance Criteria

An evaluation was performed to show that the DNB design basis is satisfied for the excessive load increase incident. Key aspects of the evaluation are provided as follows.

- The RTDP (Reference 1) was applied. Initial reactor power, RCS pressure, and RCS temperature were assumed to be at their nominal values, consistent with steady-state full-power operation. MMF was also assumed. Uncertainties in initial conditions were accounted for in the safety analysis DNBR limit value as described in Reference 1.
- The evaluation was performed for a step-load increase of 10-percent steam flow from 100-percent of core power.
- This event was evaluated for both automatic and manual rod control.
- The excessive load increase incident was evaluated for both minimum and maximum reactivity feedback conditions.

Based on its frequency of occurrence, the excessive load increase incident is considered to be a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. The following items summarize the acceptance criteria associated with this event:

- The critical heat flux should not be exceeded. This is met by demonstrating that the minimum DNBR does not go below the safety analysis limit value at any time during the transient.
- Pressures in the RCS and MSS should be maintained below 110 percent of the respective design pressures.
- The peak linear heat generation rate (expressed in kW/ft) should not exceed a value that would cause fuel centerline melt.
- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

2.8.5.1.1.2.3.3 Description of Analyses and Evaluations

Given the non-limiting nature of this event with respect to the safety analysis DNBR criterion, an explicit analysis was not performed. Instead, an evaluation of this event was performed. The evaluation process first involved the generation of plant-specific statepoints that are based on generic, conservative conditions that bound the conditions that are expected to occur following a 10-percent load increase incident. Each statepoint consists of post-incident conditions of core power, reactor vessel average temperature, and RCS pressure. The statepoints were compared to the conditions corresponding to operation at the safety analysis DNBR limit (core thermal limits) to ensure that the DNBR limit was not violated. The following three cases (statepoints) were examined in the evaluation:

- Reactor in manual rod control mode with minimum reactivity feedback
- Reactor in manual rod control mode with maximum reactivity feedback
- Reactor in automatic rod control mode (independent of the assumed reactivity feedback)

2.8.5.1.1.2.3.4 Results

The evaluation confirmed that for an excessive load increase incident at CPNPP Units 1 and 2, the minimum DNBR during the transient will not go below the safety analysis limit value, and the peak linear heat generation will not exceed the limit value, thus demonstrating that the applicable acceptance criteria for critical heat flux and fuel centerline melt are met. Following the initial load increase, the plant should reach a stabilized condition. With respect to peak pressures in the RCS and MSS, the excessive load increase incident is bounded by the loss-of-electrical-load/turbine-trip event because the loss-of-electrical-load/turbine-trip event results in a severe mismatch between the primary-side power and the secondary-side power. The analysis of the loss-of-electrical-load/turbine-trip event is described in LR subsection 2.8.5.2.1.

In addition, no adverse conditions are generated as a result of this event that would lead to a more serious plant condition without other faults occurring independently. All applicable acceptance criteria are therefore met for CPNPP Units 1 and 2.

2.8.5.1.1.2.3.5 References

1. WCAP-11397, "Revised Thermal Design Procedure," April 1989.

2.8.5.1.1.2.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a steam generator relief or safety valve event is more commonly referred to as a credible steam line break. It is always bounded by the analysis of the large steam line break (referred to as the hypothetical steam line break) presented in FSAR Section 15.1.5. The hypothetical steam line break is a Condition IV event that is analyzed to Condition II acceptance criteria. The credible steam line break is a Condition II event. Since the more severe Condition IV event is shown to meet the more restrictive Condition II

acceptance criteria, it can be concluded that the credible steam line break event also meets the Condition II acceptance criteria. As such, no explicit analysis of the credible steam line break has been performed. The analyses documented in LR subsections 2.8.5.1.2.2.1 and 2.8.5.1.2.2.2 demonstrate that all applicable acceptance criteria are met for the hypothetical steam line break and, subsequently, all acceptance criteria are met for the credible steam line break.

2.8.5.1.1.3 Conclusions

The analyses of the excess heat removal events described above have been reviewed and it is concluded that the analyses have adequately accounted for operation of the plant at the proposed uprated power level and were performed using acceptable analytical models. It is further concluded that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, Luminant Power has concluded that the plant will continue to meet the requirements of GDCs -10, -15, -20, and -26 following implementation of the proposed SPU. Therefore, the proposed SPU is acceptable with respect to the events stated.

Table 2.8.5.1.1.2.1-1			
Decrease in Feedwater Temperature Cases Analyzed			
Case	Power Level	Initial Nominal Feedwater Temperature (°F)	Rod Control
1	HFP	390.00	Manual
2	HFP	390.00	Automatic
3	HFP	450.30	Manual
4	HFP	450.30	Automatic

Table 2.8.5.1.1.2.1-2	
Time Sequence of Events – Decrease in Feedwater Temperature (High Nominal Feedwater Temperature, HFP, Manual Rod Control)	
Event	Time (Seconds)
Sudden Step Decrease in Feedwater Temperature (Event Initiation)	0.01
Overpower N-16 Setpoint Reached	39.75
Reactor Trip (from Overpower N-16 setpoint)	41.75
Peak Core Thermal Power Occurs	42.10
Minimum DNBR Occurs	42.25
Turbine Trip (from Reactor Trip)	43.75
End of Run	51.75

Table 2.8.5.1.1.2.1-3	
Results – Decrease in Feedwater Temperature (High Nominal Feedwater Temperature, HFP, Manual Rod Control)	
Minimum DNBR	1.90
Peak Core Thermal Power	118.5%

Table 2.8.5.1.1.2.2-1				
Increase in Feedwater Flow Cases Analyzed				
Case	Power Level	Failure	Affected Loop(s)	Rod Control
1	HFP	FCV	Loop 1	Manual
2	HFP	FCV	Loop 1	Automatic
3	HFP	FCV	Loops 1 and 2	Manual
4	HFP	FCV	Loops 1 and 2	Automatic
5	HZP	FCV	Loop 1	Manual
6	HZP	FCV	Loops 1 and 2	Manual

Table 2.8.5.1.1.2.2-2	
Time Sequence of Events – Increase in Feedwater Flow (HFP, Single-Loop, Manual Rod Control)	
Event	Time (Seconds)
One Feedwater Control Valve Fails Full-Open (Event Initiation)	0.01
Minimum DNBR Occurs	34.25
High-High Steam Generator Level Trip Setpoint Reached	41.94
Turbine Trip (from High-High SG Level Trip)	44.34
Reactor Trip (from Turbine Trip)	46.34
Feedwater Isolation Valves Close (from High-High SG Level Trip)	52.94 ⁽¹⁾
End of Run	200.00
Note:	
1. This includes a 0.1-second delay for valve closure.	

Table 2.8.5.1.1.2.2-3	
Results – Increase in Feedwater Flow (HFP, Single-Loop, Manual Rod Control)	
Minimum DNBR	2.10

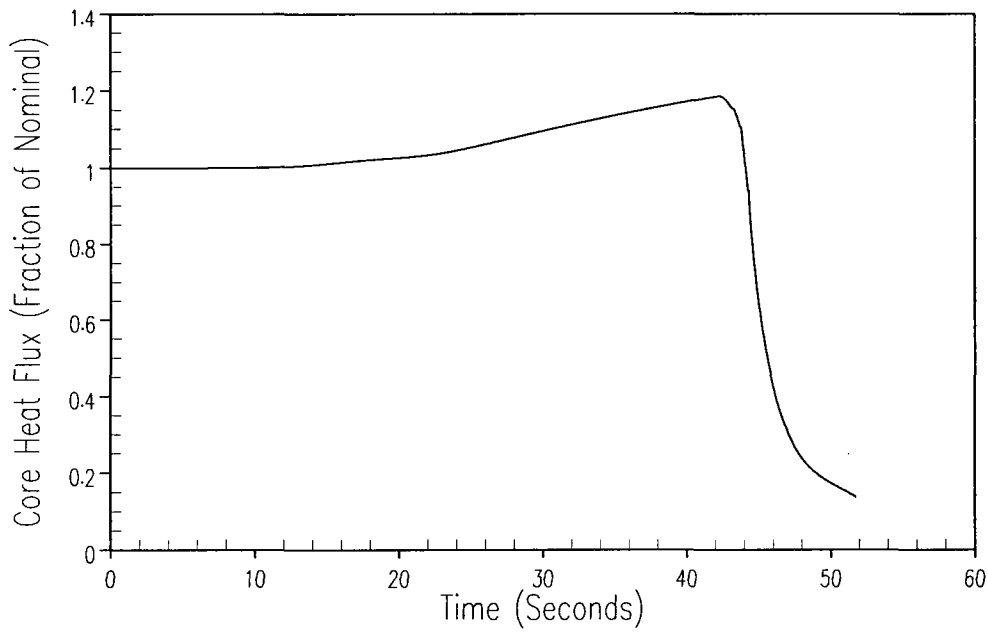
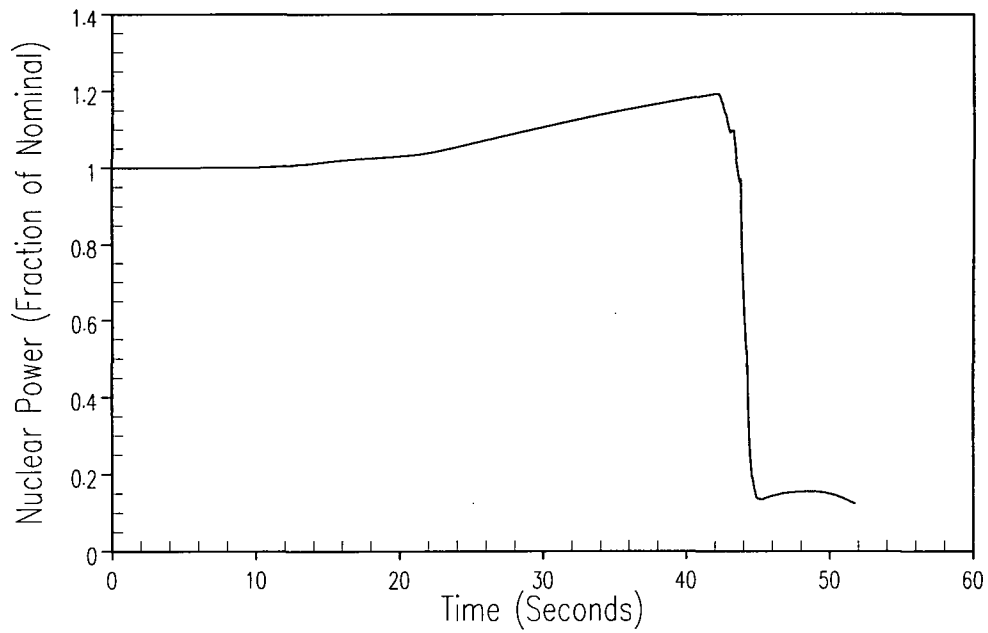


Figure 2.8.5.1.1.2.1-1 Decrease in Feedwater Temperature at Full Power – Manual Rod Control – Nuclear Power and Core Heat Flux Versus Time

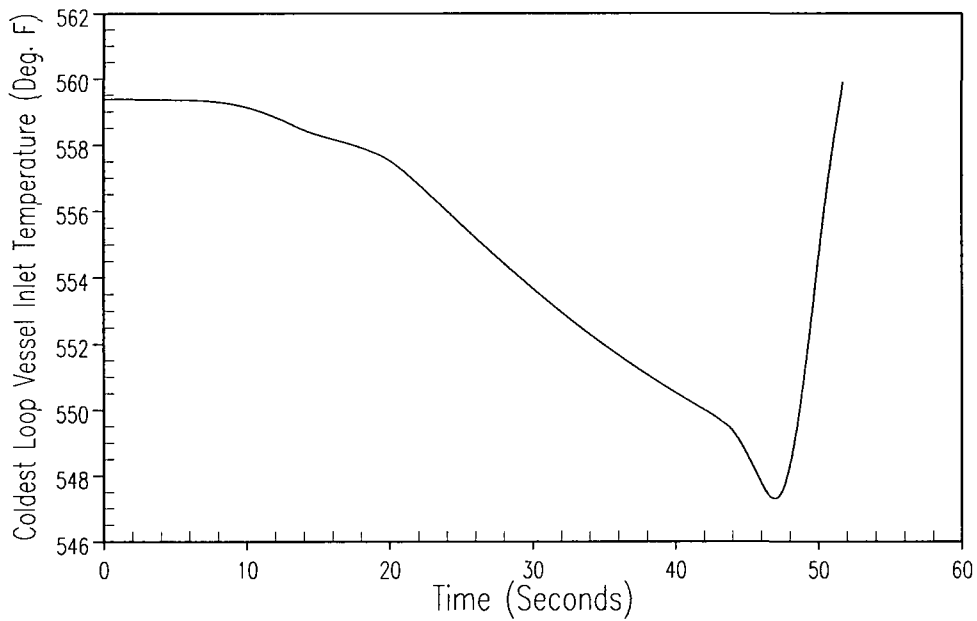
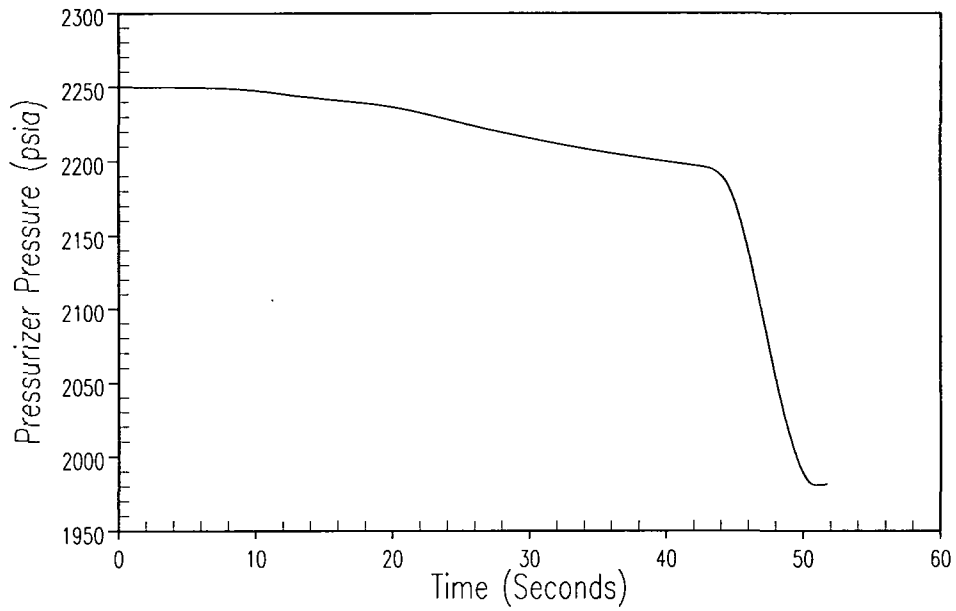


Figure 2.8.5.1.1.2.1-2 Decrease in Feedwater Temperature at Full Power – Manual Rod Control – Pressurizer Pressure and Coldest Vessel Inlet Temperature Versus Time

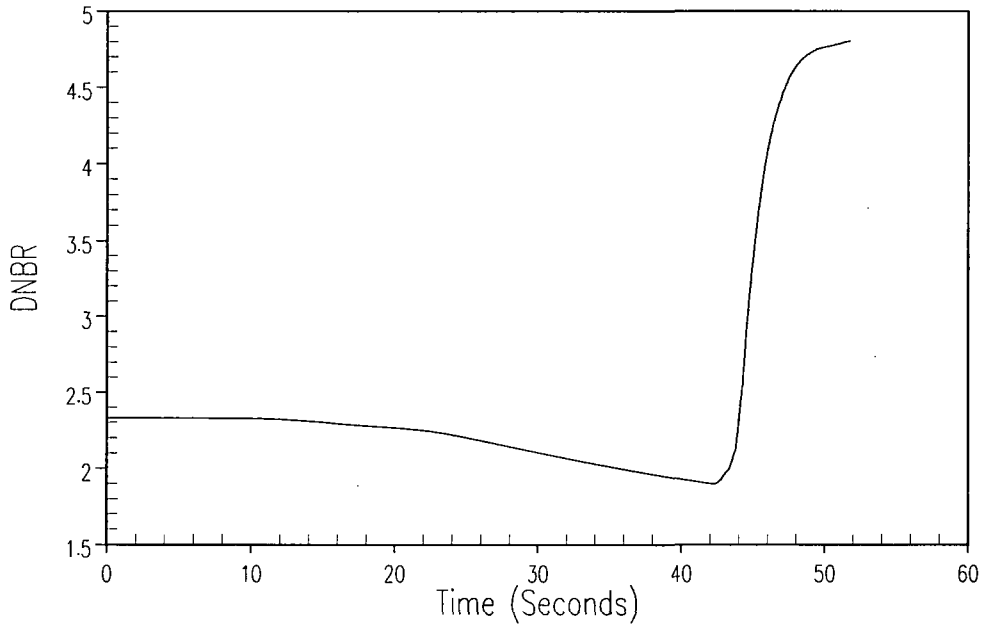
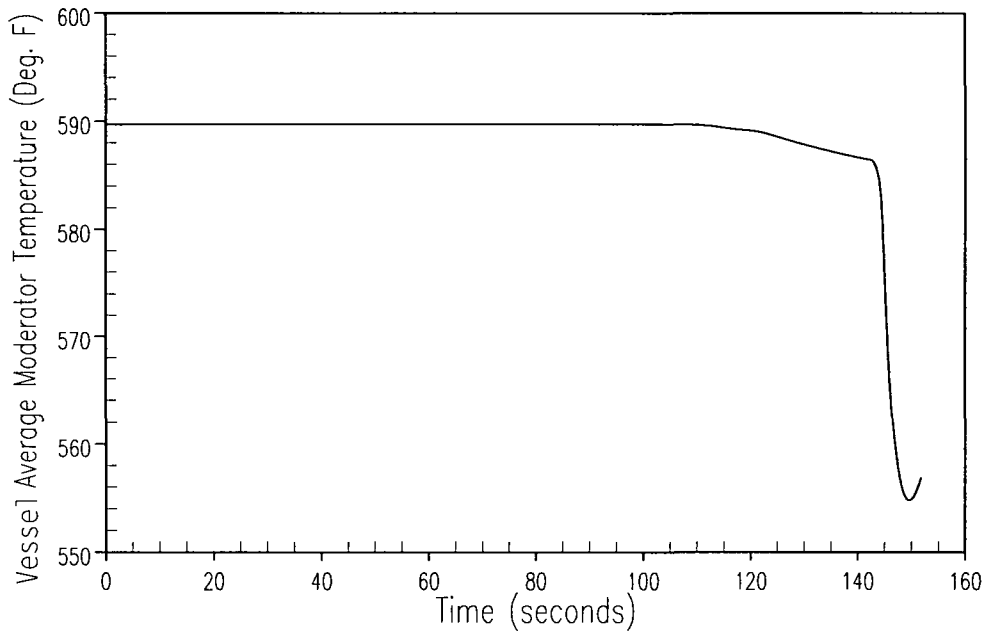
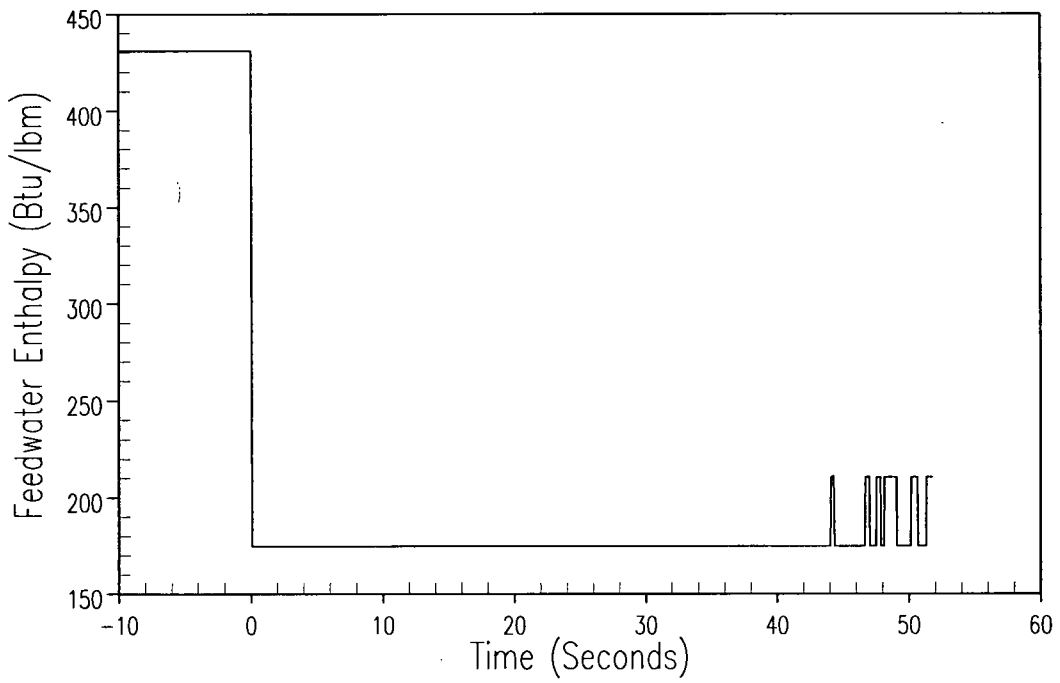


Figure 2.8.5.1.1.2.1-3 Decrease in Feedwater Temperature at Full Power – Manual Rod Control – Vessel Average Moderator Temperature and DNBR Versus Time



(Note that the x-axis scale begins at -10 seconds so that the enthalpy step reduction at time zero is visible on the plot)

Figure 2.8.5.1.1.2.1-4 Decrease in Feedwater Temperature at Full Power – Manual Rod Control – Feedwater Enthalpy Versus Time

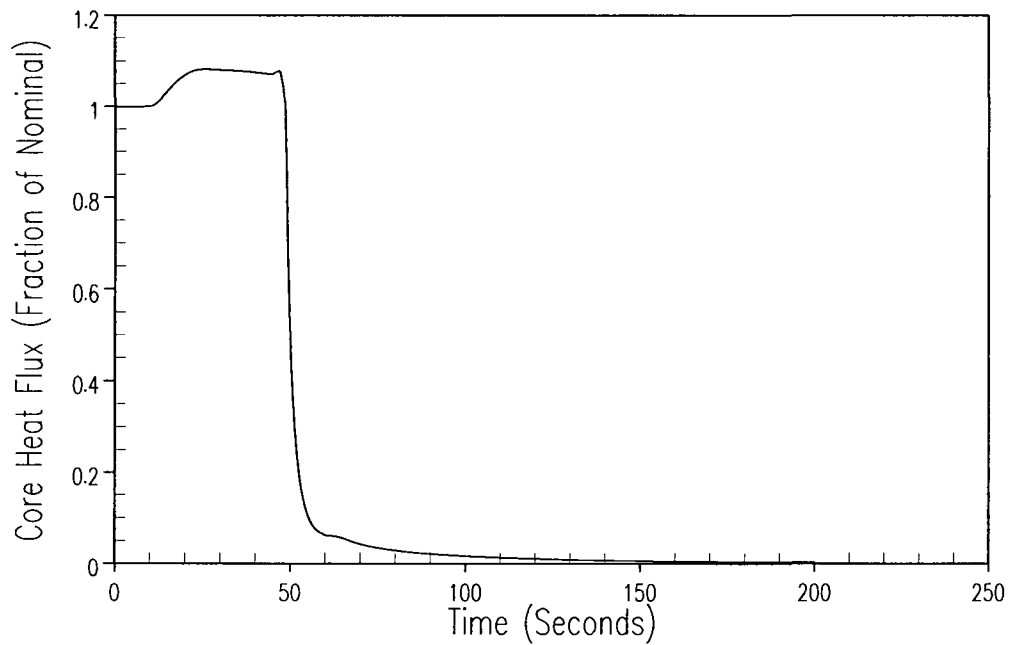
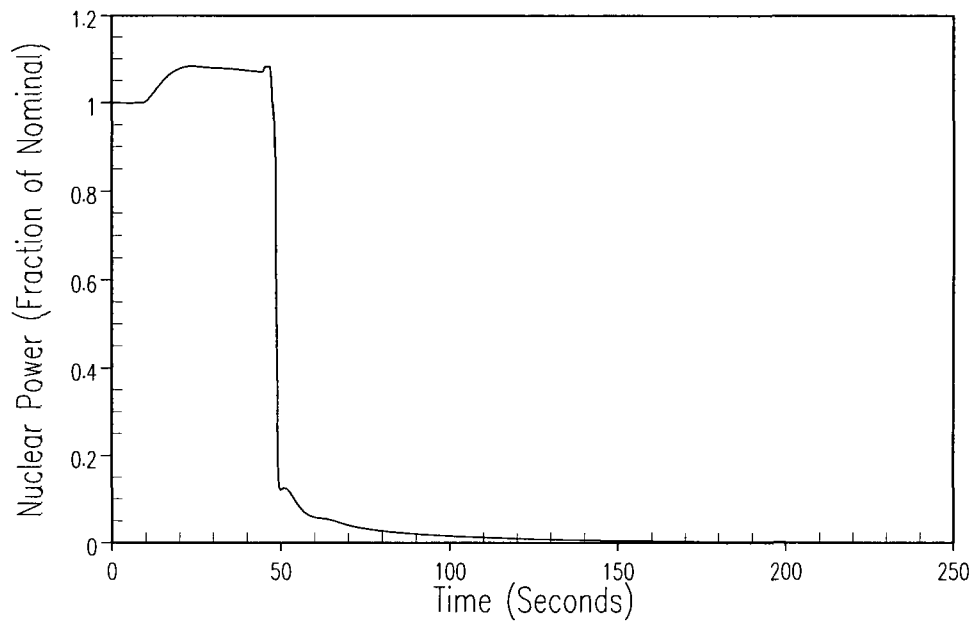


Figure 2.8.5.1.1.2.2-1 Increase in Feedwater Flow at Full Power – Manual Rod Control – Nuclear Power and Core Heat Flux Versus Time

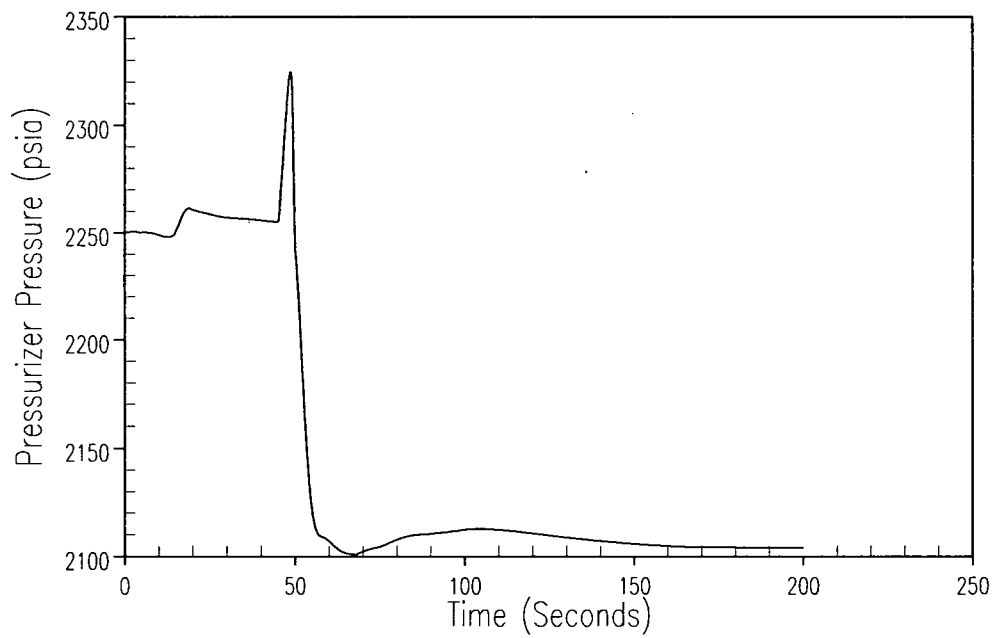
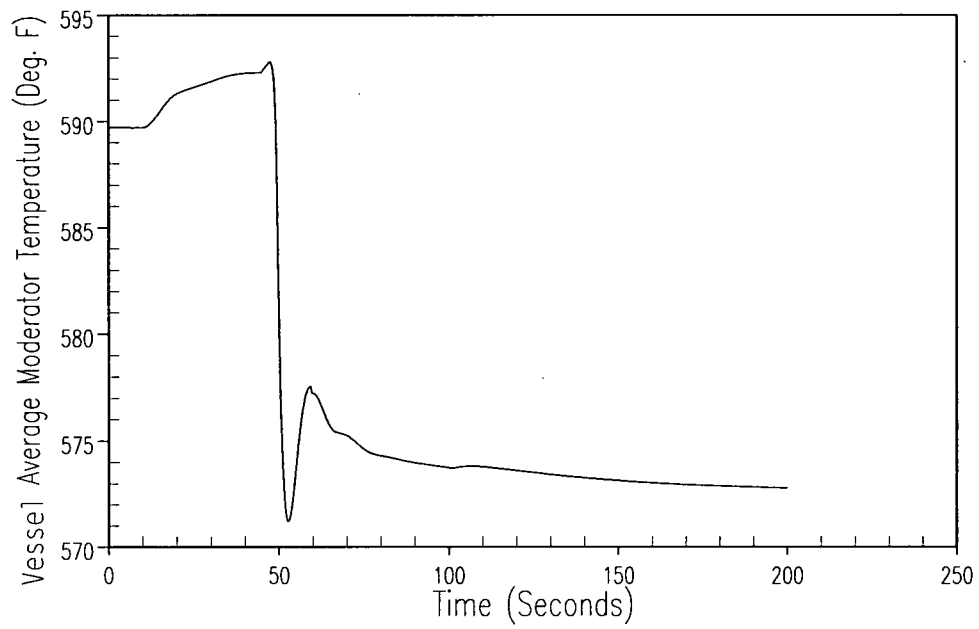


Figure 2.8.5.1.1.2.2-2 Increase in Feedwater Flow at Full Power – Manual Rod Control – Vessel Average Moderator Temperature and Pressurizer Pressure Versus Time

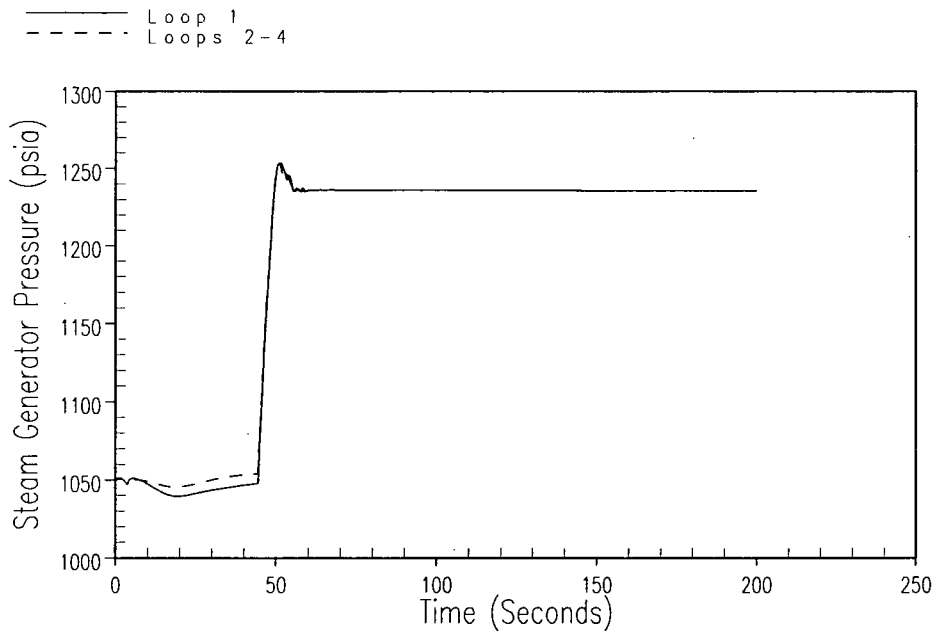
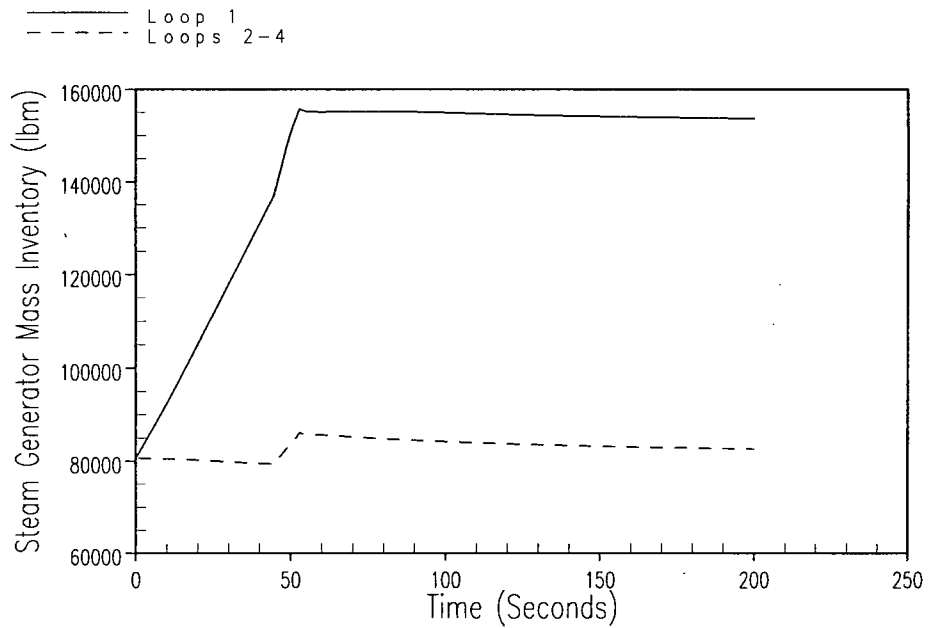


Figure 2.8.5.1.1.2.2-3 Increase in Feedwater Flow at Full Power – Manual Rod Control – Steam Generator Mass Inventory and Steam Generator Pressure Versus Time

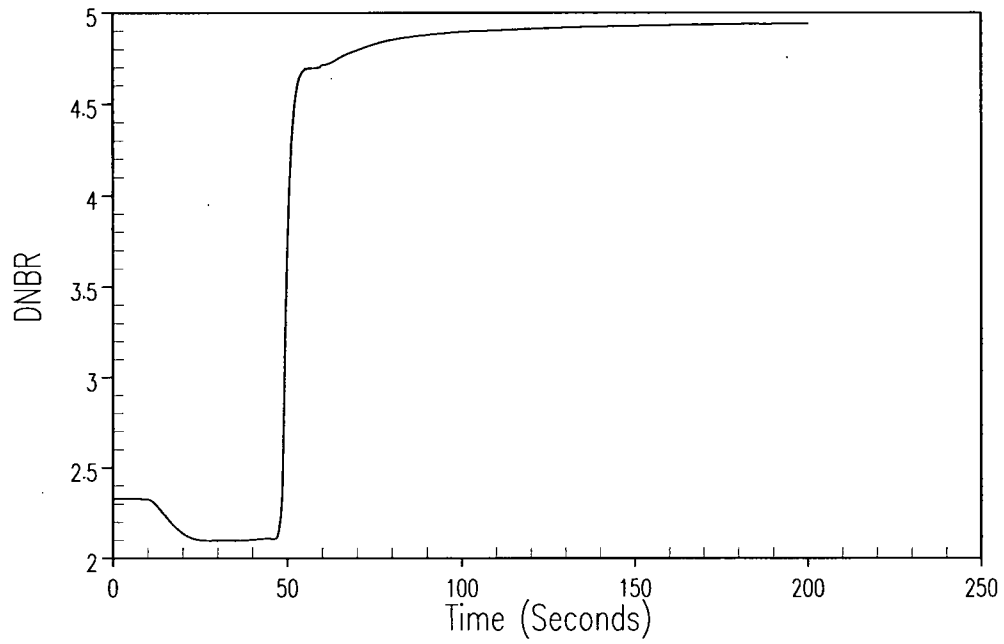


Figure 2.8.5.1.1.2.2-4 Increase in Feedwater Flow at Full Power – Manual Rod Control – DNBR Versus Time

2.8.5.1.2 Steam System Piping Failures Inside and Outside Containment

2.8.5.1.2.1 Regulatory Evaluation

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor trip and safety systems are actuated to mitigate the transient.

The review covered:

- The postulated initial core and reactor conditions
- The methods of thermal and hydraulic analyses
- The sequence of events
- The assumed responses of the reactor coolant and auxiliary systems
- The functional and operational characteristics of the reactor trip system (RTS)
- Core power excursion due to power demand caused by excessive steam flow
- Variables influencing neutronics

The results of the transient analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained
- GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core
- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specific conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized
- GDC-35, insofar as it requires that the reactor coolant system (RCS) and associated auxiliaries be designed to provide abundant emergency core cooling

Current Licensing Basis

As noted in the Final Safety Analysis Report (FSAR) Section 3.1, the design bases of CPNPP are measured against the Nuclear Regulatory Commission (NRC) GDC for Nuclear Power Plants, Appendix A to 10 CFR 50. The adequacy of the Comanche Peak Nuclear Power Plant (CPNPP) design relative to the general design criteria is discussed in the FSAR Sections 3.1.3 and 3.1.4. As stated in FSAR Sections 15.0.1.3 and 15.0.1.4, a steam system piping failure, which leads to an increase in heat removal from the RCS, may be either a Condition III (minor break) event or a Condition IV (major break) event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-27, Combined Reactivity Control Systems Capability, is described in FSAR Section 3.1.3.8.

The facility is provided with means of making the core subcritical and maintaining it at that level under any anticipated conditions and with an appropriate margin for contingencies. These means are discussed in detail in FSAR Chapters 4 and 9. Combined use of the rod cluster control assemblies (RCCAs) and the chemical shim permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. Upon trip, the single highest worth control cluster is assumed to be stuck full-out.

- GDC-28, Reactivity Limits, is described in FSAR Section 3.1.3.9.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion using control rods are limited to values that prevent rupture of the RCS boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The maximum positive reactivity insertion rates for the withdrawal of RCCAs and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCAs and of the chemical and volume control system (CVCS). Technical Specifications on shutdown margin and on RCCAs insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses in Chapter 15 of the FSAR. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in FSAR Section 4.3. The capability of the CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 15 of the FSAR.

Assurance of core cooling capability following Condition IV accidents, such as rod ejections, steam line breaks, and similar accidents, is given by keeping the RCPB stresses within faulted condition limits as specified by applicable American Society of

Mechanical Engineers (ASME) codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of necessary safety features.

- GDC-31, Fracture Prevention of Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.4.2.

Close control is maintained over material selection and fabrication for the RCS to ensure that the boundary behaves in a non-brittle manner. Those RCS materials that are exposed to the coolant are corrosion resistant, stainless steel, or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR Part 50, Appendix G.

The fabrication and quality control techniques used in the fabrication of the RCS are equivalent to those used for the reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generator are governed by the requirements of ASME Codes.

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Appendix G, Protection Against Non-Ductile Failure. The approach specifies that allowed stress intensity factors for all vessel operating conditions shall not exceed the reference stress intensity factor (KIR) for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material (RT_{NDT}) due to irradiation.

- GDC-35, Emergency Core Cooling, is described in FSAR Section 3.1.4.6.

An ECCS is provided to cope with any loss-of-coolant accident (LOCA) in the plant design basis. Abundant cooling water is available in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to ensure that clad metal-water reaction is limited to less than 1 percent. Adequate design provisions are made to ensure performance of the required safety functions even with a single failure.

FSAR Section 15.1.5.1 states the following:

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. The energy removal from the RCS 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 high power peaking factors that exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by injection of boric acid delivered by the safety injection system.

FSAR Table 15.1-2 lists the equipment required in the recovery from a high-energy line rupture. Not all equipment is required for any one particular break, since the requirements vary depending upon postulated break locations and details of balance of plant design and pipe rupture criteria as discussed elsewhere in the FSAR. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in FSAR Section 3.6.

FSAR Section 15.1.5.4 concludes that the DNB design basis is met for any steam pipe rupture assuming that the most reactive RCCA is stuck in its fully withdrawn position.

2.8.5.1.2.2 Technical Evaluation

2.8.5.1.2.2.1 Steam System Piping Failure at Hot Zero Power

2.8.5.1.2.2.1.1 Introduction

The steam release arising from a major rupture of a main steam pipe will result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The increased energy removal from the RCS causes a reduction of reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (MTC), the cooldown results in a positive reactivity insertion and subsequent reduction in core shutdown margin. 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 pipe rupture is a concern primarily because of the high-power peaking factors that would exist assuming the most-reactive RCCA is stuck in its fully withdrawn position.

The major rupture of a main steam pipe is the most limiting cooldown transient. It is analyzed at hot zero-power (HZIP) conditions with no decay heat (decay heat would retard the cooldown, thus reducing the return to power). A detailed discussion of this transient with the most limiting break size (a double-ended rupture) is presented below.

The primary design features which provide protection for steam pipe ruptures are:

- Actuation of the safety injection (SI) system from any of the following:
 - Two-out-of-four low pressurizer pressure signals
 - Two-out-of-three low steam line pressure signals in any loop
 - Two-out-of-three high-1 containment pressure signals
- Reactor trip can be actuated from overpower (neutron flux and N-16) or upon the receipt of a safety injection signal.
- Redundant isolation of the main feedwater lines to prevent sustained high feedwater flow that would cause additional cooldown. In addition to the normal control action which closes the main feedwater control valves, a safety injection signal rapidly closes all feedwater control valves and backup feedwater isolation valves, and trips the main

feedwater pumps. A trip of the main feedwater pumps results in automatic closure of the respective pump discharge isolation valves.

- Closure of the fast-acting main steam isolation valves (MSIVs) on the following:
 - Two-out-of-three high-2 containment pressure signals
 - Two-out-of-three low steam line pressure signals in any loop (above Permissive P-11)
 - Two-out-of-three high negative steam pressure rate signals in any loop (below Permissive P-11)

For any break (in any location), no more than one steam generator would experience an uncontrolled blowdown, even if one of the MSIVs fails to close. For breaks downstream of the MSIVs, closure of all MSIVs completely terminates the blowdown of all steam generators. The valves on all steam lines are closed to isolate the steam generators. Thus, even with the worst possible break location (that is, upstream of an MSIV), only one steam generator can blow down, minimizing the potential steam release and resultant RCS cooldown. The remaining steam generators would still be available for dissipation of decay heat after the initial transient is over.

Following blowdown of the faulted steam generator, the unit can be brought to a stabilized hot standby condition through control of the auxiliary feedwater (AFW) flow and safety injection flow as prescribed by plant operating procedures. The operating procedures call for operator action to limit RCS pressure and pressurizer level by terminating safety injection flow and to control steam generator level and RCS coolant temperature using the AFW system (AFWS).

2.8.5.1.2.2.1.2 Input Parameters, Assumptions, and Acceptance Criteria

The following summarizes the major input parameters and/or assumptions used in the analysis of the main steam line rupture event at HZP conditions:

- HZP conditions were modeled with four loops in service, both with and without offsite power available.
- Cases were run modeling both the Unit 1 Model $\Delta 76$ steam generators and the Unit 2 Model D-5 steam generators.
- A 1.388 ft² break size was analyzed for both types of steam generators. This break size corresponds to the flow area of the flow restrictor built into the steam exit nozzle of each steam generator. The assumed steam generator tube plugging level was 0 percent.
- All control rods were inserted except the most reactive RCCA, which was assumed to be stuck out of the core.

-
- The shutdown margin was 1.30-percent $\Delta k/k$.
 - The safety injection system and the accumulators were modeled with their minimum Technical Specification boron concentrations.
 - The low steam line pressure signal was credited for safety injection actuation.

A major break in a steam system pipe is classified as a Condition IV event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. Minor secondary system pipe breaks are classified as ANS Condition III events. The major steam system pipe break was analyzed to meet the more restrictive Condition II acceptance criteria and, therefore, bounds the less severe minor secondary system pipe break accident.

Pressure limits of the primary and secondary systems are not challenged because primary and secondary pressures decrease from their initial values during the transient. The only criterion that has the potential to be challenged during this event is that concerning the critical heat flux not being exceeded. The analysis demonstrates that this criterion is met by showing that the minimum DNBR does not go below the limit value at any time during the transient.

2.8.5.1.2.2.1.3 Description of Analyses and Evaluations

A detailed analysis using the RETRAN computer code (Reference 1) was performed in order to determine the plant transient conditions following a main steam line break. The code models the core neutron kinetics, RCS, pressurizer, steam generators, safety injection system, and the AFWs. The code computes pertinent variables, including the core heat flux, RCS temperature, and pressure. A detailed core analysis was then performed using the ANC code (Reference 2) to determine if the RETRAN-predicted reactivity feedback model is conservative. The core models developed in ANC were then used as input to the detailed thermal and hydraulic digital computer code, VIPRE (Reference 3), to determine if the DNB design basis is met.

2.8.5.1.2.2.1.4 Results

For CPNPP, the most limiting main steam line rupture at HZP case is the Unit 1 case in which offsite power was assumed to be available. The calculated sequence of events for the limiting case is shown in Table 2.8.5.1.2.2.1-1.

Figures 2.8.5.1.2.2.1-1 through 2.8.5.1.2.2.1-5 show the transient results for the most limiting case, a complete severance of a main steam pipe at initial no-load conditions with offsite power available. Since offsite power was assumed to be available, there is full reactor coolant flow.

Should the core be critical at or near zero power when the rupture occurs, the initiation of safety injection via a low steam line pressure signal trips the reactor. Steam release from more than one steam generator is prevented by automatic closure of the MSIVs in the steam lines by low steam line pressure signals.

As shown in Figure 2.8.5.1.2.2.1-4, the core attains criticality with the RCCAs inserted (that is, with the plant shut down assuming one stuck RCCA) before the transient is turned around by boron injected from the ECCS and accumulators.

The results of the major rupture of a main steam pipe event indicate that the DNB design basis is met. The calculated minimum DNBR is well above the limit value. The pressure limits of the primary and secondary systems are not challenged because primary and secondary pressures decrease from their initial values during the transient. Therefore, this event does not adversely affect the core or the RCS, and all applicable acceptance criteria are met.

2.8.5.1.2.2.1.5 References

1. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
2. WCAP-10965, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.
3. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

Table 2.8.5.1.2.2.1-1**Time Sequence of Events – Steam System Piping Failure at Hot Zero Power**

Case	Event	Time (sec)
Unit 1 Double-Ended Rupture (1.388 ft ²) with Offsite Power Available	Steam Line Break Occurs	0.0
	Low Steam Pressure Setpoint Reached in Faulted Loop (Loop 1)	10.5
	Safety Injection and Steam Line Isolation Signals Generated (on low steam pressure)	12.5
	Pressurizer Empties	13.6
	Steam Line and Feedwater Isolation Complete	17.5
	Re-criticality Occurs	25.3
	SI Flow Initiated (Borated Water)	37.5
	Accumulators Inject	93.9
	Peak Core Heat Flux Reached	108.0
	Minimum DNBR Reached	108.3

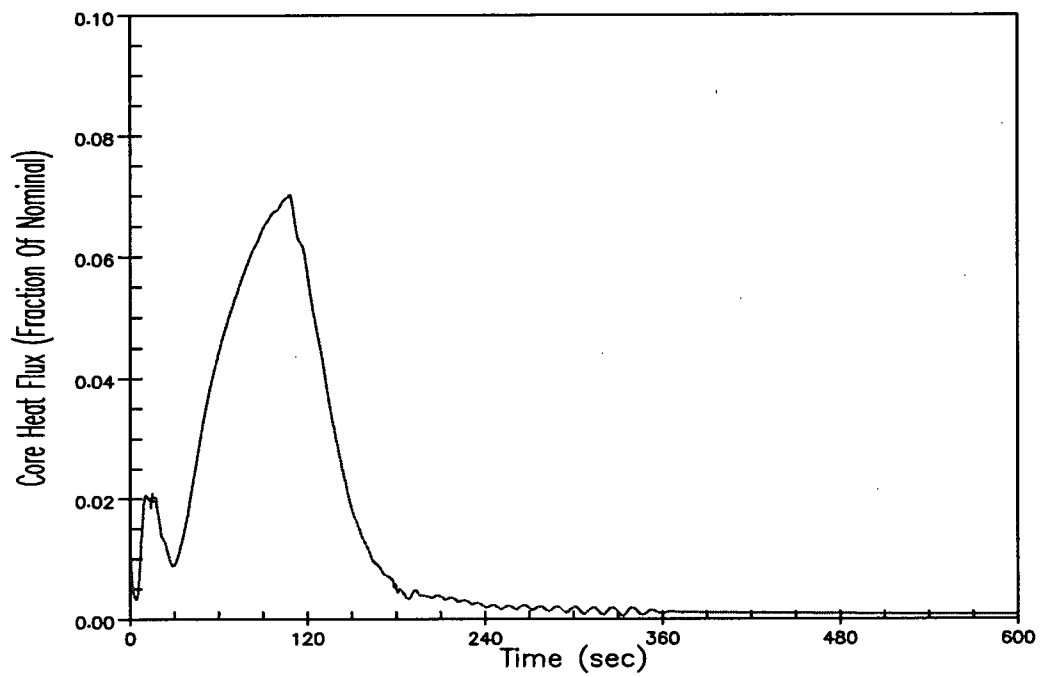
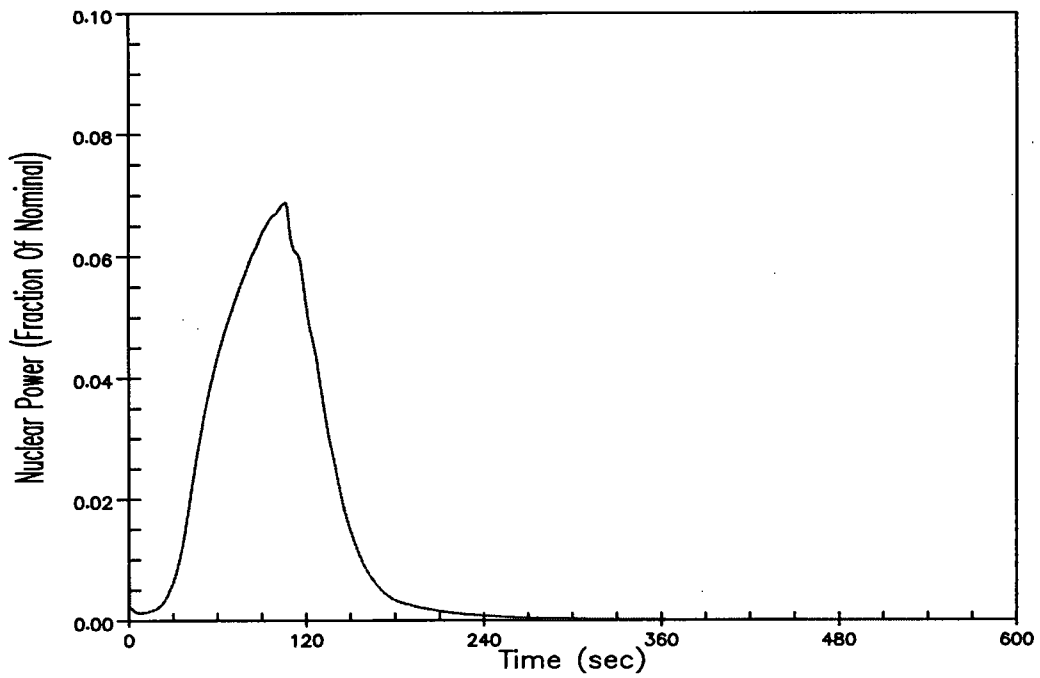


Figure 2.8.5.1.2.2.1-1 Piping Failure at Hot Zero Power – 1.388 ft² Break (with Offsite Power Available) – Nuclear Power and Core Heat Flux Versus Time

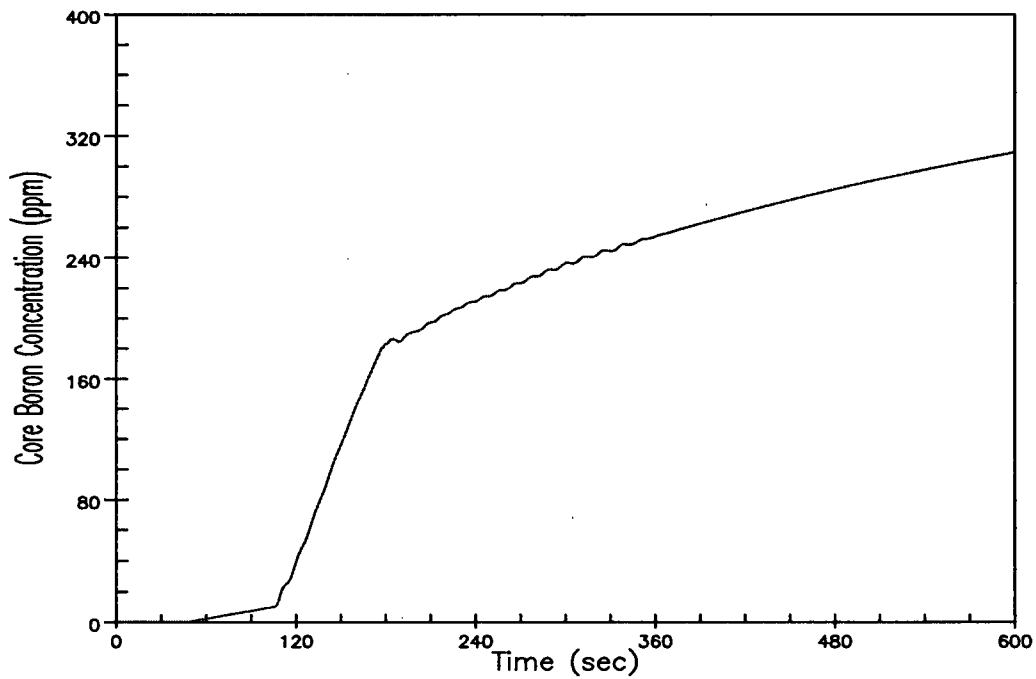
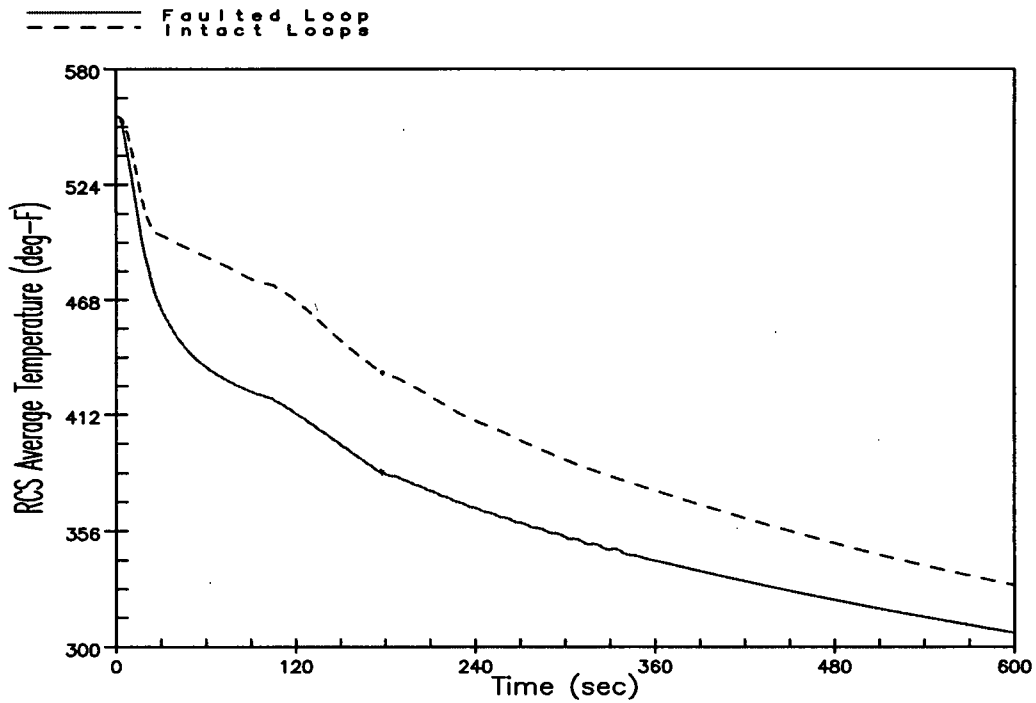


Figure 2.8.5.1.2.2.1-2 Piping Failure at Hot Zero Power – 1.388 ft² Break (with Offsite Power Available) – RCS Average Temperature and Core Boron Concentration Versus Time

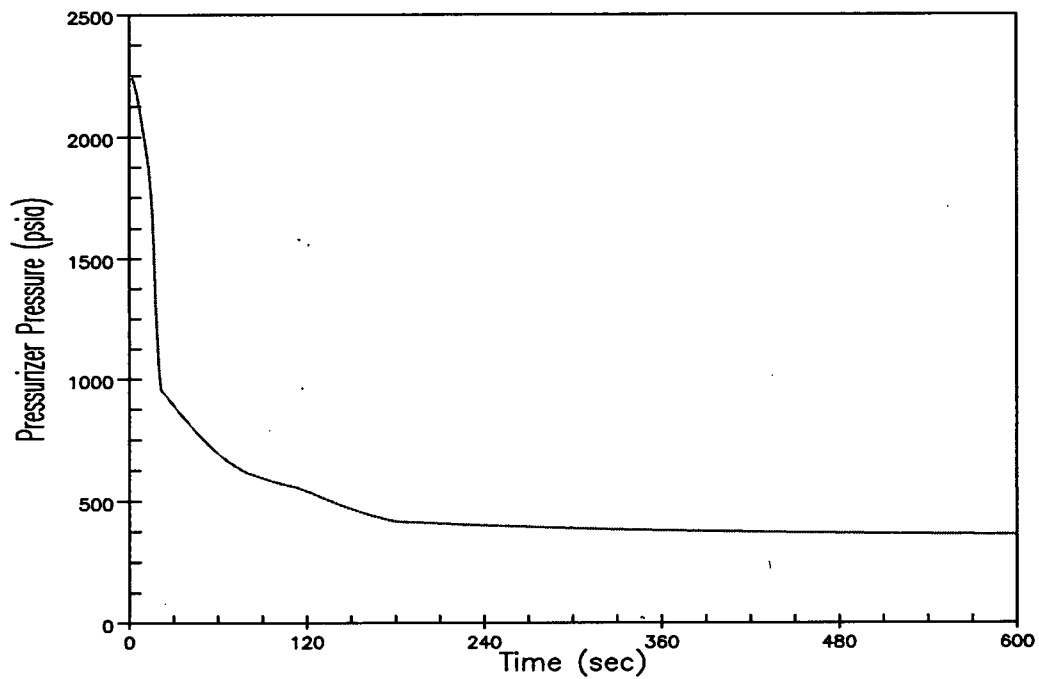
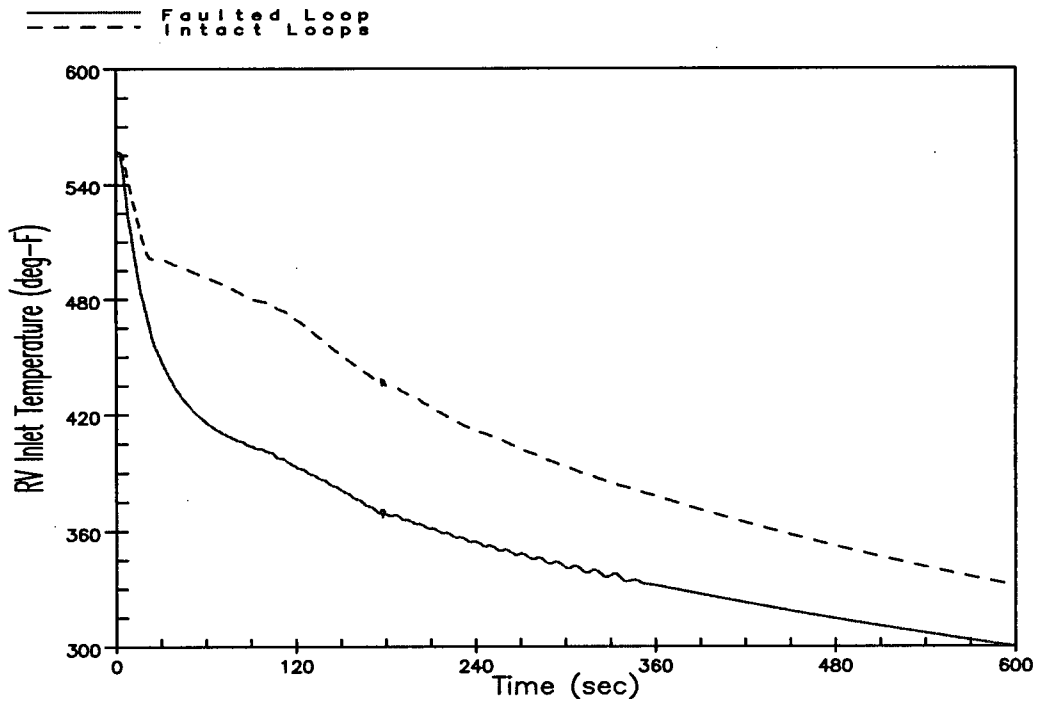


Figure 2.8.5.1.2.2.1-3 Piping Failure at Hot Zero Power – 1.388 ft² Break (with Offsite Power Available) – Reactor Vessel Inlet Temperature and Pressurizer Pressure Versus Time

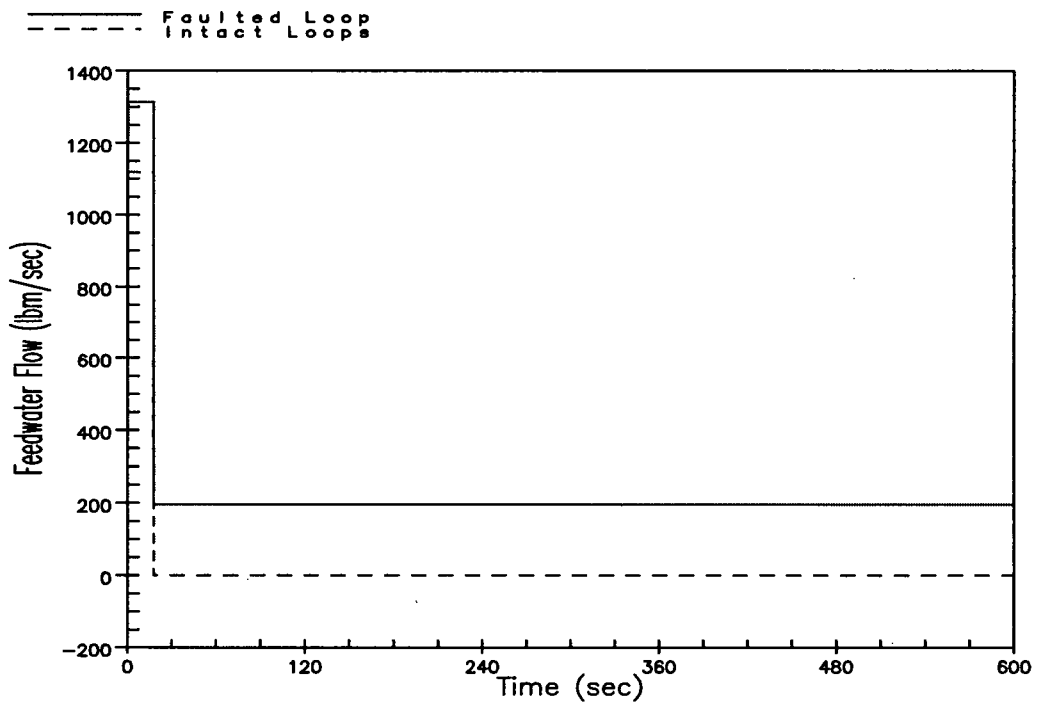
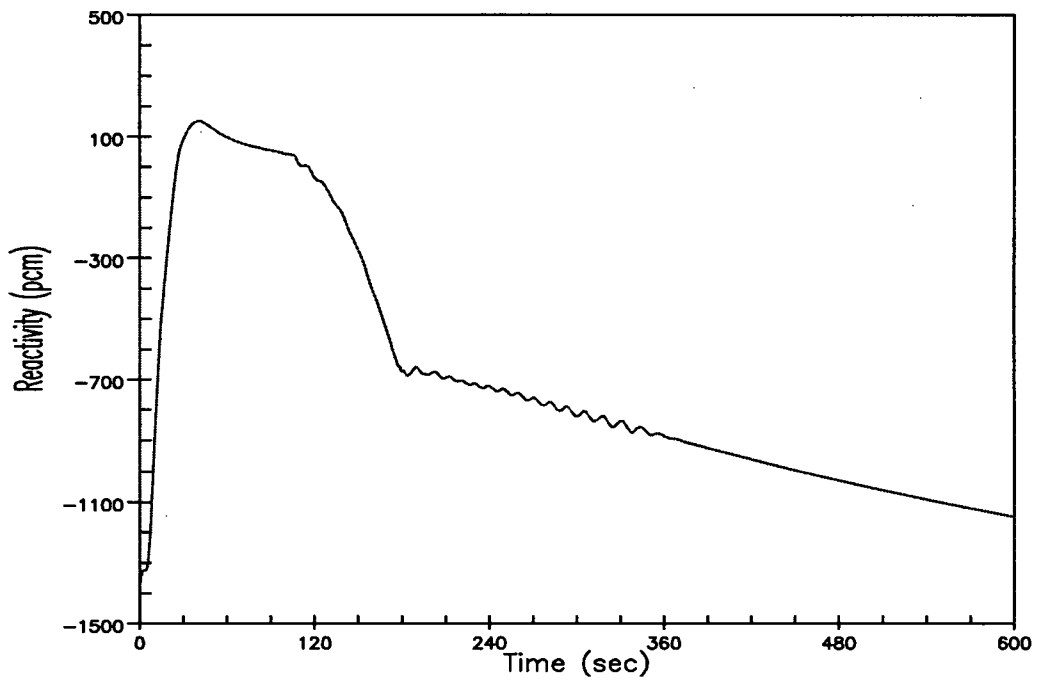


Figure 2.8.5.1.2.2.1-4 Piping Failure at Hot Zero Power – 1.388 ft² Break (with Offsite Power Available) – Reactivity and Feedwater Flow Versus Time

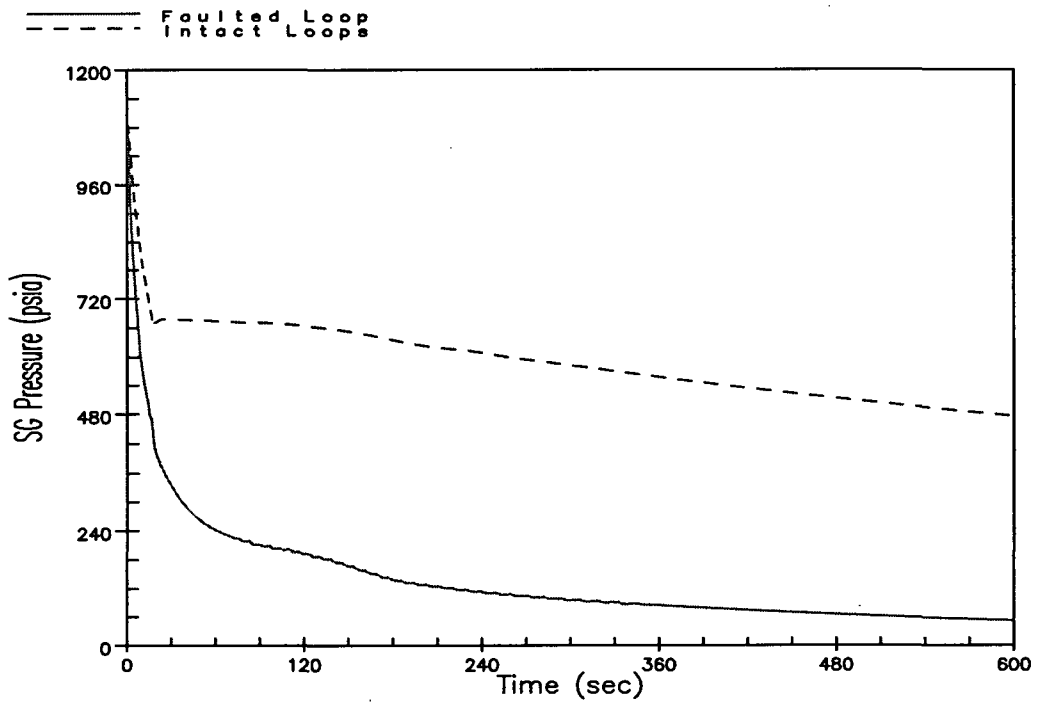
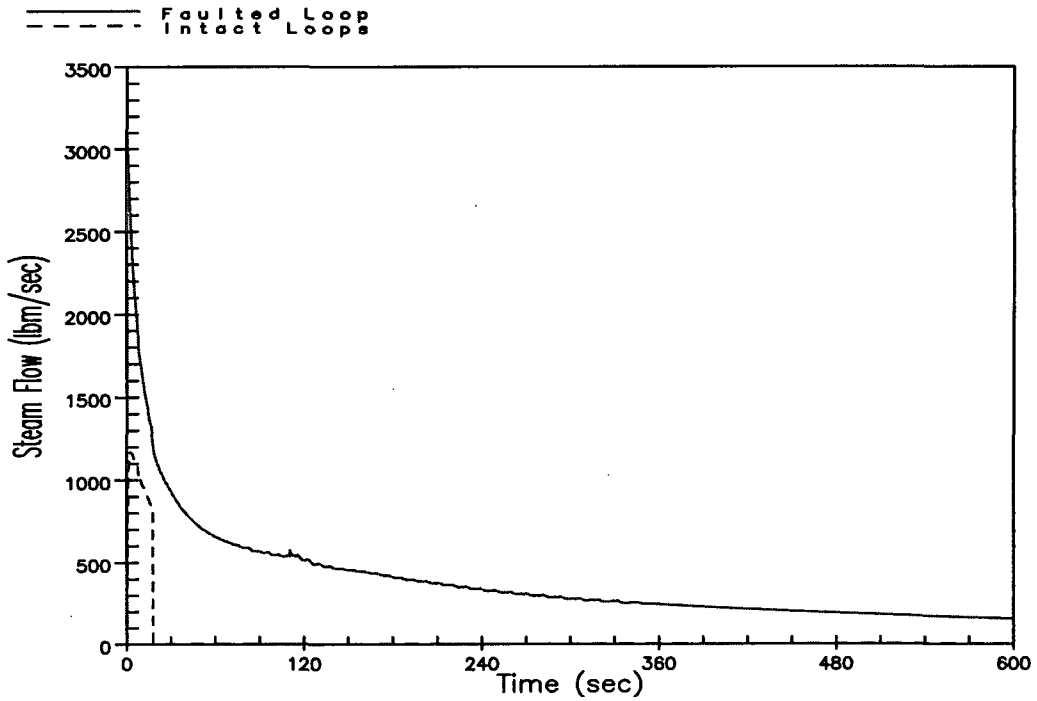


Figure 2.8.5.1.2.2.1-5 Piping Failure at Hot Zero Power – 1.388 ft² Break (with Offsite Power Available) – Steam Flow and Steam Generator Pressure Versus Time

2.8.5.1.2.2.2 Steam System Piping Failure at Full-Power

2.8.5.1.2.2.2.1 Introduction

This section describes the analysis of a steam system piping failure occurring from full-power initial conditions to demonstrate that core protection is maintained prior to, and immediately following, reactor trip. The steam release from a major rupture of a main steam pipe at full power will result in an increase in steam flow that stabilizes at a higher-than-initial flow rate as the steam pressure falls. The increased energy removal from the RCS causes a reduction of reactor coolant temperature and pressure. In the presence of a negative MTC, the cooldown results in a positive reactivity insertion and subsequent increase in power due to the higher steam load. The power increase is ultimately terminated by a reactor trip on either an overpower N-16 signal or a low steam pressure safety injection signal.

2.8.5.1.2.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Limiting transient condition statepoints (a table of critical analysis results for each time step, such as core average heat flux, vessel inlet temperatures, and core pressure) were generated using the Revised Thermal Design Procedure (RTDP) (Reference 1). For RTDP applications, uncertainties on RCS initial conditions (temperature, pressure, and power) are statistically included in the development of the departure from nucleate boiling ratio (DNBR) limit value.

The following summarizes the major input parameters and/or assumptions used in the analysis of the full power main steam line rupture event:

- Initial conditions – The initial core power (3,612 MWt), RCS temperature, and RCS pressure were assumed to be at their nominal steady-state, full-power values.
- RCS average temperature – The full-power RCS T_{avg} range is from 574.2° to 589.2°F. Since the full-power steam line rupture event is a DNB event, assuming a maximum RCS average temperature is limiting. Therefore, an initial RCS average temperature of 589.2°F was assumed.
- RCS flow – Minimum measured RCS flow was assumed. The initial loop flows were assumed to be symmetric.
- Feedwater temperature – The main feedwater analytical temperature range is from 390° to 450.3°F. A nominal feedwater temperature of 450.3°F is more limiting with respect to DNB for this event. Thus, a feedwater temperature of 450.3°F was assumed. For this event, the feedwater flow was set to match the steam flow.
- Break size – A spectrum of break sizes was analyzed to identify the most limiting overpower condition, which is typically identified by the largest break to produce a reactor trip on an overpower N-16 signal. The steam generators (Model Δ 76 for Unit 1 and Model D-5 for Unit 2) have a steam exit nozzle flow restrictor that limits the flow area to 1.388 ft². Therefore, break sizes up to 1.388 ft² were analyzed. In addition, the

largest break size for which there is no reactor trip was examined to determine if it is more limiting with respect to peak power level.

- Reactivity coefficients – The analysis assumed maximum moderator reactivity feedback and minimum Doppler power feedback to maximize the power increase following the break.
- Protection system – This analysis only considers the initial phase of the steam line rupture transient from full power conditions (that is, before and just following reactor trip). Protection in this phase of the transient is provided by reactor trip, if necessary. The primary credited functions for this case are the low steam pressure safety injection signal and the overpower N-16 reactor trip.
- Control systems – No control systems were assumed.

Depending on the size of the break, this event is classified as either a Condition III (infrequent fault) or Condition IV (limiting fault) event. However, the analysis was done to the more conservative Condition II acceptance criteria. Specifically, the acceptance criteria are met by showing that the minimum DNBR does not go below the limit value (1.61) and the peak linear heat rate (kW/ft) does not exceed the fuel melt limit value (22.4 kW/ft) at any time during the transient.

2.8.5.1.2.2.3 Description of Analysis and Evaluations

The analysis of the full-power steam line rupture event for the SPU was performed as follows:

- The RETRAN computer code (Reference 2) was used to calculate the nuclear power, core average heat flux, vessel inlet temperatures, and core pressure transients resulting from the cooldown following the steam line break.
- The core radial and axial peaking factors were determined using the thermal-hydraulic conditions from RETRAN as input to the nuclear core models. The detailed thermal-hydraulic computer code VIPRE (Reference 3) was used to calculate the DNBR for the limiting time during the transient. The DNBR calculations were performed using the WRB-2 DNB correlation and RTDP.

2.8.5.1.2.2.4 Results

The limiting break size from the spectrum of break sizes analyzed is 1.388 ft² for both units, which is the largest possible break size due to the steam generator outlet flow restrictors. This is expected since the reactor trips on an overpower N-16 signal for the entire spectrum of breaks (that is, the low steam pressure setpoint was not reached due to relaxed dynamic compensation), and the largest break that trips on overpower N-16 is typically limiting.

The limiting case between the two units with respect to minimum DNBR is for Unit 2, in which the minimum DNBR is 1.963/2.015 (thimble cell/typical cell) versus a 1.61 limit. With respect to

peak linear heat rate (kW/ft), the limiting case between the two units is for Unit 1, in which the peak kW/ft is 21.6 kW/ft versus a 22.4 kW/ft limit.

The sequence of events for the limiting cases (the 1.388 ft² break) for Unit 1 and Unit 2 are shown in Tables 2.8.5.1.2.2.2-1 and 2.8.5.1.2.2.2-2, respectively. Plots for these cases are provided in Figures 2.8.5.1.2.2.2-1 through 2.8.5.1.2.2.2-4 (Unit 1) and Figures 2.8.5.1.2.2.2-5 through 2.8.5.1.2.2.2-8 (Unit 2).

The DNB design basis and peak linear heat rate limits are met for both units. Therefore, a steam line rupture occurring from full power conditions will not adversely affect the core or RCS.

The results of the analysis performed for the steam system piping failure at full power for the nuclear steam supply system power of 3,628 MWt support the implementation of the SPU. Furthermore, the results and conclusions of this analysis will be confirmed on a cycle-specific basis as part of the normal reload safety evaluation process.

2.8.5.1.2.3 Conclusions

The analyses of steam system piping failure events have been reviewed and Luminant Power has concluded that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. It is further concluded that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a non-brittle manner, the probability of a propagating fracture of the RCPB is minimized, and abundant core cooling will be provided. Based on this, the conclusion is that the plant will continue to meet the requirements of GDCs -27, -28, -31, and -35 following implementation of the proposed SPU. Therefore, the proposed uprate is acceptable with respect to steam system piping failures.

2.8.5.1.2.4 References

1. WCAP-11397, "Revised Thermal Design Procedure," April 1989.
2. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
3. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

Table 2.8.5.1.2.2.2-1	
Time Sequence of Events – Steam System Piping Failure at Full-Power (Unit 1 Core Response – 1.388 ft² break)	
Event	Time (seconds)
Steam Line Ruptures	0.0
Overpower N-16 Reactor Trip Setpoint Reached	11.7
Rods Begin to Drop	13.7
Minimum DNBR Occurs	14.5
Peak Core Heat Flux Occurs	14.5

Table 2.8.5.1.2.2.2-2	
Time Sequence of Events – Steam System Piping Failure at Full-Power (Unit 2 Core Response – 1.388 ft² break)	
Event	Time (seconds)
Steam Line Ruptures	0.0
Overpower N-16 Reactor Trip Setpoint Reached	13.1
Rods Begin to Drop	15.1
Minimum DNBR Occurs	15.5
Peak Core Heat Flux Occurs	15.5

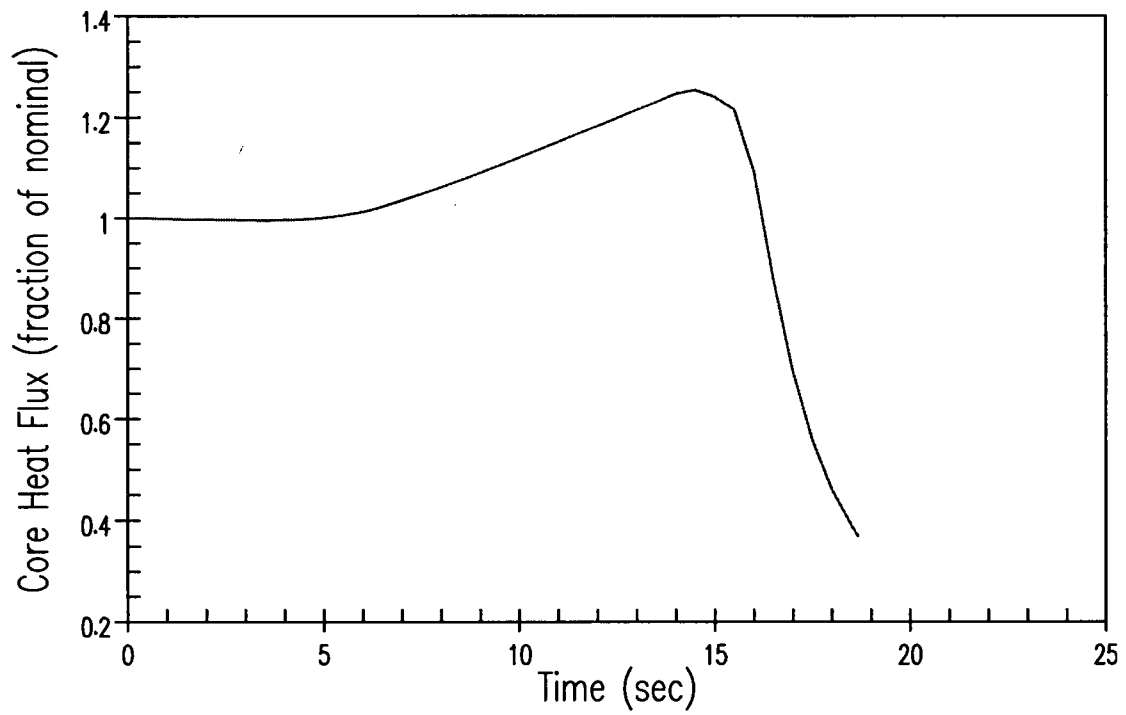
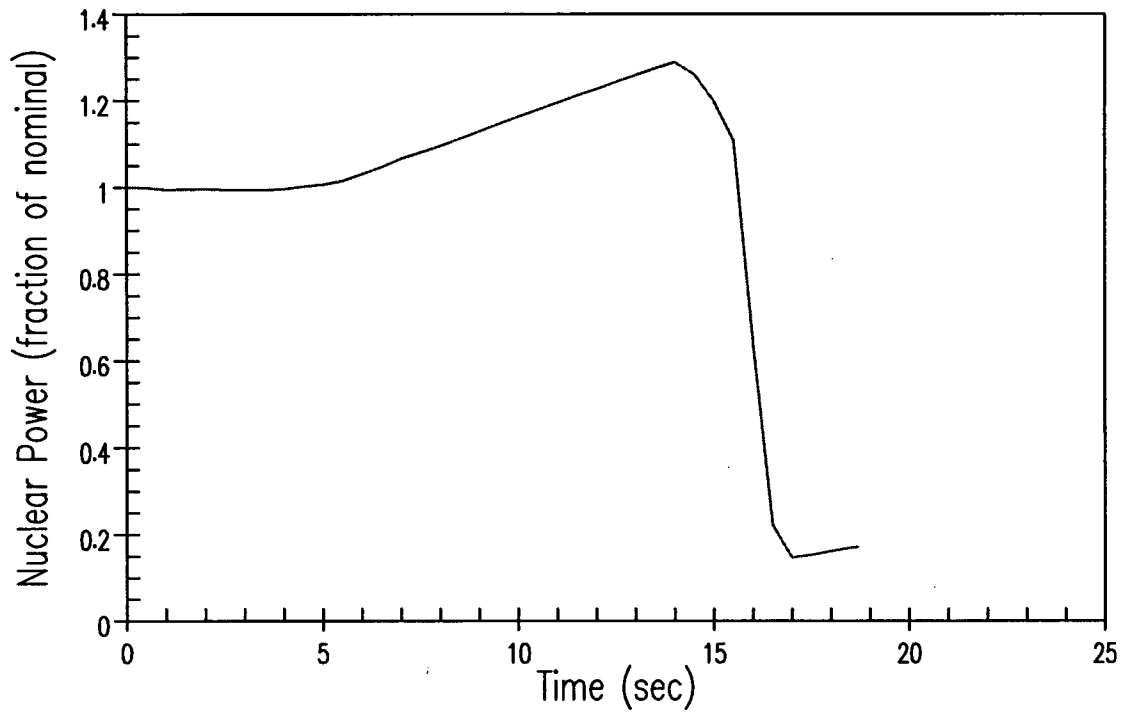


Figure 2.8.5.1.2.2.2-1 Steam System Piping Failure at Full-Power (Unit 1) – 1.388 ft² Break – Nuclear Power and Core Heat Flux Versus Time

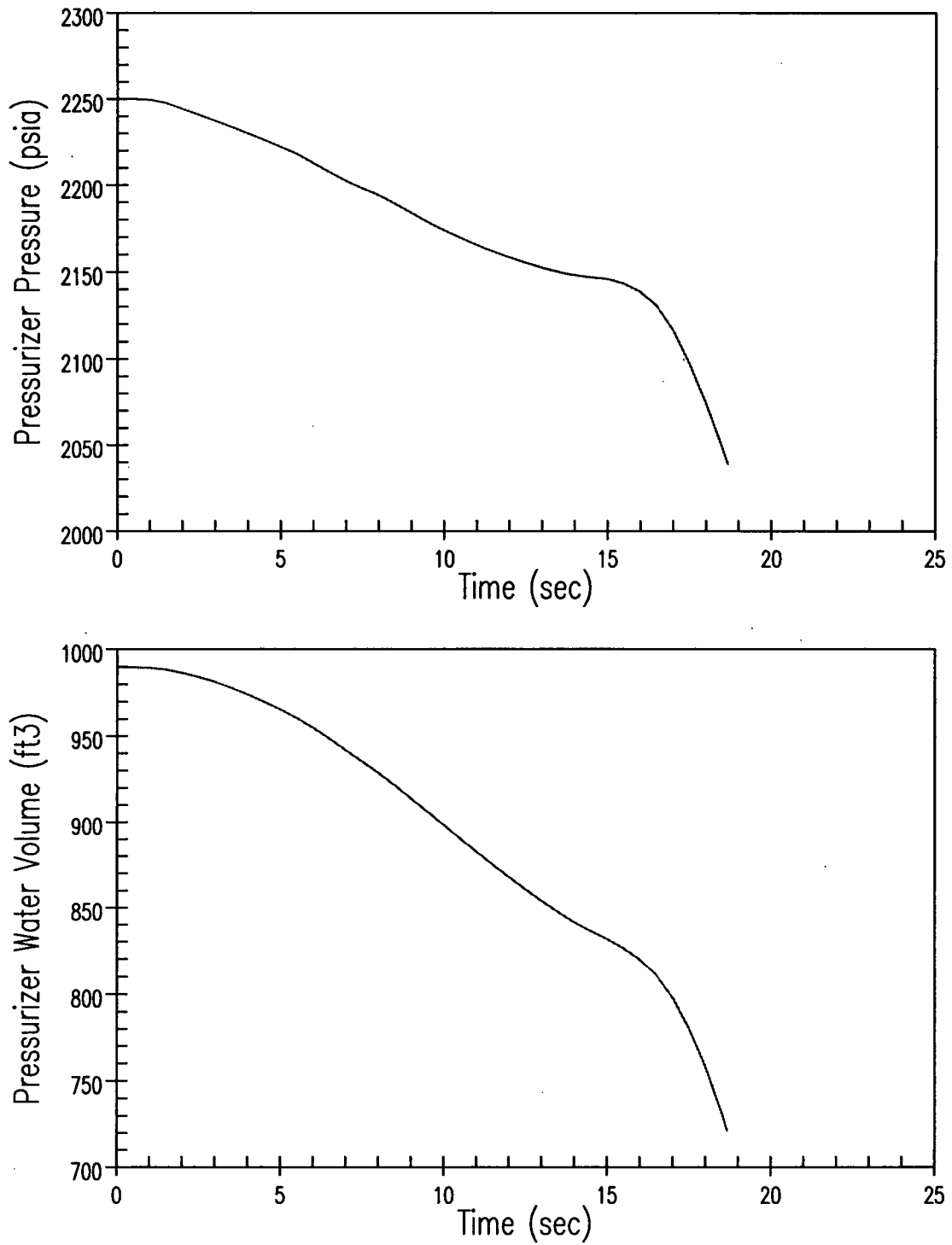


Figure 2.8.5.1.2.2.2-2 Steam System Piping Failure at Full-Power (Unit 1) – 1.388 ft² Break – Pressurizer Pressure and Pressurizer Water Volume Versus Time

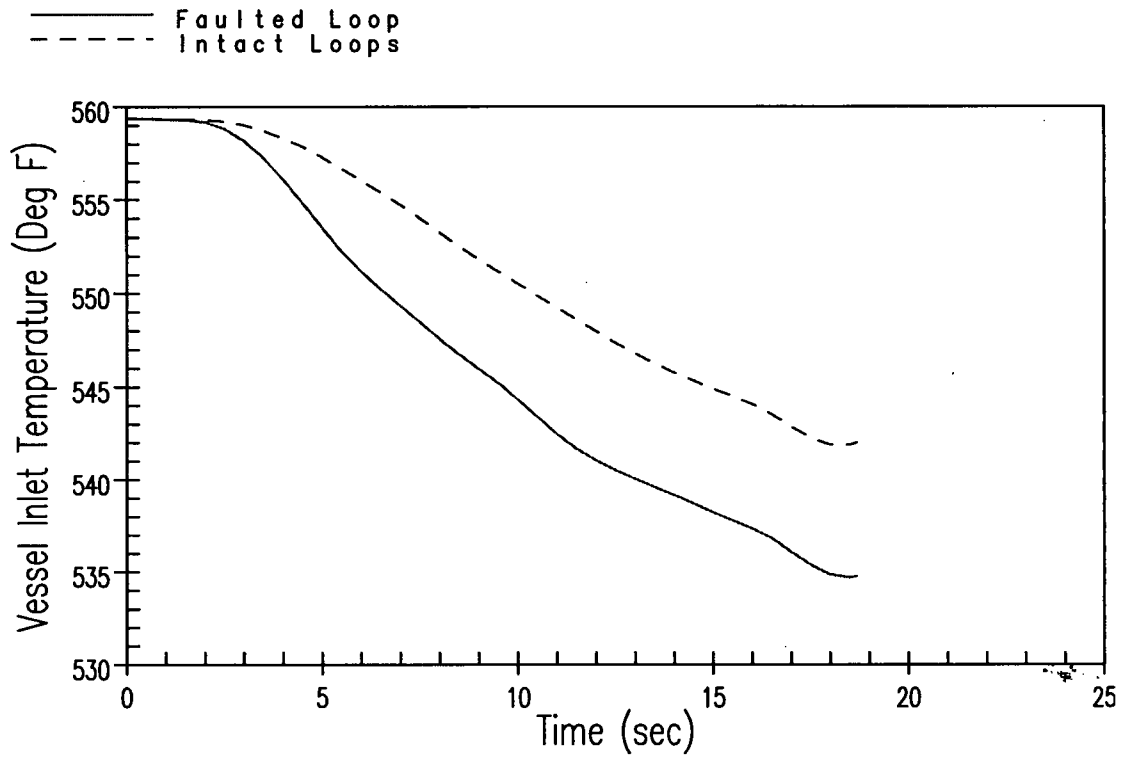


Figure 2.8.5.1.2.2.2-3 Steam System Piping Failure at Full-Power (Unit 1) – 1.388 ft² Break – Vessel Inlet Temperature Versus Time

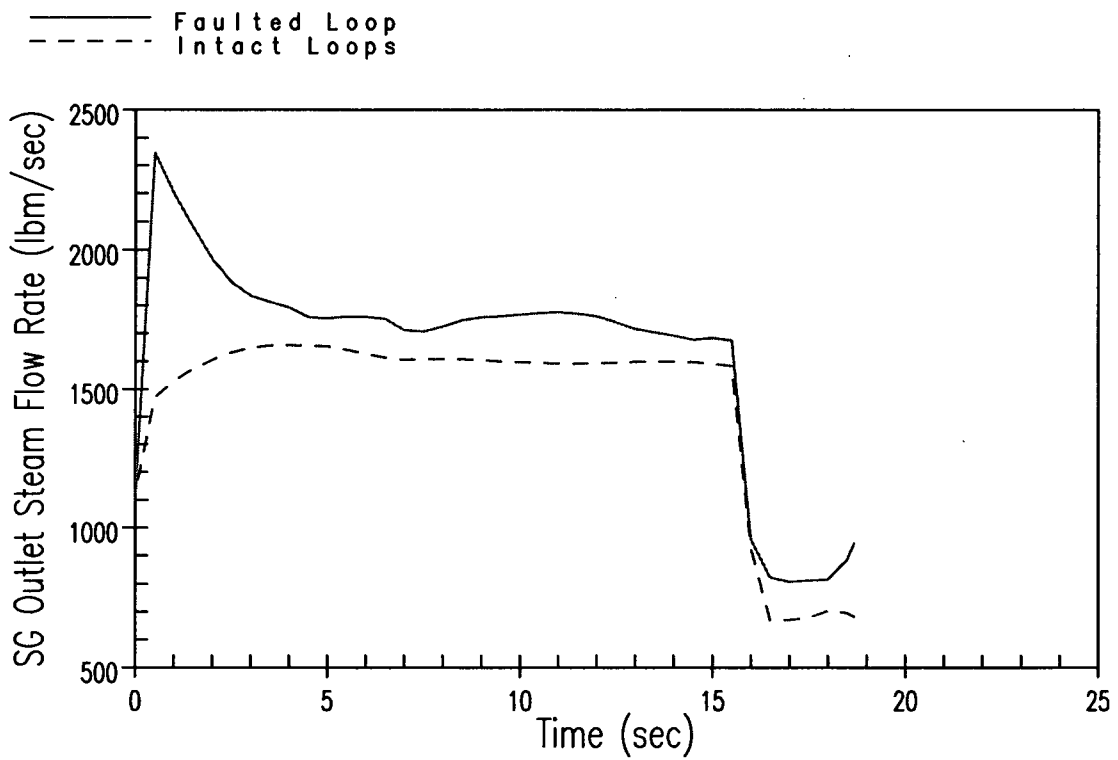
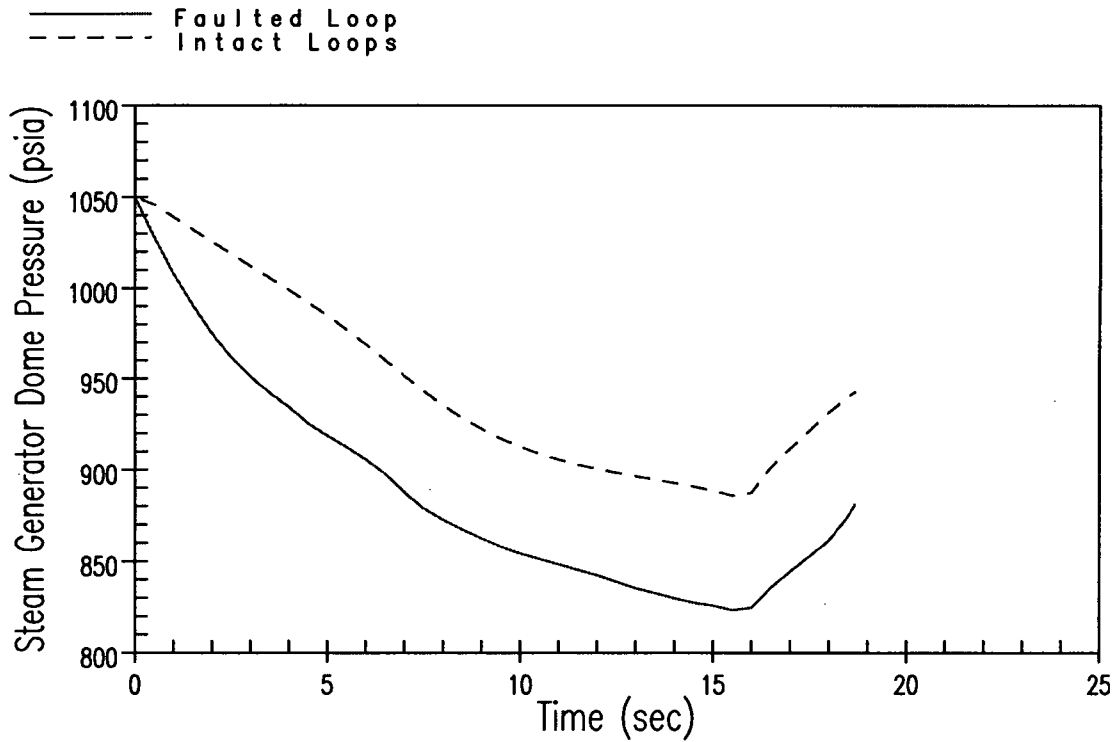


Figure 2.8.5.1.2.2.2-4 Steam System Piping Failure at Full-Power (Unit 1) – 1.388 ft² Break – Steam Generator Dome Pressure and Steam Generator Outlet Steam Flow Rate Versus Time

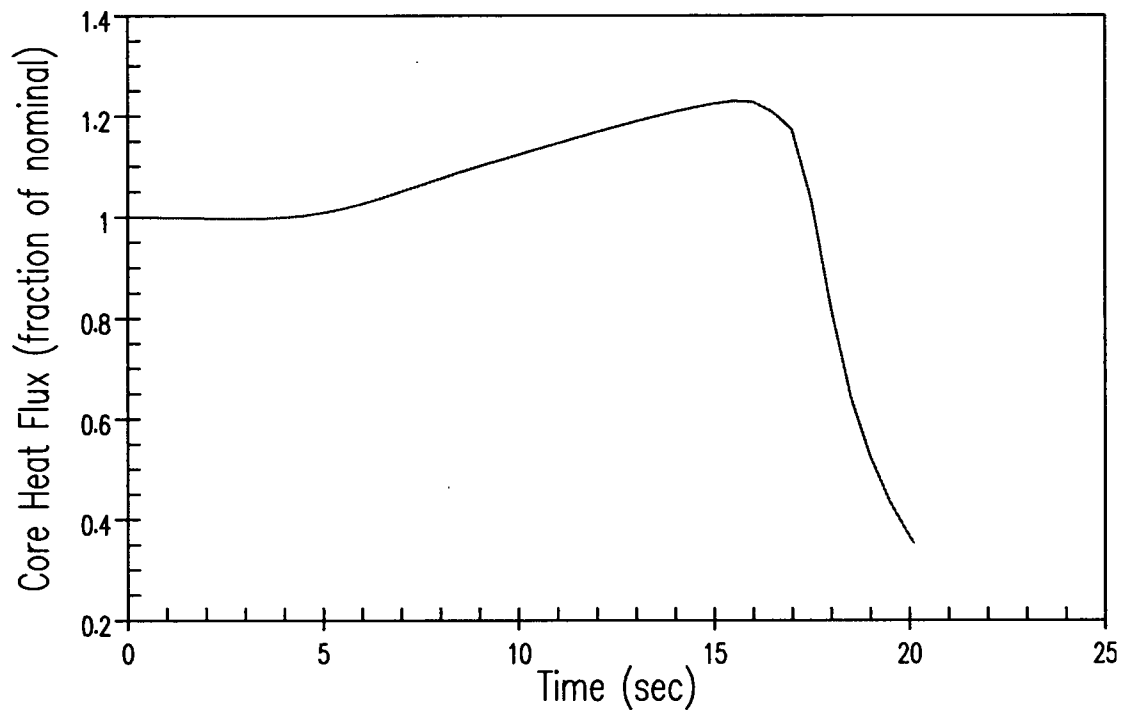
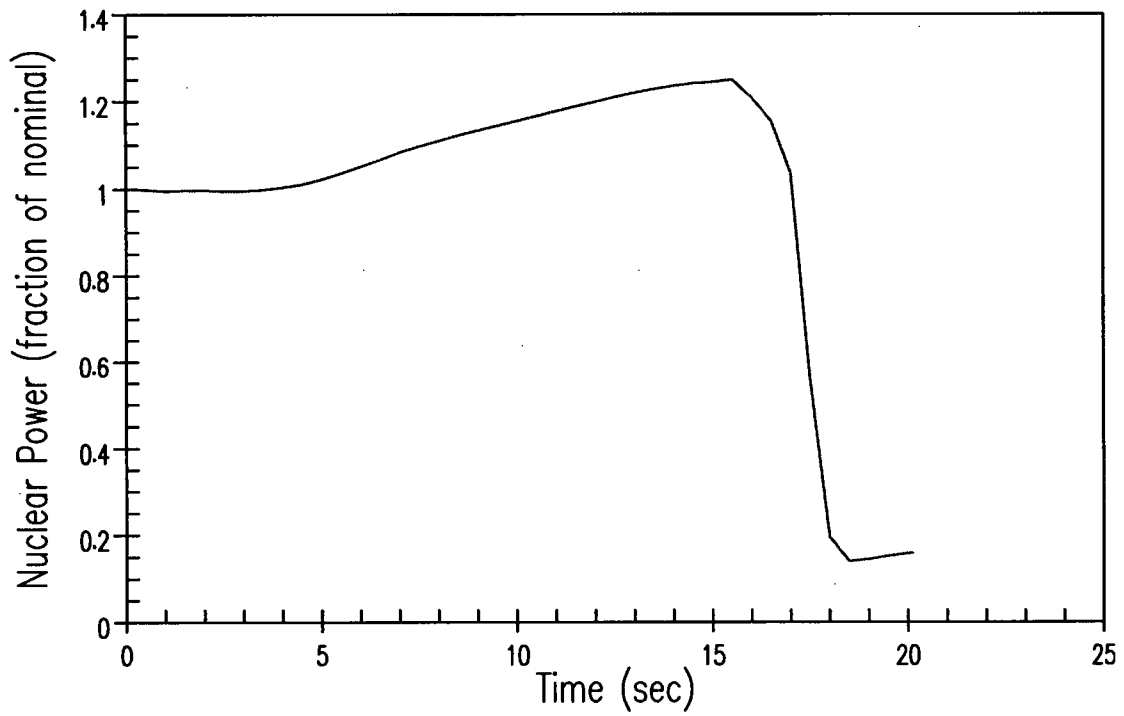


Figure 2.8.5.1.2.2.2-5 Steam System Piping Failure at Full-Power (Unit 2) – 1.388 ft² Break – Nuclear Power and Core Heat Flux Versus Time

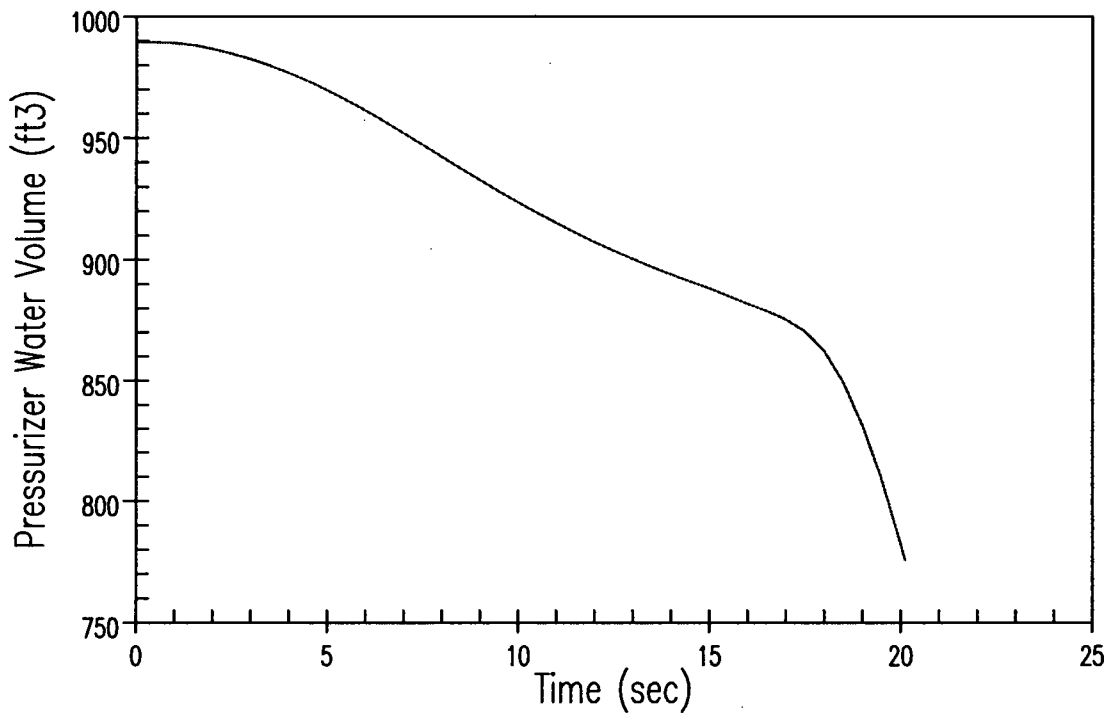
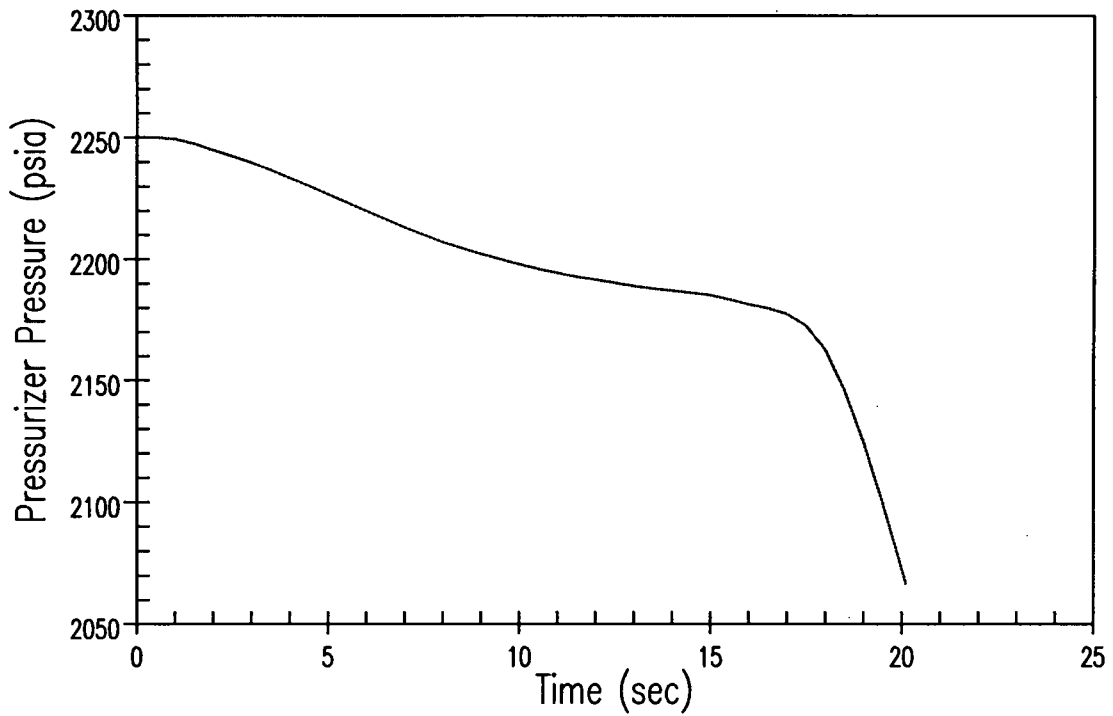


Figure 2.8.5.1.2.2.2-6 Steam System Piping Failure at Full-Power (Unit 2) – 1.388 ft² Break – Pressurizer Pressure and Pressurizer Water Volume Versus Time

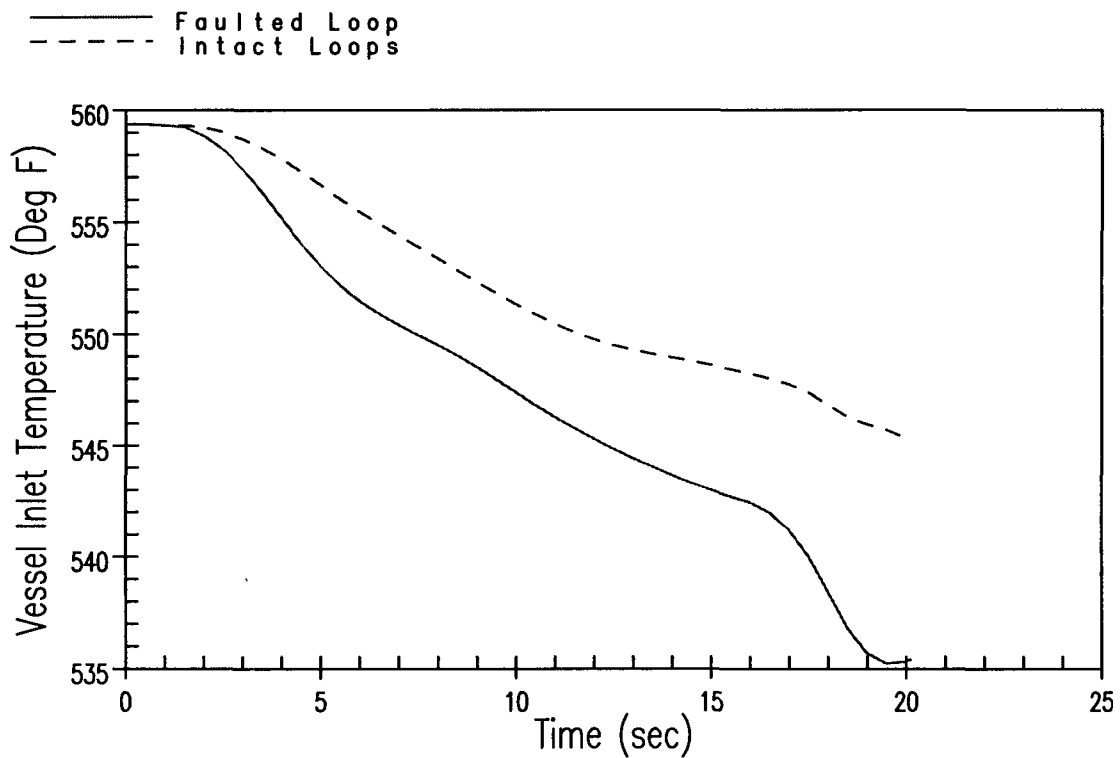


Figure 2.8.5.1.2.2.2-7 Steam System Piping Failure at Full-Power (Unit 2) – 1.388 ft² Break – Vessel Inlet Temperature Versus Time

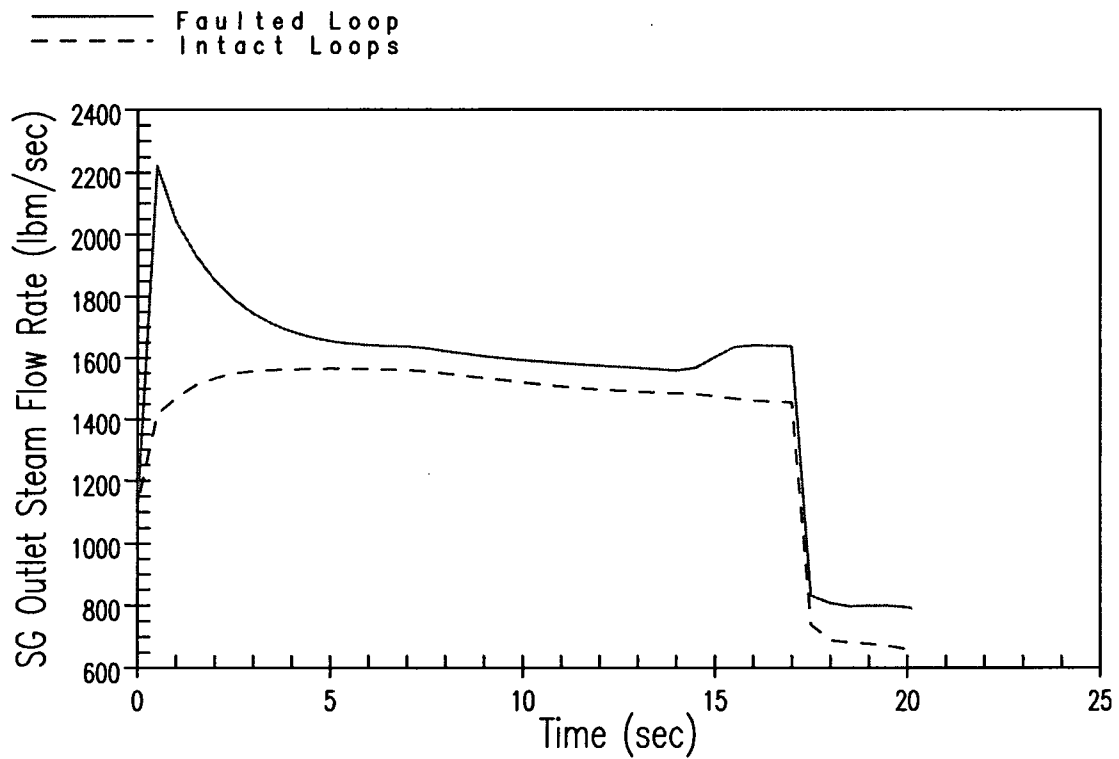
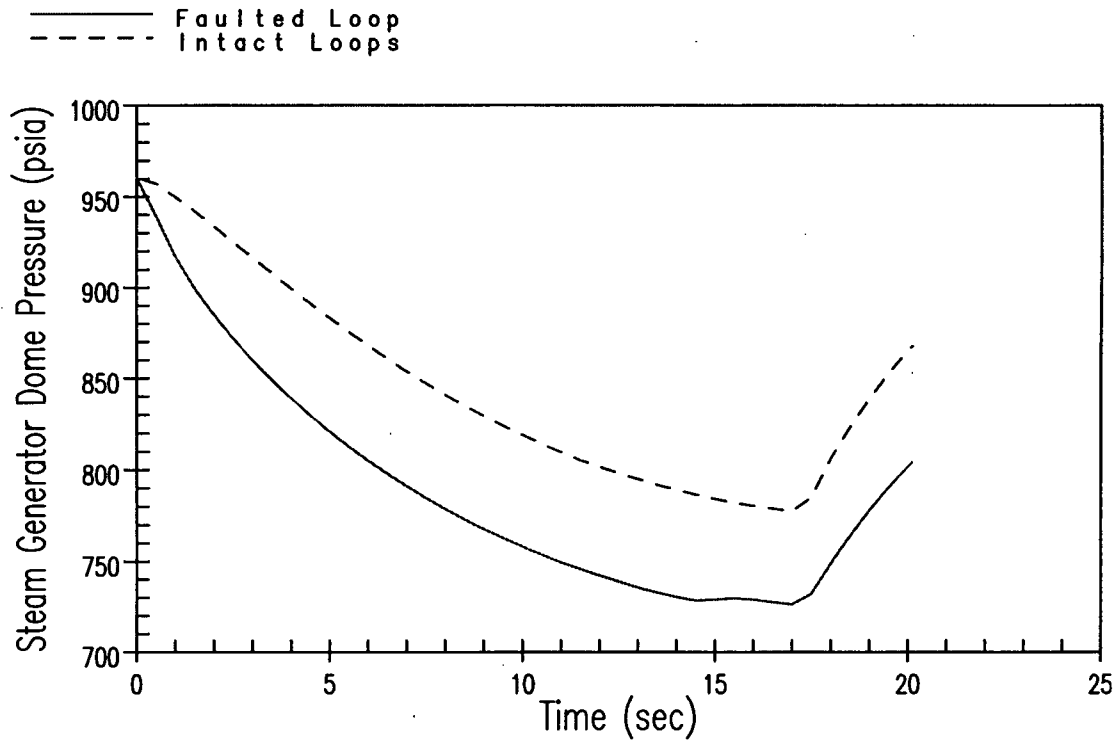


Figure 2.8.5.1.2.2.2-8 Steam System Piping Failure at Full-Power (Unit 2) – 1.388 ft² Break – Steam Generator Dome Pressure and Steam Generator Outlet Steam Flow Rate Versus Time

2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1 Loss of External Load, Turbine Trip, Steam Pressure Regulator Failure, and Loss of Condenser Vacuum

2.8.5.2.1.1 Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient.

The review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses.

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including anticipated operational occurrences (AOOs).
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation.
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II).
2. To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.
3. To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components. FSAR Chapter 7 provides the details of the control and protection systems instrumentation design and logic. This information supports the FSAR Chapter 15 accident analysis, which shows that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the reactor coolant and associated auxiliary control and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

Additionally, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, the use of proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components. FSAR Chapter 5 discusses the RCS design.

- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control systems are provided. They are the rod cluster control assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The rod control system automatically maintains a programmed average reactor temperature compensating for

reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and AOOs, thereby ensuring that specific fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses, and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The chemical and volume control system (CVCS) maintains the reactor in the cold shutdown state independent of the position of the control rods. It can compensate for xenon burnout transients.

FSAR Chapter 4 presents details of the construction of the RCCAs. FSAR Chapter 7 discusses their operation. FSAR Chapter 9 describes the means of controlling boric acid concentration. FSAR Chapter 15 includes performance analyses under accident conditions.

FSAR Sections 15.2.1, 15.2.2, 15.2.3, 15.2.4, and 15.2.5 provide the analyses of the loss of external electrical load, turbine trip, inadvertent closure of the main steam isolation valves, and loss of condenser vacuum, respectively. The loss of external load and turbine trip events are classified as Condition II events as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. As stated in FSAR Section 15.2.5, the inadvertent closure of the main steam isolation valves and the loss of condenser vacuum are two of the events that can cause a turbine trip.

Steam Pressure Regulator Failure

There are no steam pressure regulators at CPNPP whose failure or malfunction could cause a steam flow transient (FSAR Section 15.2.1).

Loss of External Electrical Load

FSAR Section 15.2.2.1 concludes that a loss of external load event results in a nuclear steam supply (NSSS) system transient that is less severe than a turbine trip event (FSAR Section 15.2.3). Therefore, a detailed transient analysis is not presented for the loss of external load.

Turbine Trip

For a turbine trip event, the reactor would be tripped directly (unless below approximately 50-percent power) from a signal derived from the turbine stop emergency trip fluid pressure and turbine stop valves. The turbine stop valves close rapidly (with a delay time typically less than 0.5 seconds) on loss of trip fluid pressure actuated by a turbine trip signal.

- This transient, to be referred to as the loss-of-load/turbine trip (LOL/TT) event, is analyzed with the RETRAN code to compute pertinent plant variables including temperatures, pressures, and power level.

-
- FSAR Section 15.2.3.3 concludes that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system (MSS). Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits. The integrity of the core is maintained by operation of the reactor protection system, that is, the departure from nucleate boiling ratio (DNBR) is maintained above the safety analysis limit. The DNBR design basis is described in FSAR Section 4.4. Applicable acceptance criteria as listed in FSAR Section 15.0.1 have been met. The analysis demonstrates the ability of the NSSS to safely withstand a full-load rejection.

Inadvertent Closure of the Main Steam Isolation Valves

As stated in FSAR Section 15.2.4, an inadvertent closure of the main steam isolation valves could cause a turbine trip. These are discussed in detail in FSAR Section 15.2.3. Therefore, the analysis results and conclusions contained in FSAR Section 15.2.3 (discussed above) apply to this incident as well.

Loss of Condenser Vacuum

As stated in FSAR Section 15.2.5, loss of condenser vacuum is one of the events that can cause a turbine trip. A loss of condenser vacuum would preclude the use of steam dumps to the condenser. However, since steam dumps are assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in FSAR Section 15.2.3 (discussed above) apply to loss of condenser vacuum.

2.8.5.2.1.2 Technical Evaluation

2.8.5.2.1.2.1 Introduction

A major load loss on the plant can result from either a loss of external electrical load or from a turbine trip. A loss of external electrical load can result from an abnormal variation in network frequency or other adverse network operating condition. In either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps (RCPs).

The plant is designed to accept a 50-percent loss of electrical load while operating at full power, or a complete loss of load while operating below the P-9 setpoint without actuating a reactor trip with all NSSS control systems in automatic. A 50-percent loss of electrical load is handled by the steam dump system (which accommodates 40 percent of the nominal full-power load), the rod control system (which accommodates the remaining 10 percent of the load rejection by driving rods in to reduce coolant average temperature), and the pressurizer (which absorbs the change in coolant volume due to the heat addition resulting from the load rejection). Should a complete loss of load occur from full power, the reactor trip system automatically actuates a reactor trip.

The most likely source of a complete loss of load on the NSSS is a trip of the turbine generator. In this case, there is a direct reactor trip signal derived from either the turbine auto-stop oil pressure or closure of the turbine stop valves, provided the reactor is operating above the P-9 setpoint. Reactor temperature and pressure do not increase significantly if the steam dump system and pressurizer pressure control system are functioning properly. However, the RCS and MSS pressure-relieving capacities are designed to ensure the safety of the plant without requiring the use of automatic rod control, pressurizer pressure control, and/or steam dump control systems. In this analysis, the behavior of the plant is evaluated for a complete loss of steam load from full power without direct reactor trip in order to demonstrate the adequacy of the pressure-relieving devices and core protection margins.

In the event the steam dump valves fail to open following a large loss of load, the main steam safety valves (MSSVs) can lift and the reactor can be tripped by the high pressurizer pressure signal, the overtemperature N-16 signal, or the overpower N-16 signal. The steam generator shell-side pressure and reactor coolant temperatures increase rapidly. The pressurizer safety valves (PSVs) and MSSVs are sized to protect the RCS and steam generator against overpressurization for all load losses without assuming the operation of the steam dump system, pressurizer sprays, pressurizer power-operated relief valves (PORVs), automatic rod control, or the direct reactor trip on turbine trip.

2.8.5.2.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Three cases were analyzed for a LOL/TT event from full-power conditions:

- With automatic pressure control (DNBR case)
- With automatic pressure control and minimum steam generator tube plugging (SGTP) (MSS pressure case)
- Without automatic pressure control and maximum SGTP (RCS pressure case)

The DNBR case was analyzed using the Revised Thermal Design Procedure (RTDP) (Reference 1). NSSS power, RCS temperature, and pressure were assumed to be at their nominal values consistent with steady-state, full-power operation. Minimum measured flow was modeled. Uncertainties in initial conditions were included in the safety analysis DNBR limit, as described in Reference 1.

The RCS and MSS peak pressure cases were analyzed using the standard thermal design procedure (STDP). Initial uncertainties on NSSS power, RCS temperature, and pressure were applied in the most conservative direction to obtain the initial plant conditions for the transient. Both cases modeled thermal design flow.

The LOL/TT transient was conservatively analyzed with minimum reactivity feedback (beginning of core life). All cases assumed the least-negative Doppler power coefficient and a 0 pcm/°F moderator temperature coefficient, which bounded part-power conditions, assuming a positive moderator temperature coefficient. Minimum reactivity conditions were conservative since

reactor power was maintained until the time of reactor trip, which exacerbated the calculated minimum DNBR and maximum RCS and MSS pressures.

Manual rod control was modeled for all cases. If the reactor had been in automatic rod control, the control rod banks would have driven into the core prior to reactor trip, thereby reducing the severity of the transient.

The LOL/TT event was analyzed both with and without pressurizer pressure control. The pressurizer PORVs and sprays were assumed operable for the DNBR case to minimize the increase in primary pressure, which was conservative for the DNBR criterion. The pressurizer PORVs and sprays were also assumed operable for the MSS peak pressure case to minimize the increase in primary pressure, which delayed or prevented reactor trip on a high pressurizer pressure signal, resulting in a conservatively high calculated peak secondary-side pressure. The RCS pressure case was analyzed without pressure control to conservatively maximize the RCS pressure increase. In all cases, the MSSVs and pressurizer safety valves were assumed to be operable.

A total PSV setpoint tolerance of $-3\%/+1\%$ was accounted for in the analysis. For the DNBR case, the negative tolerance was applied to conservatively reduce the setpoint. For the RCS peak pressure case and the MSS peak pressure case, the positive tolerance (in addition to a 0.9-percent setpoint shift) was applied to conservatively increase the setpoint pressure; in addition to this, the peak RCS pressure case includes a 1.05-second purge time delay associated with the existence of PSV water-filled loop seals, as described in Reference 3.

Main feedwater flow to the steam generators was assumed to be lost at the time of turbine trip. The auxiliary feedwater system is modeled. However, the low-low steam generator water level setpoint is not reached to initiate auxiliary feedwater flow.

The following reactor trip setpoints are assumed to be operable:

- Reactor trip on high pressurizer pressure
- Reactor trip on overtemperature N-16
- Reactor trip on overpower N-16
- Reactor trip on low-low steam generator water level

The MSSV model for all cases includes a 3-percent setpoint tolerance and an accumulation model that assumes that the safety valves are wide open once the pressure exceeds the setpoint (plus tolerance) by 5 psi.

The limiting single failure is failure of one train of the reactor trip system. The remaining (operable) train trips the reactor. As described in FSAR Section 3.1.1, the MSSVs and pressurizer safety valves (that is, code safety valves) are considered especially qualified active components and are assumed to open on demand. Control systems are not assumed to operate abnormally during a transient except as an initial condition (such as a rod withdrawal event). Thus, a failure of a control system is not applicable as a limiting single failure. Feedwater isolation (redundant valves with different closure times), auxiliary feedwater (multiple

pumps) and safety injection (multiple pumps) are susceptible to a single failure. However, none of these systems provide any mitigation for a LOL/TT event. Thus, these systems are not applicable as a limiting single failure. Furthermore, the protection system is designed to be single failure proof.

Maximum (10-percent) steam generator tube plugging is assumed in the DNBR case and RCS peak pressure case since it maximizes the RCS temperature transient following event initiation. However, the MSS peak pressure case is analyzed at zero steam generator tube plugging since this conservatively maximizes the primary-to-secondary side heat transfer; this assumption is slightly more limiting with respect to the secondary-side pressure transient.

Based on its frequency of occurrence, the LOL/TT accident is considered a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. The specific criteria for this accident, as stated in the Standard Review Plan (SRP), are as follows:

- Pressure in the RCS and MSS are maintained below 110 percent of the design values (for the CPNPP units, this represents an RCS pressure limit of 2,748.2 psia and secondary-side pressure limit of 1,318.2 psia).
- Fuel cladding integrity is maintained by demonstrating that the minimum DNBR remains above the 95/95 DNBR limit for pressurized water reactors (PWRs) (for the CPNPP units, the applicable safety analysis DNBR limit is 1.61).
- An incident of moderate frequency does not generate a more serious plant condition without other faults occurring independently.

This criterion is conservatively satisfied by verifying that the pressurizer does not fill.

- An incident of moderate frequency in combination with any single active component failure, or single operator error, is considered an event for which an estimate of the number of potential fuel failures is provided for radiological dose calculations. For such accidents, fuel failure is assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown that, based on an acceptable fuel damage model, fewer failures occur. There is no loss of function of any fission product barrier other than the fuel cladding.

This criterion is satisfied by verifying that the DNBR remains above the 95/95 DNBR limit, discussed above.

2.8.5.2.1.2.3 Description of Analyses and Evaluations

A detailed analysis using the RETRAN (Reference 2) computer code was performed to determine the plant transient conditions following a total loss of load due to turbine trip without a direct reactor trip. The code models the core neutron kinetics, RCS, pressurizer, pressurizer PORVs and sprays, steam generators, MSSVs, and the auxiliary feedwater system. RETRAN

computes pertinent variables, including the pressurizer pressure, steam generator pressure, and reactor coolant average temperature.

RETRAN has been approved by the NRC for the analysis of LOL/TT transient (Reference 2).

2.8.5.2.1.2.4 Results

The calculated sequence of events for all cases for both units is listed in Table 2.8.5.2.1.2.4-1, while the limiting values for each case are presented in Table 2.8.5.2.1.2.4-2.

2.8.5.2.1.2.4.1 DNBR

For both Units 1 and 2, the DNBR case was analyzed at the high nominal T_{avg} value (i.e., 589.2°F), nominal pressure (i.e., 2,250 psia), minimum measured flow, pressurizer pressure control available, 10 percent SGTP, and high main feedwater temperature (450.3°F) conditions.

The transient response plot results for the total loss-of-load/turbine trip event (DNBR case) are shown in Figures 2.8.5.2.1-1 through 2.8.5.2.1-6. The following results discussion is applicable to both units. The reactor was tripped on the OTN-16 reactor trip function. The nuclear power increased slightly until the reactor was tripped. The pressurizer PORVs, safety valves, and sprays actuated to minimize the primary pressure transient, which was conservative for DNBR. Although the DNBR decreased below the initial value, it remained well above the safety analysis limit throughout the entire transient. The peak pressurizer water volume remained below the total volume of the pressurizer, demonstrating that this event did not generate a more serious plant condition. The MSSVs actuated to maintain the secondary side pressure below 110 percent of the design value.

2.8.5.2.1.2.4.2 MSS Pressure Case

For both Units 1 and 2, the MSS Pressure case was analyzed at the high nominal T_{avg} value plus uncertainties (that is, 589.2°F + 6°F), nominal pressure minus uncertainties (that is, 2,250 psia – 30 psi), thermal design flow, pressurizer pressure control available, 0-percent SGTP, and high main feedwater temperature (450.3°F) conditions.

The transient response plot results for the LOL/TT event (MSS pressure case) are shown in Figures 2.8.5.2.1-7 through 2.8.5.2.1-12. The reactor was tripped on the OTN-16 reactor trip function. The nuclear power remained essentially constant at full power until the reactor was tripped and the pressurizer PORVs and sprays (Units 1 and 2), and safety relief valves (Unit 2 only) minimized the primary pressure transient, which was conservative to prevent a reactor trip on high pressurizer pressure and exacerbate the peak secondary-side pressure. The MSSVs actuated to maintain the secondary-side pressure below 110 percent of the design value. The peak pressurizer water volume remained below the total volume of the pressurizer, demonstrating that this event did not generate a more serious plant condition.

2.8.5.2.1.2.4.3 RCS Pressure Case

For Unit 1 with the Model Δ 76 steam generators, the most limiting LOL/TT RCS overpressurization case was that with the temperature uncertainty subtracted from the high nominal T_{avg} value (that is, 589.2° – 6°F), pressure uncertainty subtracted from the nominal value (that is, 2,250 – 30 psi), thermal design flow, pressurizer pressure control (PORVs and sprays) not available, 10-percent SGTP, and high main feedwater temperature (450.3°F) conditions.

For Unit 2 with the Model D-5 steam generators, the most limiting LOL/TT RCS overpressurization case was at the high nominal T_{avg} value (that is, 589.2°F), pressure uncertainty subtracted from the nominal value (that is, 2,250 – 30 psi), thermal design flow, pressurizer pressure control (PORVs, sprays) not available, 10-percent SGTP, and high main feedwater temperature (450.3°F) conditions.

The transient response plot results for the LOL/TT event (RCS pressure case) are shown in Figures 2.8.5.2.1-13 through 2.8.5.2.1-18. The following results discussion is applicable to both units. The reactor was tripped on the high pressurizer pressure reactor trip function. The nuclear power remained essentially constant at full power until the reactor was tripped. The PSVs actuated and maintained the primary-side pressure below 110 percent of the design value. The MSSVs also actuated and maintained the secondary side pressure below 110 percent of the design value. The peak pressurizer water volume remained below the total volume of the pressurizer, demonstrating that this event did not generate a more serious plant condition.

2.8.5.2.1.3 Conclusions

From a review of the updated analyses of the LOL/TT event, it is concluded that these have adequately accounted for operation of the plant at the proposed uprated power level and that they were performed using acceptable analytical models. The results obtained demonstrate that the reactor protection and safety systems will continue to ensure that the SAFDLs are met and the RCS and MSS pressure boundary limits will not be exceeded as a result of the LOL/TT event. Furthermore, this event will not generate a more serious plant condition. Based on this, both units will continue to meet the requirements of GDCs -10, -15, and -26 following implementation of the SPU. Therefore, the SPU is acceptable with respect to the LOL/TT event.

2.8.5.2.1.4 References

1. WCAP-11397, "Revised Thermal Design Procedure," April 1989.
2. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
3. WCAP-12910, Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift," May 1993.

Table 2.8.5.2.1.2.4-1			
Time Sequence of Events – Loss of External Electrical Load and/or Turbine Trip			
Case	Event	Time (sec)	
		Unit 1	Unit 2
DNBR Case	Loss of Electrical Load/Turbine Trip Occurs	0.0	0.0
	OTN-16 Reactor Trip Setpoint Reached	11.7	10.9
	Minimum DNBR Occurs	11.7	14.0
	Rods Begin to Drop	13.7	12.9
MSS Pressure Case	Loss of Electrical Load/Turbine Trip Occurs	0.0	0.0
	OTN-16 Reactor Trip Setpoint Reached	16.8	15.9
	Rods Begin to Drop	18.8	17.9
	Peak Secondary Side Pressure Occurs	21.7	21.7
RCS Pressure Case	Loss of Electrical Load/Turbine Trip Occurs	0.0	0.0
	High Pressurizer Pressure Reactor Trip Setpoint Reached	5.62	5.67
	Rods Begin to Drop	6.87	6.92
	Peak RCS Pressure Occurs	8.0	8.3

Table 2.8.5.2.1.2.4-2			
Limiting Results – Loss of External Electrical Load and/or Turbine Trip			
Case	Parameter	Value	
		Unit 1	Unit 2
DNBR Case	Minimum DNBR	2.08	1.98
MSS Pressure Case	Peak MSS Pressure (psia)	1,298.4	1,297.2
RCS Pressure Case	Peak RCS Pressure (psia)	2,734.7	2,746.0

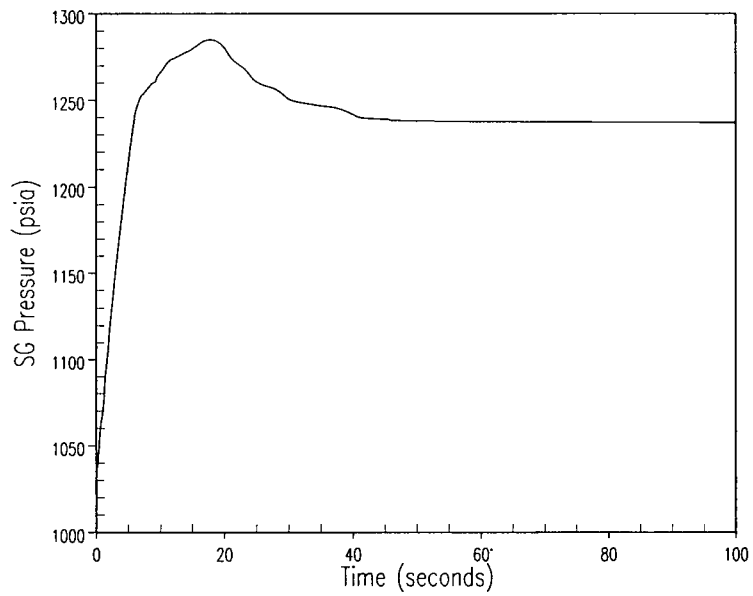
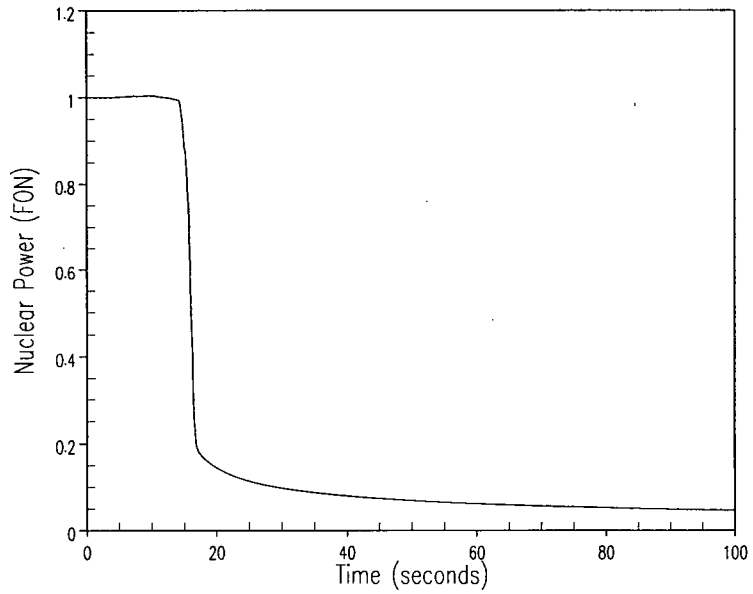


Figure 2.8.5.2.1-1 Unit 1 Loss of Load/Turbine Trip DNBR Case – Nuclear Power/Heat Flux and Steam Generator Pressure Versus Time

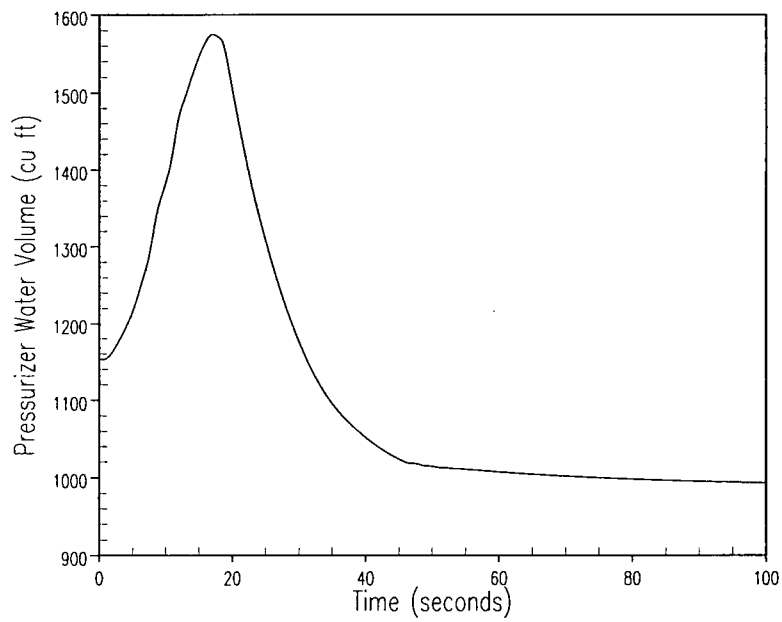
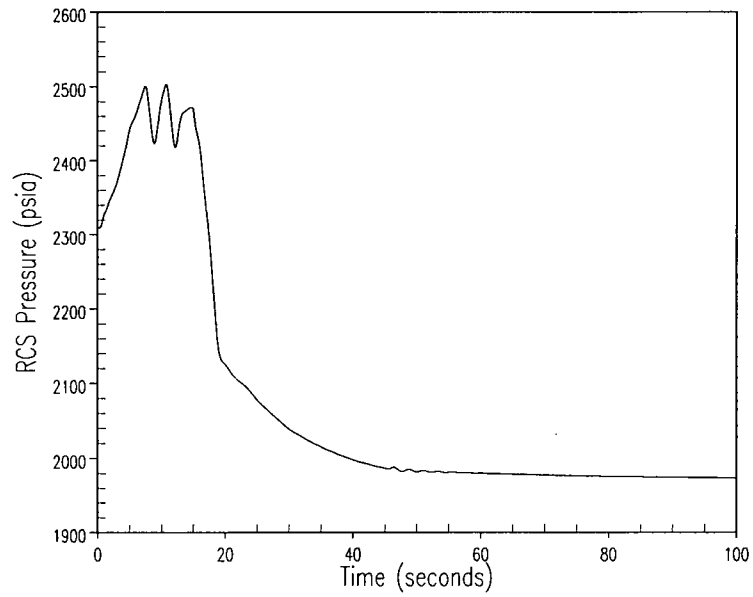


Figure 2.8.5.2.1-2 Unit 1 Loss of Load/Turbine Trip DNBR Case – RCS Pressure and Pressurizer Water Volume Versus Time

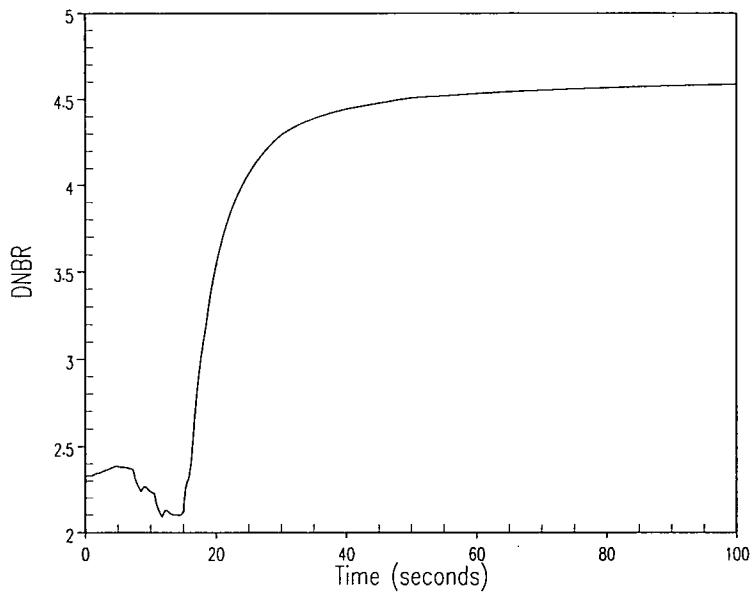
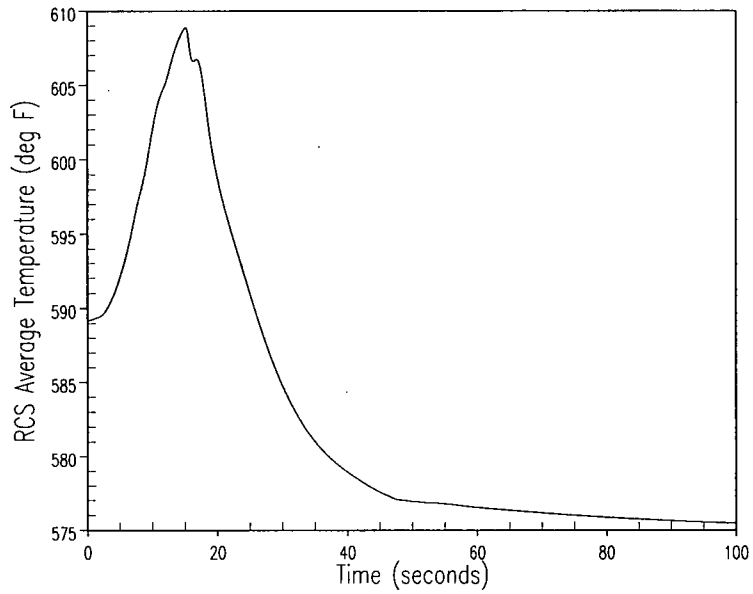


Figure 2.8.5.2.1-3 Unit 1 Loss of Load/Turbine Trip DNBR Case – RCS Average Temperature and DNBR Versus Time

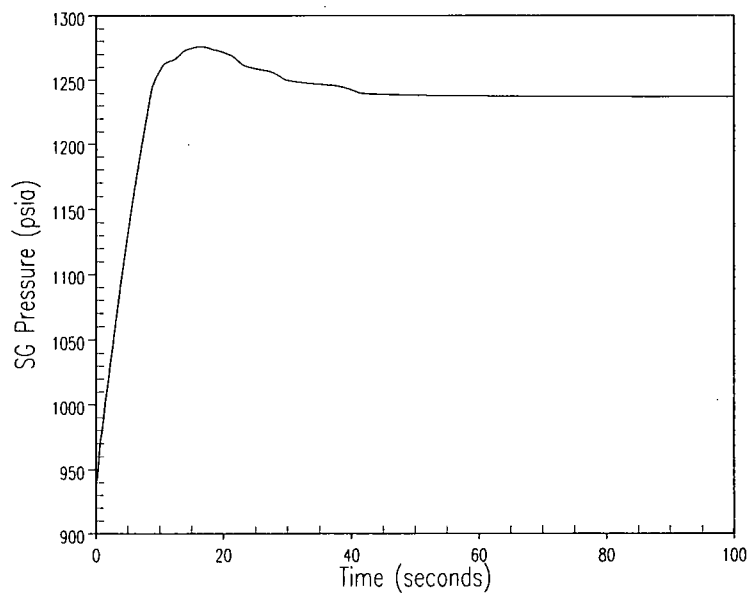
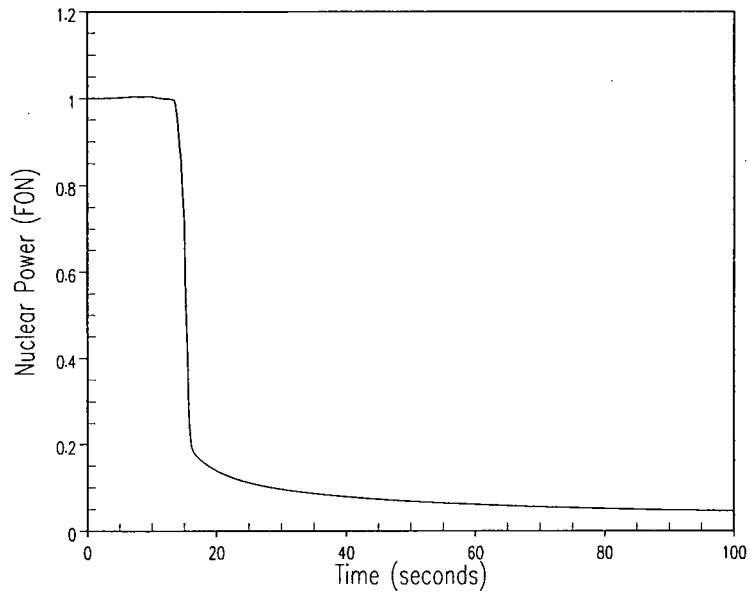


Figure 2.8.5.2.1-4 Unit 2 Loss of Load/Turbine Trip DNBR Case – Nuclear Power and Steam Generator Pressure Versus Time

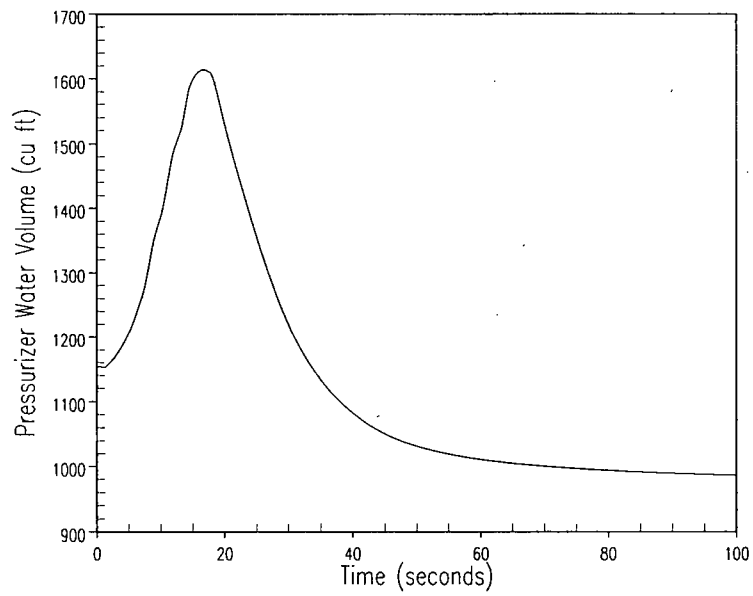
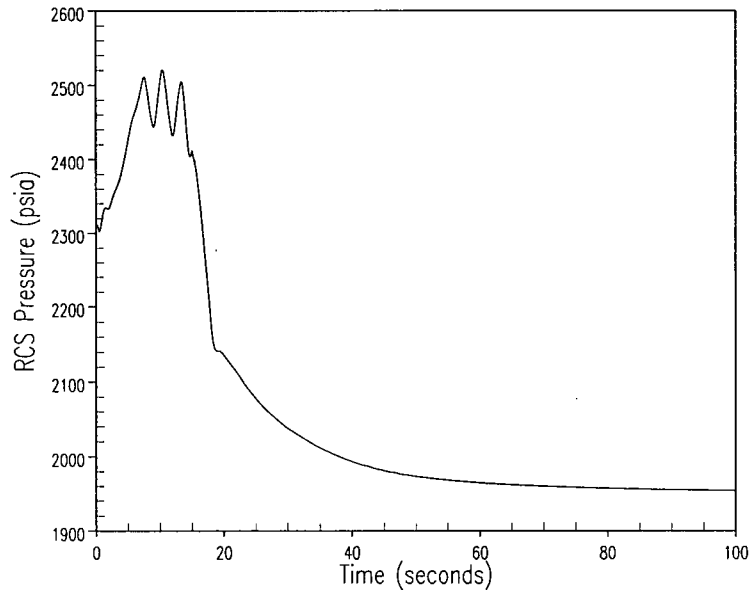


Figure 2.8.5.2.1-5 Unit 2 Loss of Load/Turbine Trip DNBR Case – RCS Pressure and Pressurizer Water Volume Versus Time

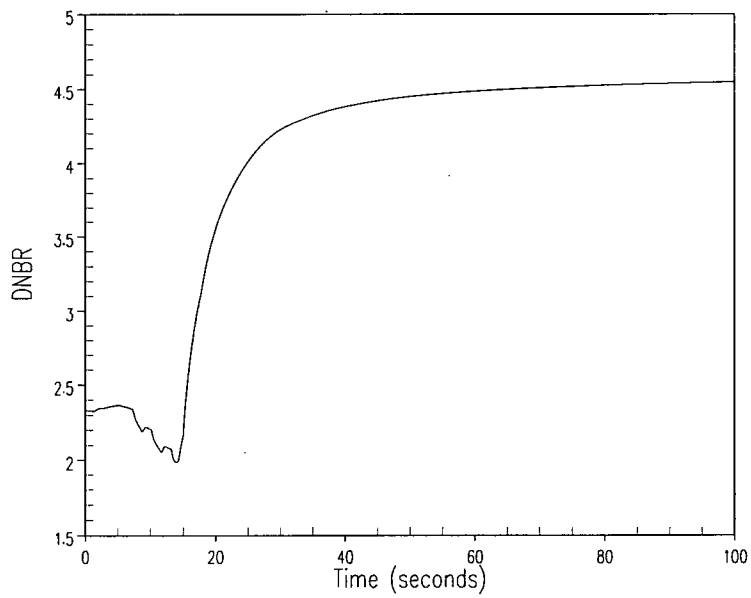
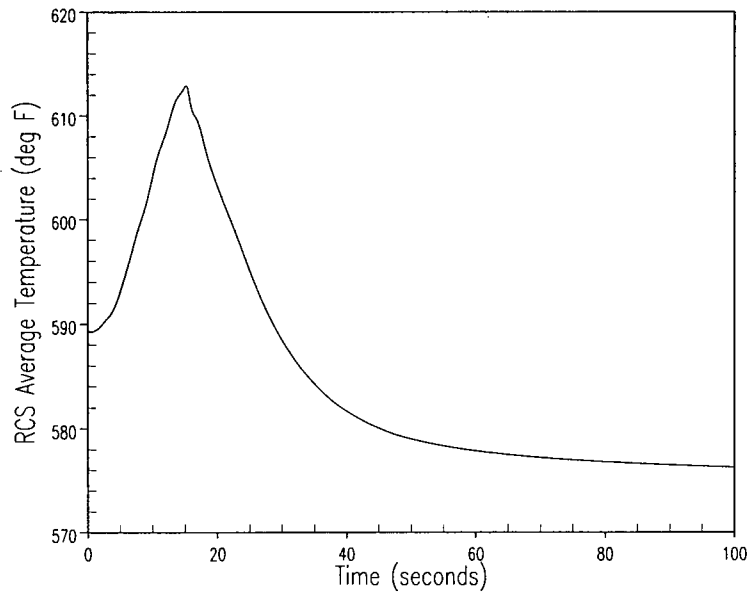


Figure 2.8.5.2.1-6 Unit 2 Loss of Load/Turbine Trip DNBR Case – RCS Average Temperature and DNBR Versus Time

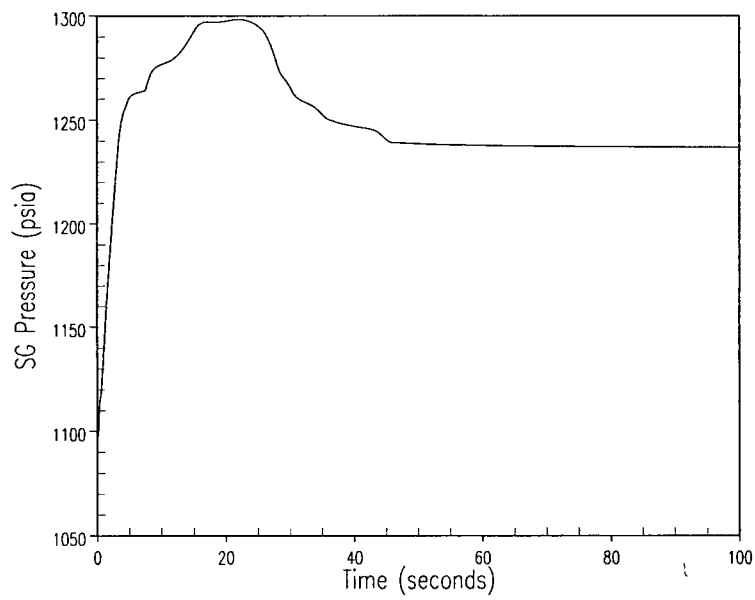
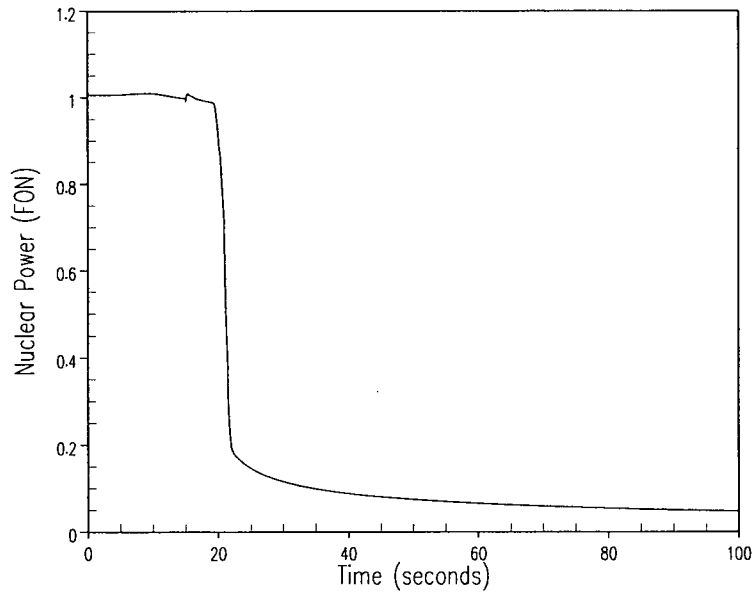


Figure 2.8.5.2.1-7 Unit 1 Loss of Load/Turbine Trip MSS Pressure Case – Nuclear Power and Steam Generator Pressure Versus Time

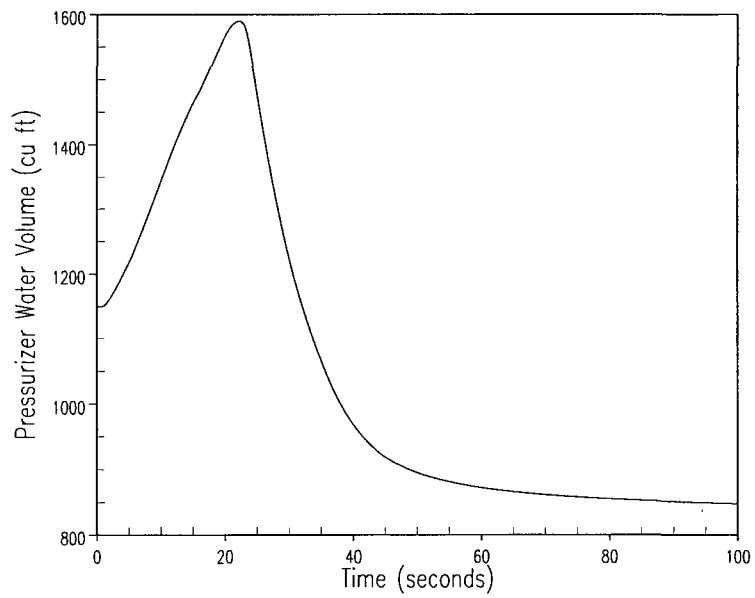
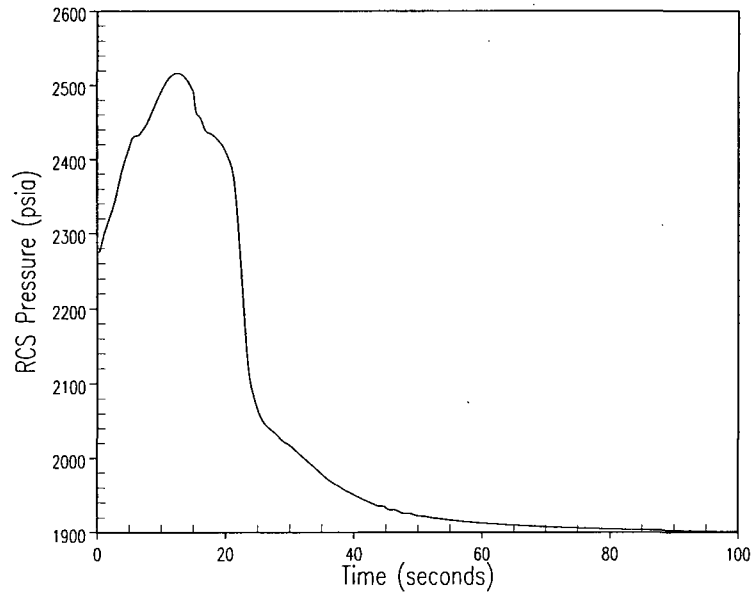


Figure 2.8.5.2.1-8 Unit 1 Loss of Load/Turbine Trip MSS Pressure Case – RCS Pressure and Pressurizer Water Volume Versus Time

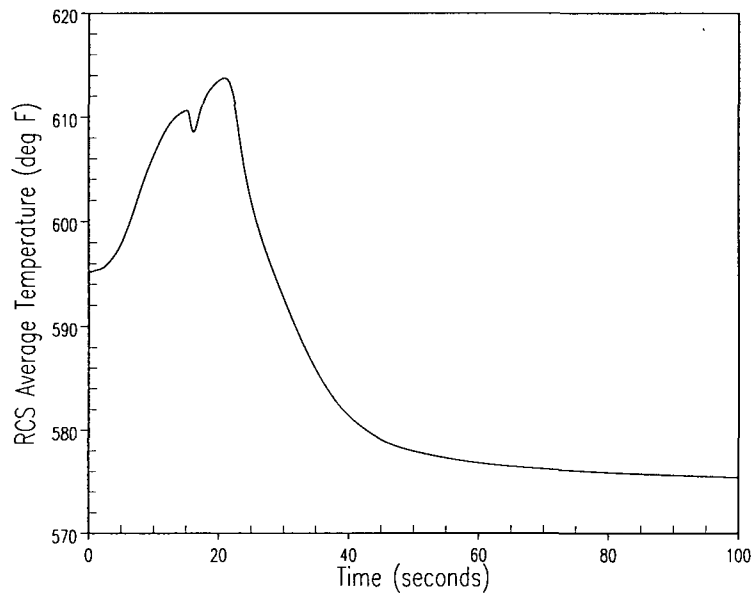


Figure 2.8.5.2.1-9 Unit 1 Loss of Load/Turbine Trip MSS Pressure Case – RCS Average Temperature Versus Time

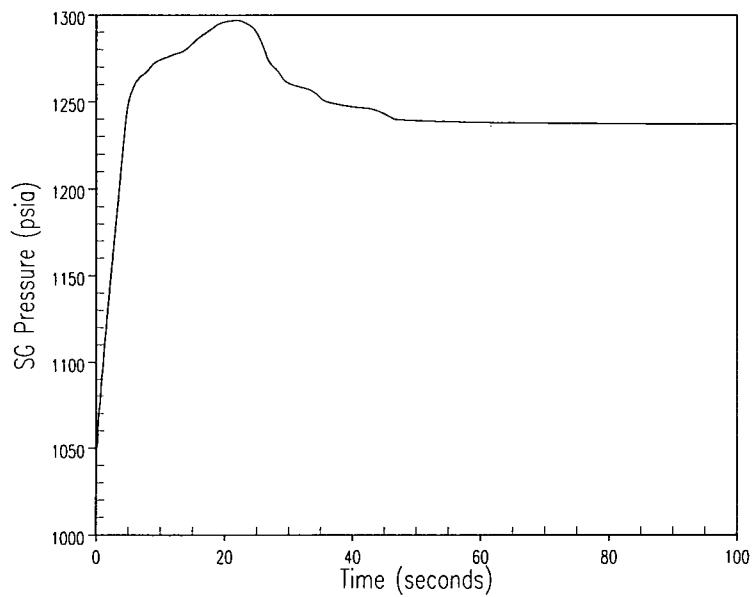
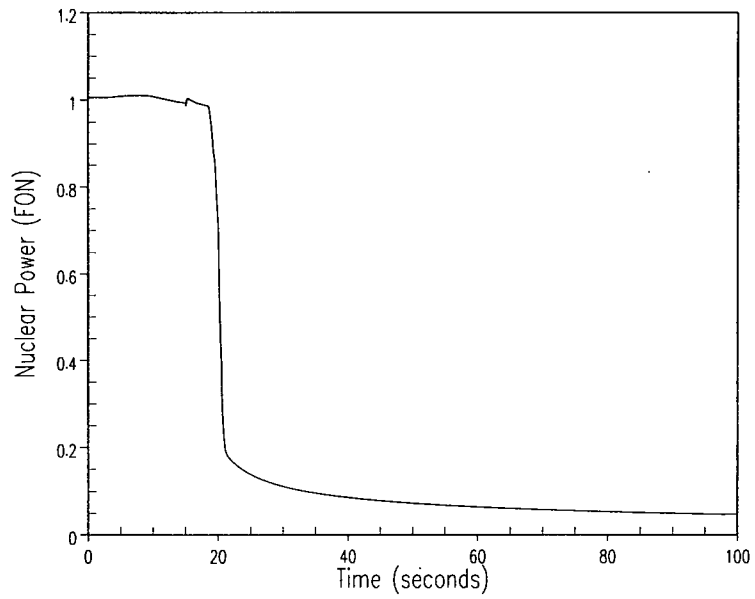


Figure 2.8.5.2.1-10 Unit 2 Loss of Load/Turbine Trip MSS Pressure Case – Nuclear Power and Steam Generator Pressure Versus Time

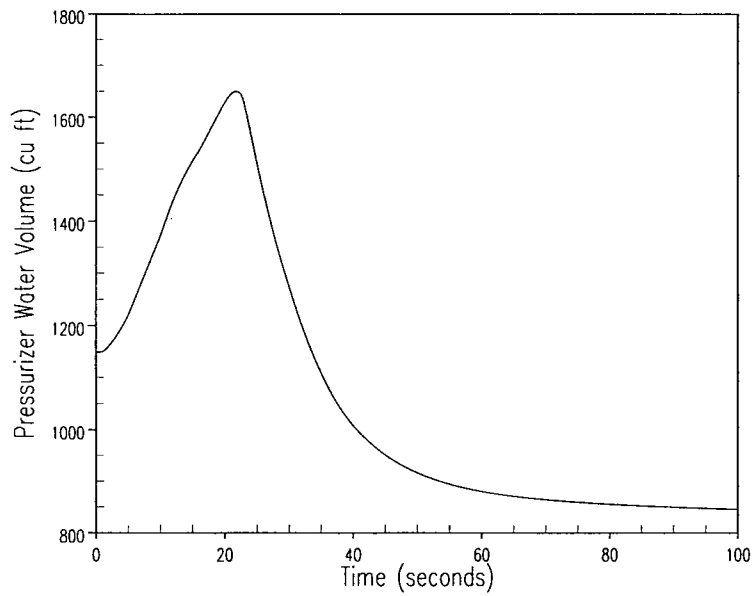
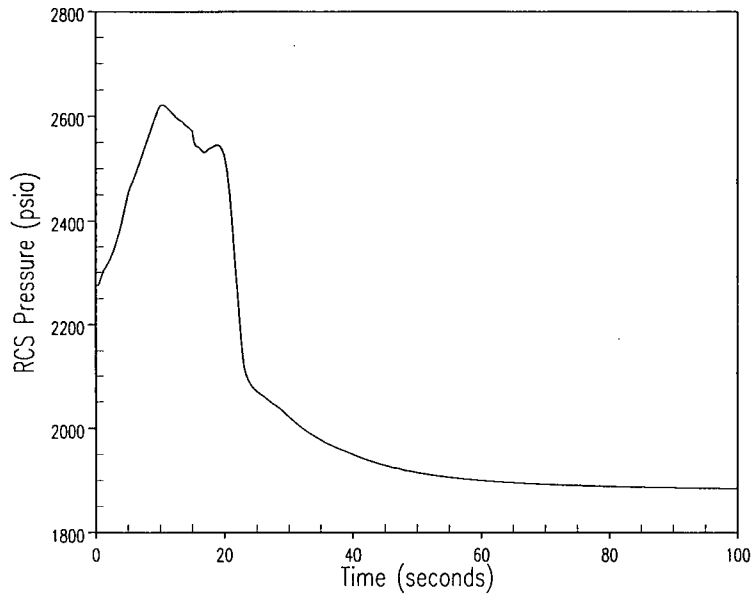


Figure 2.8.5.2.1-11 Unit 2 Loss of Load/Turbine Trip MSS Pressure Case – RCS Pressure and Pressurizer Water Volume Versus Time

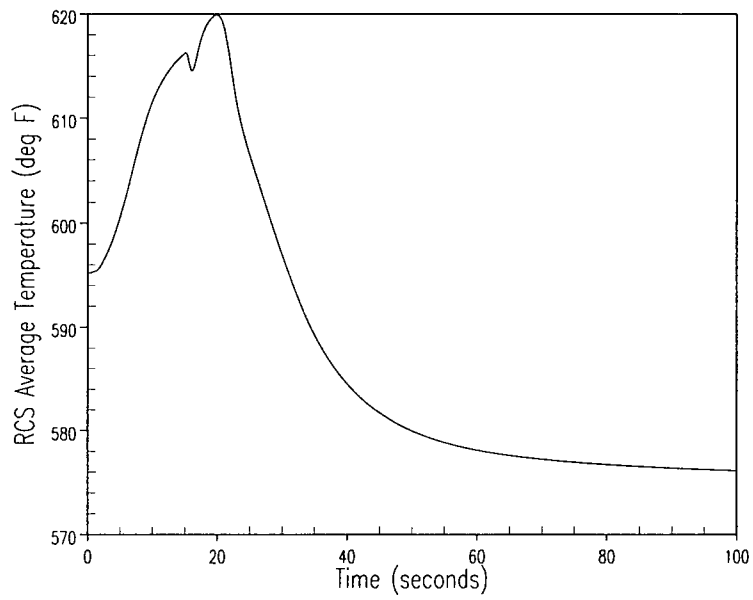


Figure 2.8.5.2.1-12 Unit 2 Loss of Load/Turbine Trip MSS Pressure Case – RCS Average Temperature Versus Time

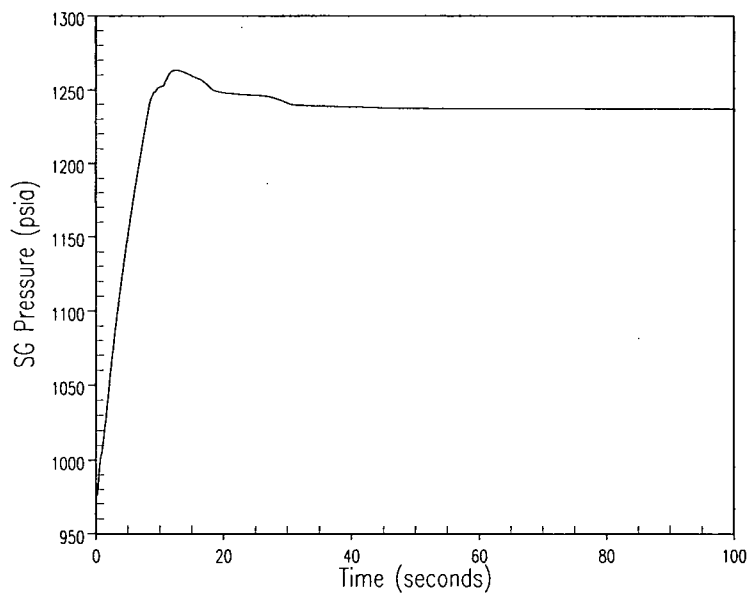
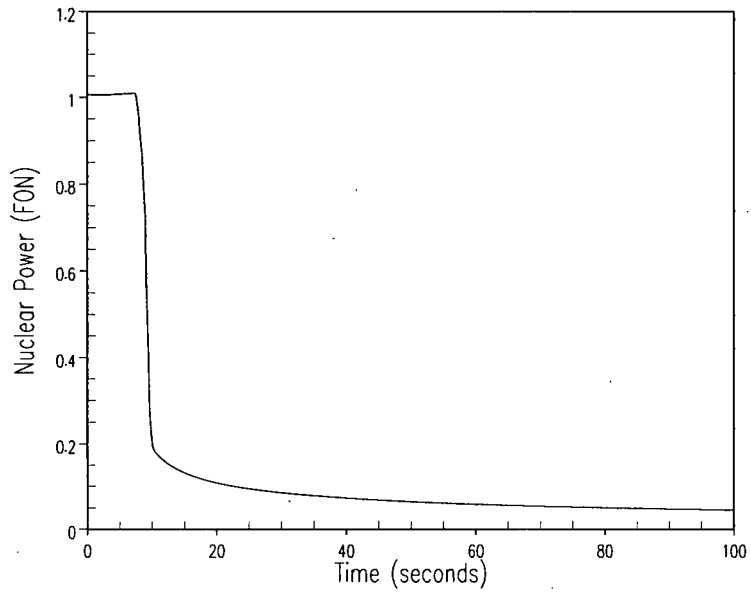


Figure 2.8.5.2.1-13 Unit 1 Loss of Load/Turbine Trip RCS Pressure Case – Nuclear Power and Steam Generator Pressure Versus Time

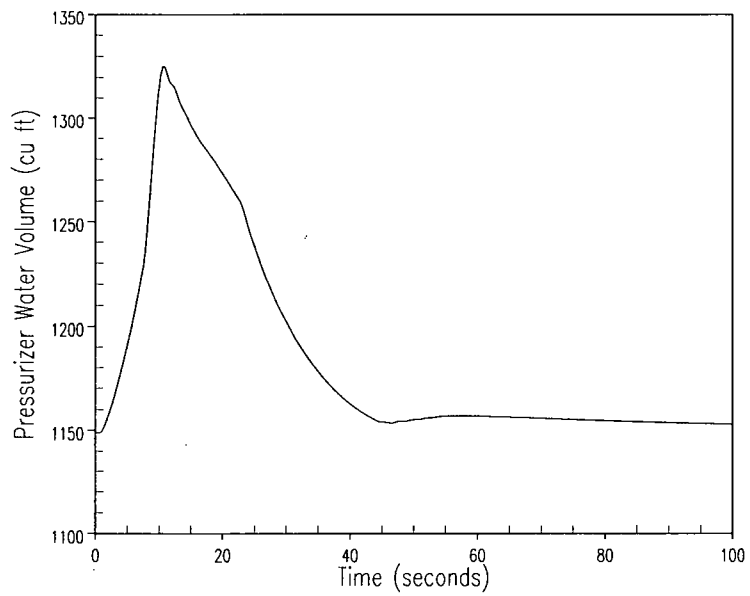
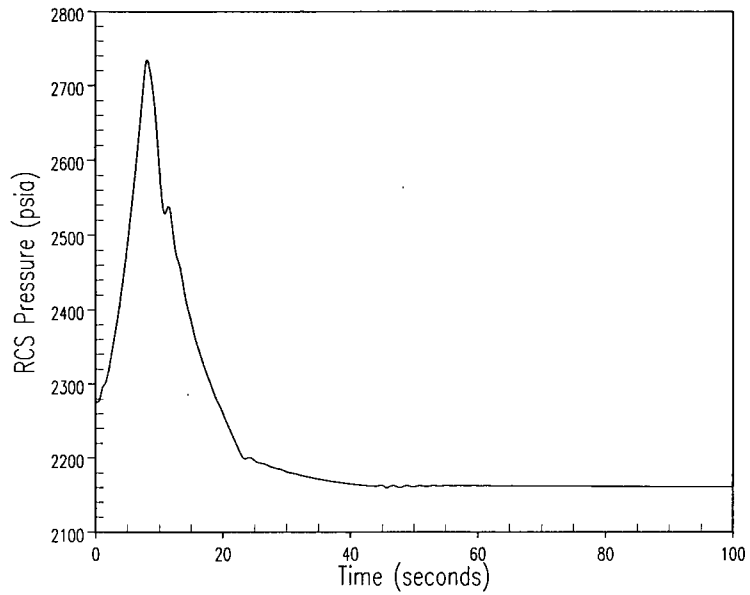


Figure 2.8.5.2.1-14 Unit 1 Loss of Load/Turbine Trip RCS Pressure Case – RCS Pressure and Pressurizer Water Volume Versus Time

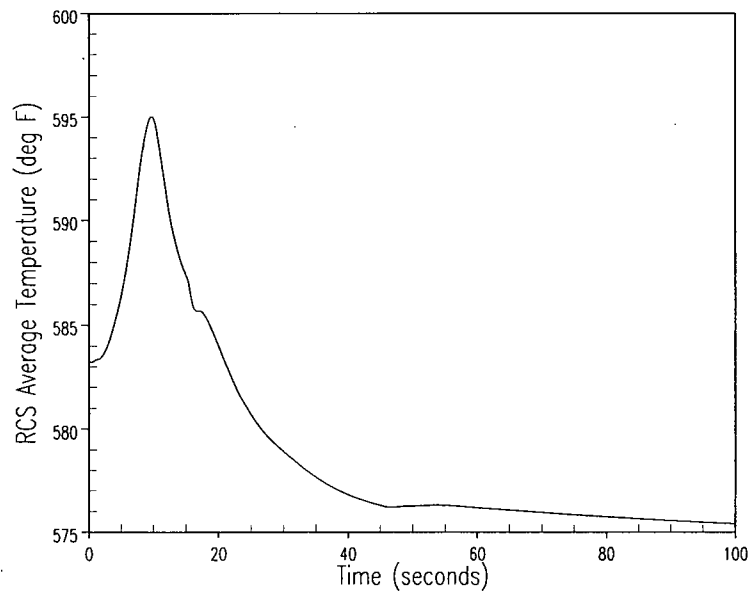


Figure 2.8.5.2.1-15 Unit 1 Loss of Load/Turbine Trip RCS Pressure Case – RCS Average Temperature Versus Time

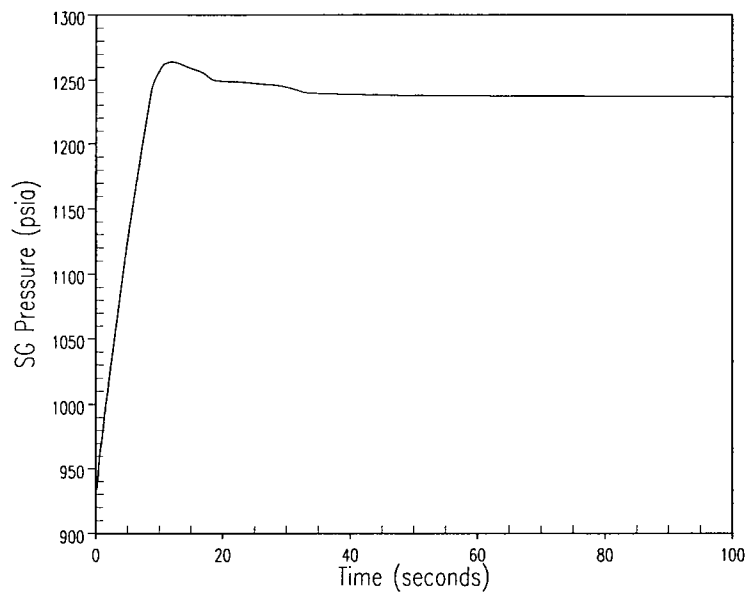
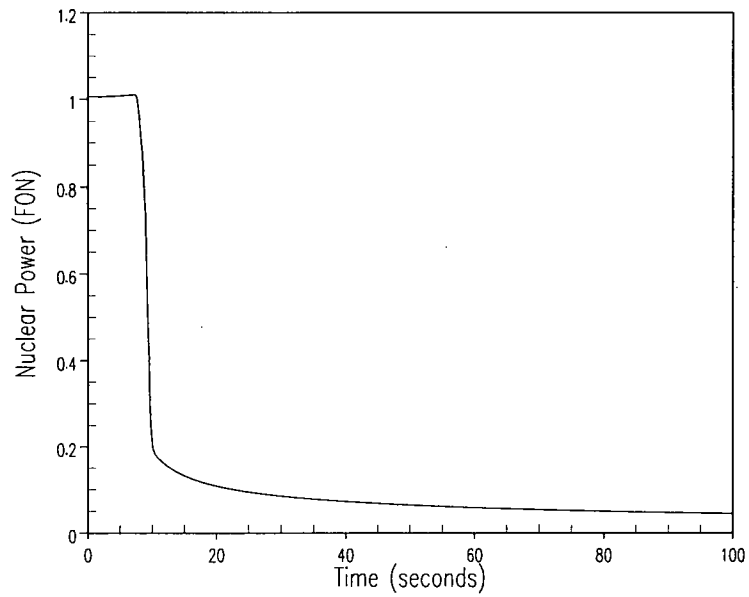


Figure 2.8.5.2.1-16 Unit 2 Loss of Load/Turbine Trip RCS Pressure Case – Nuclear Power and Steam Generator Pressure Versus Time

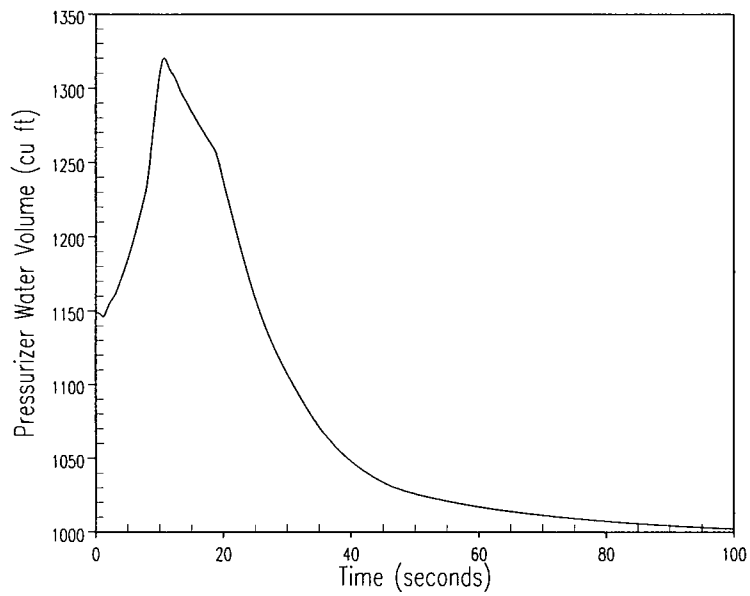
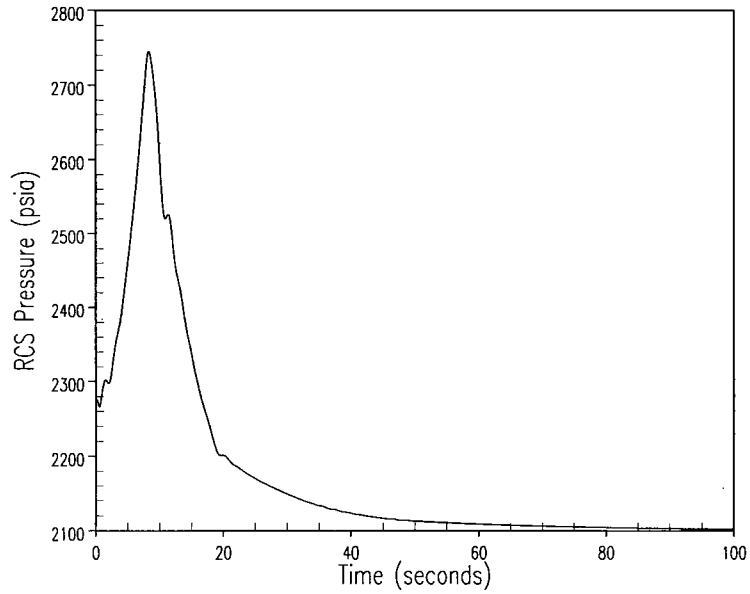


Figure 2.8.5.2.1-17 Unit 2 Loss of Load/Turbine Trip RCS Pressure Case – RCS Pressure and Pressurizer Water Volume Versus Time

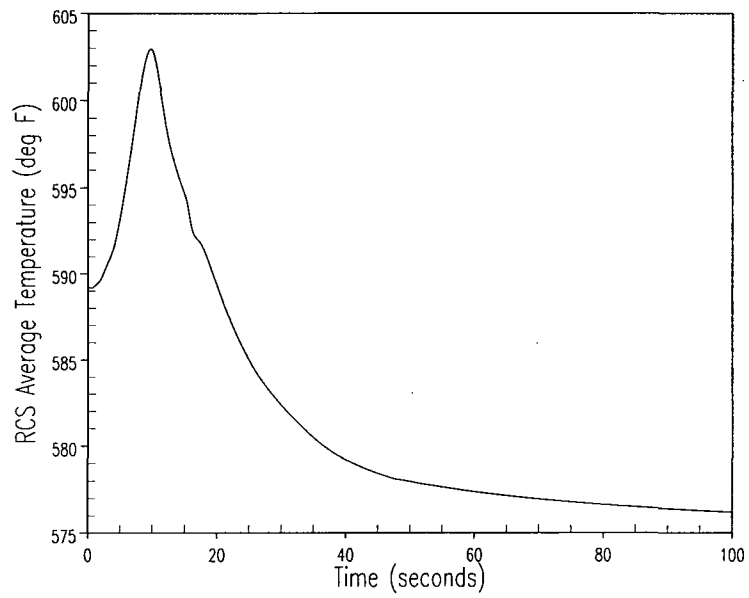


Figure 2.8.5.2.1-18 Unit 2 Loss of Load/Turbine Trip RCS Pressure Case – RCS Average Temperature Versus Time

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

2.8.5.2.2.1 Regulatory Evaluation

The loss of non-emergency AC power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant pumps (RCPs). This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient.

The review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses.

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any normal operations, including anticipated operational occurrences (AOOs).
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation.
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II).

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2. To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.
 3. To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components, including the fuel, reactor vessel internals, and reactivity control systems. Details of the control and protection systems instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the accident analysis of FSAR Chapter 15, which shows that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, reactor coolant pressure boundary (RCPB) components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and of integrated hydrostatic testing of assembled components.

FSAR Chapter 5 discusses the RCS design.

- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control methods are provided. These are rod control cluster assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the full-length control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and AOOs, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCA are presented in FSAR Chapter 4; the operation is discussed in FSAR Chapter 7. The means of controlling the boric acid concentration are described in FSAR Chapter 9. Performance analyses under accident conditions are included in FSAR Chapter 15.

FSAR Section 15.2.6 addresses the impact of a loss of non-emergency AC power to the station auxiliaries. A complete loss of non-emergency AC power may result in a loss of all power to the station auxiliaries, that is, the RCPs, condensate pumps, and so forth. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system. This event is classified as an American Nuclear Society (ANS) Condition II event, fault of moderate frequency.

Upon loss of power to the RCPs, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. FSAR Section 15.2.6.3 concludes that analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or cladding damage.

2.8.5.2.2.2 Technical Evaluation

2.8.5.2.2.2.1 Introduction

A complete loss of non-emergency AC power (FSAR Section 15.2.6) may result in a loss of power to the plant auxiliaries, that is, the RCPs, main feedwater pumps, condensate pumps, etc. The loss of power may be caused by a complete loss-of-the-offsite grid accompanied by a turbine generator trip at the station, or by a loss-of-the-onsite-AC distribution system. The events following a loss-of-AC power with turbine and reactor trip are described in the sequence listed below:

- Plant vital instruments are supplied by emergency DC power sources.
- The steam generator power-operated relief valves (PORVs) are automatically opened to the atmosphere as the steam system pressure rises following the trip. The condenser is assumed unavailable for steam dump. If the relief capacity of the PORVs is inadequate, the main steam safety valves (MSSVs) can lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.
- The steam generator PORVs (or MSSVs, if the PORVs are unavailable) are used to dissipate the residual decay heat and to maintain the plant at the Mode 3 (hot standby) condition as the no-load temperature is approached.
- The emergency diesel generators start on loss of voltage to the plant emergency busses and begin to supply plant vital loads.

The auxiliary feedwater system is started automatically as follows:

- Two motor-driven auxiliary feedwater (MDAFW) pumps are started on any of the following:
 - Low-low water level in two-out-of-four level signals in any steam generator
 - Trip of all main feedwater pumps
 - Safety injection signal
 - Loss of offsite power
 - Manual actuation
 - Anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) actuation signal
- One turbine-driven auxiliary feedwater (TDAFW) pump is started on any of the following:
 - Low-low water level in two-out-of-four level signals in any two of the four steam generators
 - Loss of offsite power
 - Manual actuation
 - AMSAC actuation signal

Following the loss of power to the RCPs, heat removal is maintained by natural circulation in the RCS loops. Following the RCP coastdown, the natural circulation capability of the RCS will remove decay heat from the core, aided by the auxiliary feedwater (AFW) flow in the secondary system. Demonstrating that acceptable results can be obtained for this event proves that the resultant natural circulation flow in the RCS is adequate to remove decay heat from the core.

2.8.5.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

This event is considered to be bounded by other events as described below. Therefore, there are no explicit input parameters or assumptions.

Based on its frequency of occurrence, the loss-of-non-emergency-AC-power accident is considered a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. The following items summarize the acceptance criteria associated with this event:

- Fuel cladding integrity is maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.

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- Pressures in the RCS and main steam system (MSS) are maintained below 110 percent of the design pressures.
 - An incident of moderate frequency does not generate a more serious plant condition without other faults occurring independently.

The first few seconds after a loss-of-AC-power to the RCPs closely resembles the analysis of the complete loss-of-flow event (see Licensing Report (LR) subsection 2.8.5.3.1) in that the RCS experiences a rapid flow reduction transient. This aspect of the loss-of-AC-power event is bounded by the analysis performed for the complete loss-of-flow event that demonstrates that the DNB design basis is met. With respect to overpressurization of the primary and secondary sides, this event is bounded by the loss of load/turbine trip event (see LR subsection 2.8.5.2.1).

The analysis of the loss of normal feedwater event with loss-of-AC-power (see LR subsection 2.8.5.2.3) demonstrates that RCS natural circulation and the AFW system are capable of removing the stored and residual heat. The plant is therefore able to return to a safe condition. A restrictive acceptance criterion that the pressurizer does not become water-solid was used for this event. This criterion establishes the acceptable capacity of the AFW system, ensuring that the pressure criteria and minimum DNBR criterion remained satisfied for the long-term portion of the event, and demonstrated that a more serious plant condition is precluded.

2.8.5.2.2.3 Description of Analyses and Evaluations

As noted above, this event is bounded by events described in other sections of this Licensing Report. Therefore, no explicit analyses were performed.

2.8.5.2.2.4 Results

As noted above, this event is bounded by events described in other sections of this Licensing Report. Therefore, no explicit results are reported here. In addition, the transient response of the RCS following a loss-of-AC-power is less severe than for the loss of normal feedwater with a loss of offsite power event reported in LR subsection 2.8.5.2.3.2.4. Those results demonstrate that the available natural circulation flow is sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

2.8.5.2.2.3 Conclusions

Luminant Power has concluded from the evaluation of the loss of non-emergency AC power to the station auxiliaries event that operation of the plant at the proposed power level is acceptable. It is further concluded that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits are met and the RCPB pressure limits will not be exceeded as a result of the loss of non-emergency AC power to the station auxiliaries. Based on this, the plant will continue to meet the requirements of GDCs -10, -15, and -26 following implementation of the SPU.

Therefore, the SPU is acceptable with respect to the loss of non-emergency AC power to the station auxiliaries event.

2.8.5.2.3 Loss of Normal Feedwater Flow

2.8.5.2.3.1 Regulatory Evaluation

A loss of normal feedwater flow (LONF) could occur from pump failures, valve malfunctions, or a loss-of-offsite power (LOOP). Loss of normal feedwater flow results in an increase in reactor coolant temperature and pressure that eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from the fuel following a LONF. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient.

The review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses.

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any normal operations, including anticipated operational occurrences (AOOs).
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation.
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

- To preclude significant fuel damage during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II).
- To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.
- To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components, including the fuel, reactor vessel internals, and reactivity control systems. Details of the control and protection systems instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the accident analysis of FSAR Chapter 15, which shows that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and of integrated hydrostatic testing of assembled components.

FSAR Chapter 5 discusses the RCS design.

- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control methods are provided. These are rod control cluster assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation the shutdown rod banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the full-length control banks, are

designed to shut down the reactor with adequate margin under conditions of normal operation and AOOs, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCA are presented in FSAR Chapter 4. The operation is discussed in FSAR Chapter 7. The means of controlling the boric acid concentration are described in FSAR Chapter 9. Performance analyses under accident conditions are included in FSAR Chapter 15.

FSAR Section 15.2.7 addresses the impact of a loss of normal feedwater flow. It states that a loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a departure from nucleate boiling (DNB) condition. This event is classified as a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973, a fault of moderate frequency.

The analysis of the LONF flow transient considers two cases with four loops operating initially. The first is a case where offsite AC power is maintained, and the second is a case where offsite AC power is lost.

FSAR Section 15.2.7.3 concludes that the results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves, and the water level in all steam generators is maintained above the tubesheets.

2.8.5.2.3.2 Technical Evaluation

2.8.5.2.3.2.1 Introduction

A LONF flow (FSAR Section 15.2.7) (from pump failures, valve malfunctions, or a complete loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater is not supplied, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer could occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables do not approach a condition that causes a DNB ratio (DNBR) limit violation.

Two scenarios are analyzed for a LONF event. The first is a case where offsite AC power is maintained, and the second is a case where offsite AC power is lost, which results in reactor coolant pump coastdown as discussed in Licensing Report (LR) subsection 2.8.5.2.2.

The following events occur following the reactor trip for the LONF:

- The steam generator power-operated relief valves (PORVs) are automatically opened to the atmosphere as the steam system pressure rises following a loss of feedwater. The condenser is assumed unavailable for steam dump. If the relief capacity of the PORVs is inadequate, the main steam safety valves (MSSVs) can lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.
- Plant vital instruments are supplied from emergency DC power sources for the case with a loss of offsite power.

The following provide the necessary protection in the event of a LONF:

- The reactor can be tripped on one or more of the following reactor trip signals:
 - Pressurizer high pressure trip signal
 - Overtemperature N-16 trip signal
 - Low-low steam generator water level trip signal in any steam generator
- Two motor-driven auxiliary feedwater (MDAFW) pumps are started on any of the following:
 - Low-low water level in two-out-of-four level signals in any steam generator
 - Trip of all main feedwater pumps
 - Safety injection signal
 - Loss of offsite power
 - Manual actuation
 - Anticipated transient without SCRAM (ATWS) mitigation system actuation circuitry (AMSAC) actuation signal
- One turbine-driven auxiliary feedwater (TDAFW) pump is started on any of the following:
 - Low-low water level in two-out-of-four level signals in any two of the four steam generators
 - Loss of offsite power

-
- Manual actuation
 - AMSAC actuation signal
 - The MSSVs open to provide an additional heat sink and protection against secondary side overpressure.
 - The pressurizer safety valves (PSVs) may open to provide protection against overpressure of the RCS.

The analysis showed that following a LONF (with or without offsite power), the auxiliary feedwater (AFW) system is capable of removing the stored and residual heat, thus preventing overpressurization of the RCS, overpressurization of the secondary side, water relief from the pressurizer, and uncovery of the reactor core.

2.8.5.2.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The following assumptions were made in the LONF analyses:

- The plant is initially operating at 100.6 percent of the nuclear steam supply system (NSSS) power of 3,628 MWt.
 -
- For the case with offsite power, a maximum reactor coolant pump (RCP) heat of 20.0 MWt was conservatively modeled. The RCPs were assumed to continuously operate throughout the transient providing a constant reactor coolant volumetric flow equal to the thermal design flow value. Although not assumed in this case, the RCPs could be manually tripped at some later time in the transient to reduce the heat addition to the RCS caused by the operation of the pumps.
- For the case without offsite power, power was assumed to be lost to the RCPs after the start of rod motion. For this case, the nominal RCP heat of 16.0 MWt was modeled. Assuming a nominal RCP heat was conservative since the RCPs coasted down and ceased to add heat to the primary coolant while the core decay heat was based on a slightly higher initial core power. The post-trip heat removal from the core relied upon natural circulation flow in the RCS loops.
- Main feedwater temperature conditions at 390° and 450.3°F were analyzed.
- Reactor vessel average coolant temperature (T_{avg}) conditions at the low and high ends of the full-power temperature window (574.2° to 589.2°F) were considered. In addition, since the pressurizer level program has a breakpoint at 584.7°F (that is, pressurizer level program is linear from 25-percent span at the no-load temperature of 557°F to 60-percent span at a full-power temperature of 584.7°F) that point was also specifically analyzed.

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- The direction of conservatism for both initial reactor vessel average coolant temperature and pressurizer pressure can vary. As such, cases were considered with the initial temperature and pressure uncertainties applied in each direction. The initial average temperature uncertainty was assumed to be $\pm 6.0^\circ\text{F}$. The initial pressurizer pressure uncertainty was assumed to be ± 30 psi.
 - Reactor trip occurs on steam generator low-low water level at 0 percent of the narrow-range span for Unit 1 with the Model $\Delta 76$ steam generators and at 10 percent of the narrow-range span for Unit 2 with the Model D-5 steam generators.
 - It was assumed that two MDAFW pumps are available to supply flow to all four steam generators, 60 seconds following a low-low steam generator water level signal. The worst single failure for this analysis is the loss of the TDAFW pump.
 - The pressurizer heaters were modeled to exacerbate the heatup and volumetric expansion of the water in the pressurizer. In addition, the pressurizer sprays were assumed to be operable, and cases were analyzed with and without the PORVs available to determine the limiting configuration. It was found that the cases without PORV availability were more limiting.
 - Secondary system steam relief is achieved through the self-actuated MSSVs. Note that steam relief would normally be provided by the PORVs or condenser dump valves for most LONF cases. However, the condenser dump valves and the PORVs were assumed to be unavailable.
 - The MSSVs were modeled assuming a 3-percent tolerance and an accumulation model that assumes that the valves were wide open once the pressure exceeded the setpoint (plus tolerance) by 5 psi (accumulation).
 - Core residual heat generation was based on the 1979 version of ANS 5.1 (Reference 1). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip was assumed.
 - Steam generator tube plugging (SGTP) levels of both 0 percent and 10 percent were analyzed.

Based on its frequency of occurrence, the LONF accident is considered a Condition II event as defined by the ANS. The following items summarize the acceptance criteria associated with this event:

- Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit.
- Pressures in the RCS and main steam system (MSS) are maintained below 110 percent of the design pressures.
- An incident of moderate frequency does not generate a more serious plant condition without other faults occurring independently.

With respect to overpressurization, the LONF event, both with and without offsite power, is bounded by the loss of load/turbine trip (LOL/TT) event discussed in Licensing Report (LR) subsection 2.8.5.2.1 in which assumptions are made to conservatively calculate the RCS and MSS pressure transients. For the LONF event, turbine trip occurs after reactor trip, whereas for LOL/TT the turbine trip is the initiating fault. Therefore, the primary/secondary power mismatch and resultant RCS and MSS heatup and pressurization transients are always more severe for LOL/TT than for LONF.

With respect to DNB, the LONF event with offsite power is also bounded by the LOL/TT event. Both of these events represent a reduction in the heat removal capability of the secondary system. For the LONF event, the RCS temperature increases gradually as the steam generators boil down to the low-low level trip setpoint, at which time reactor trip occurs, followed by turbine trip. For the LOL/TT event, the turbine trip is the initiating event, and the loss of heat sink is much more severe. As such, the initial RCS heatup will be much more severe for the LOL/TT event than for the LONF event, and the LOL/TT event will always be more severe with respect to the minimum DNBR criterion. The LONF event without offsite power is bounded by the complete loss-of-flow event discussed in LR subsection 2.8.5.3.1. The DNBR consequences of the LONF event without offsite power are similar to those of the LONF event with offsite power, with the additional effect of a reduction in the core flow rate caused by loss of power to the RCPs. However, the LONF event without offsite power is bounded by the complete loss-of-flow event, for which the RCP coastdown is the initiating fault and the reactor trip occurs when the core flow is already degraded.

The restrictive acceptance criterion that the pressurizer does not reach a water-solid condition was used for this event. This criterion demonstrates that the capacity of the AFW system is sufficient to dissipate core residual heat, stored energy, and RCP heat (for the cases with offsite power available) without the event progressing into a more serious plant condition (water relief through the pressurizer PORVs or safety valves is precluded).

2.8.5.2.3.2.3 Description of Analyses and Evaluations

A detailed analysis using the RETRAN (Reference 2) computer code was performed to determine the plant transient conditions following a LONF. The code modeled the core neutron kinetics, RCS, pressurizer, pressurizer heaters, pressurizer sprays, steam generators, MSSVs, and the AFW system. The code also computed pertinent variables, including the pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

2.8.5.2.3.2.4 Results

LONF Flow Results with Offsite Power

For Unit 1 with the Model $\Delta 76$ steam generators, the most limiting LONF case with offsite power available was that with the temperature uncertainty subtracted from the nominal T_{avg} value at the low end of the full-power temperature window (that is, 574.2°F - 6°F), pressure uncertainty

added to the nominal value (that is, 2,250 psia + 30 psi), PORVs not available, 0-percent SGTP, and high main feedwater temperature (450.3°F) conditions.

For Unit 2 with the Model D-5 steam generators, the most limiting LONF case with offsite power available was that with the temperature uncertainty added to the nominal T_{avg} value at the high end of the full-power temperature window (that is, 589.2°F + 6°F), pressure uncertainty subtracted from the nominal value (that is, 2,250 psia - 30 psi), PORVs not available, 10-percent SGTP, and low main feedwater temperature (390.0°F) conditions.

The calculated sequence of events for this event is listed in Tables 2.8.5.2.3-1 and 2.8.5.2.3-2 for Units 1 and 2, respectively. Figures 2.8.5.2.3-1 through 2.8.5.2.3-4 (Unit 1) and Figures 2.8.5.2.3-5 through 2.8.5.2.3-8 (Unit 2) present transient plots of the significant plant parameters following a LONF with offsite power for the limiting cases discussed above for Units 1 and 2. Note that there are noticeable differences between Unit 1 and Unit 2 in some of the transient trends shown in the figures, e.g., pressurizer pressure. These differences are due to the fact that the pressurizer safety valves actuate in the limiting Unit 2 case, but not the limiting Unit 1 case.

Following the reactor and turbine trip from full load, the water level in the steam generators fell due to a reduction of the steam generator void fraction and because steam flow through the safety valves continued to dissipate the stored and generated heat. One minute following the initiation of the low-low steam generator level trip, the MDAFW pumps automatically started, consequently reducing the rate at which the steam generator water level was decreasing.

The capacity of the MDAFW pumps enabled sufficient heat transfer from each steam generator to dissipate the core residual heat without the pressurizer reaching a water solid condition (as shown in Figures 2.8.5.2.3-3 and 2.8.5.2.3-7 for Units 1 and 2, respectively). This precluded any water relief through the RCS pressurizer relief valves or PSVs.

LONF Flow Results without Offsite Power

For Unit 1 with the Model $\Delta 76$ steam generators, the most limiting LONF case without offsite power available was with the temperature uncertainty subtracted from the nominal T_{avg} value at the low end of the full-power temperature window (that is, 574.2°F - 6°F), pressure uncertainty added to the nominal value (that is, 2,250 psia + 30 psi), PORVs not available, 0-percent SGTP, and low main feedwater temperature (390.0°F) conditions.

For Unit 2 with the Model D-5 steam generators, the most limiting LONF case without offsite power available was with the temperature uncertainty added to the nominal T_{avg} value at the high end of the full-power temperature window (that is, 589.2°F + 6°F), pressure uncertainty subtracted from the nominal value (that is, 2,250 psia - 30 psi), PORVs not available, 10-percent SGTP, and low main feedwater temperature (390.0°F) conditions.

The calculated sequence of events for this event is listed in Tables 2.8.5.2.3-3 and 2.8.5.2.3-4 for Units 1 and 2, respectively. Figures 2.8.5.2.3-9 through 2.8.5.2.3-12 (Unit 1) and Figures 2.8.5.2.3-13 through 2.8.5.2.3-16 (Unit 2) present transient plots of the significant plant

parameters following a LONF without offsite power for the limiting cases discussed above for Units 1 and 2. Note that the pressurizer safety valves actuate in both the limiting Unit 1 case and the limiting Unit 2 case.

Following the reactor and turbine trip from full load, the water level in the steam generators fell due to a reduction of the steam generator void fraction and because steam flow through the safety valves continued to dissipate the stored and generated heat. One minute following the initiation of the low-low steam generator level trip, the MDAFW pumps automatically started, consequently reducing the rate at which the steam generator water level was decreasing.

The capacity of the MDAFW pumps enabled sufficient heat transfer from each steam generator to dissipate the core residual heat without the pressurizer reaching a water-solid condition (as shown in Figures 2.8.5.2.3-11 and 2.8.5.2.3-15 for Units 1 and 2, respectively). This precluded any water relief through the RCS pressurizer relief valves or PSVs.

The results of the analysis showed that the pressurizer did not reach a water-solid condition. Based on this, the LONF event will not progress into a more serious plant condition. Also, as discussed in LR subsection 2.8.5.2.3.2.2, with respect to overpressurization, the LONF event is bounded by the LOL/TT event. With respect to DNB, the LONF event with offsite power is bounded by the LOL/TT event and the LONF event without offsite power is bounded by the complete loss-of-flow event. Therefore, the LONF event will not adversely affect the core, the RCS, or the MSS.

2.8.5.2.3.3 Conclusion

The analyses of the LONF event were reviewed and Luminant Power has concluded that the analyses have adequately accounted for operation of the plant at the proposed uprated power level and was performed using acceptable analytical models. It is further concluded that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits are met and the RCPB pressure limits will not be exceeded as a result of the LONF. Based on this, the plant will continue to meet the requirements of GDCs -10, -15, and -26 following implementation of the SPU. Therefore, the SPU is acceptable with respect to the LONF event.

2.8.5.2.3.4 References

1. ANSI/ANS-5.1 – 1979, “American National Standard for Decay Heat Power in Light Water Reactors,” August 1979.
2. WCAP-14882, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” April 1999.

Table 2.8.5.2.3-1	
Time Sequence of Events – Unit 1 LONF with Offsite Power	
Event	Time (sec)
Main Feedwater Flow Stops	0.0
Low-Low Steam Generator Water Level Reactor Trip Setpoint Reached	50.7
Rods Begin to Drop	52.7
Flow from Two MDAFW Pumps is Initiated	110.7
Long-Term Peak Water Level in Pressurizer Occurs	5,484.0

Table 2.8.5.2.3-2	
Time Sequence of Events – Unit 2 LONF with Offsite Power	
Event	Time (sec)
Main Feedwater Flow Stops	0.0
Low-Low Steam Generator Water Level Reactor Trip Setpoint Reached	61.5
Rods Begin to Drop	63.5
Flow from Two MDAFW Pumps is Initiated	121.5
Long-Term Peak Water Level in Pressurizer Occurs	343.0

Table 2.8.5.2.3-3	
Time Sequence of Events – Unit 1 LONF without Offsite Power	
Event	Time (sec)
Main Feedwater Flow Stops	0.0
Low-Low Steam Generator Water Level Reactor Trip Setpoint Reached	59.1
Rods Begin to Drop	61.1
Reactor Coolant Pumps Tripped	63.1
Flow from Two MDAFW Pumps is Initiated	119.1
Long-Term Peak Water Level in Pressurizer Occurs	2,198.0

Table 2.8.5.2.3-4	
Time Sequence of Events – Unit 2 LONF without Offsite Power	
Event	Time (sec)
Main Feedwater Flow Stops	0.0
Low-Low Steam Generator Water Level Reactor Trip Setpoint Reached	55.1
Rods Begin to Drop	57.1
Reactor Coolant Pumps Tripped	59.1
Flow from Two MDAFW Pumps is Initiated	115.1
Long-Term Peak Water Level in Pressurizer Occurs	1,672.0

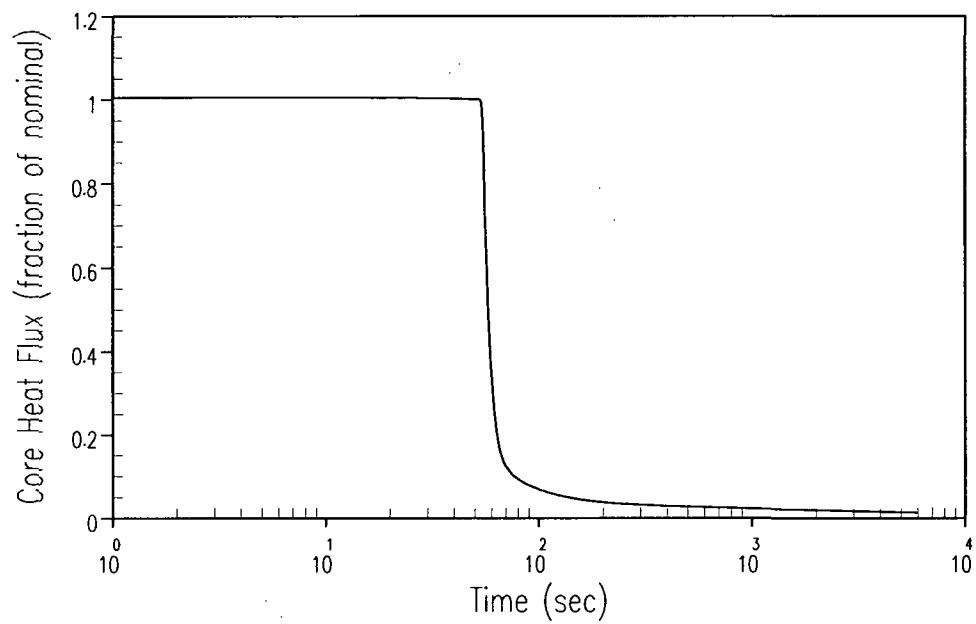
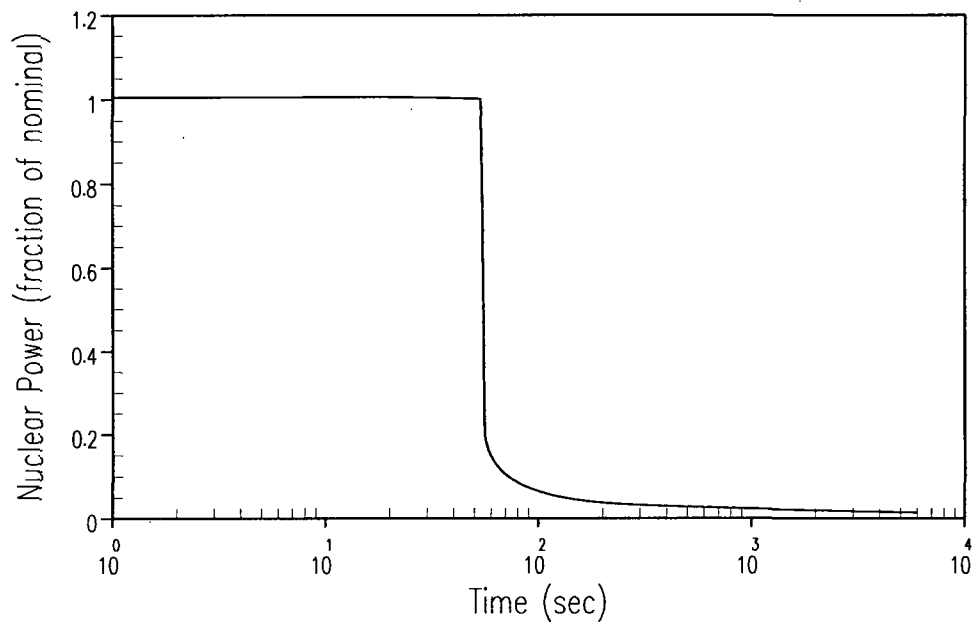
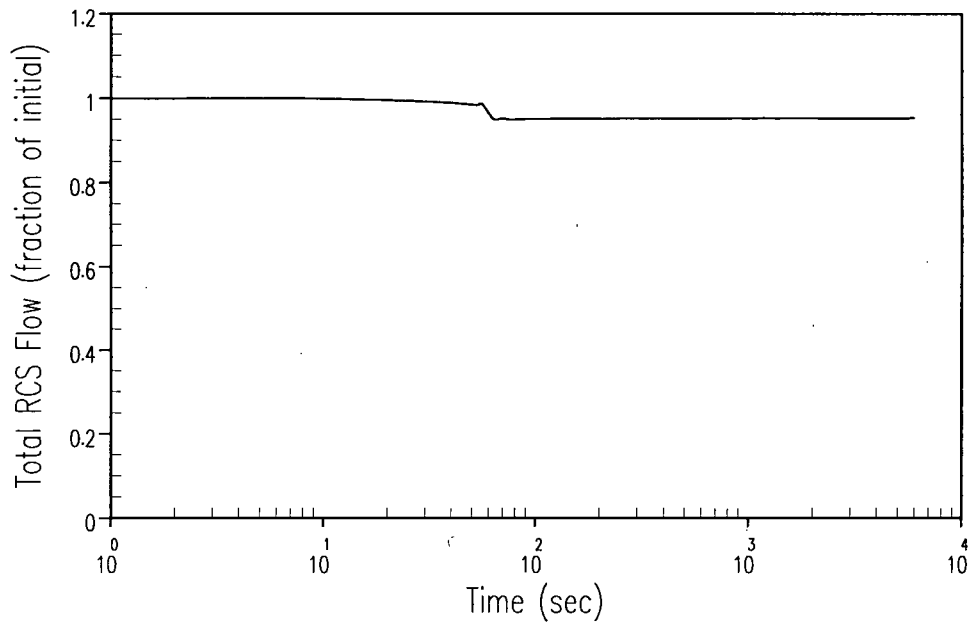


Figure 2.8.5.2.3-1 Unit 1 LONF with Offsite Power – Nuclear Power and Core Average Heat Flux Versus Time



— Hot Leg
 - - - Cold Leg
 - · - Loop Average

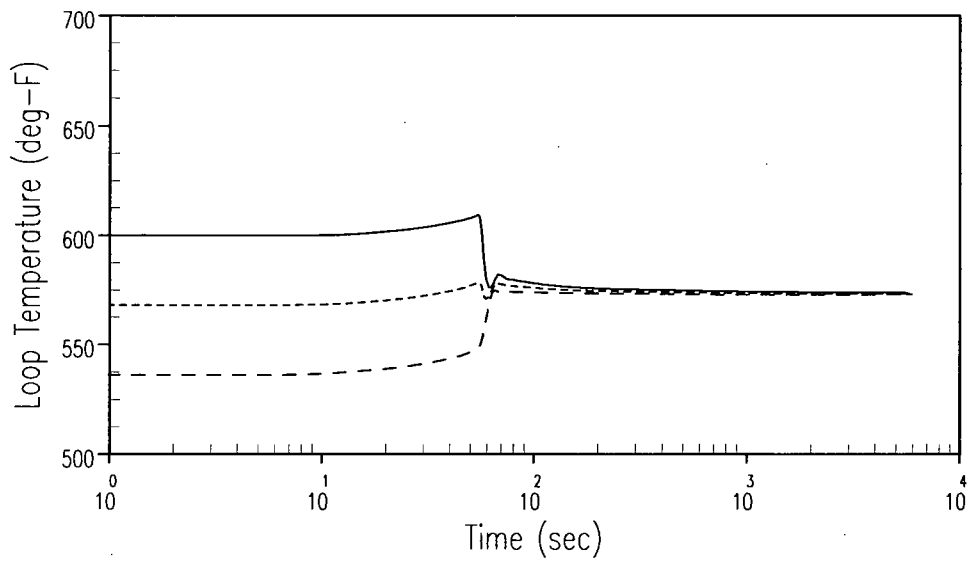


Figure 2.8.5.2.3-2 Unit 1 LONF with Offsite Power – Reactor Coolant Flow Rate and Loop Temperature Versus Time

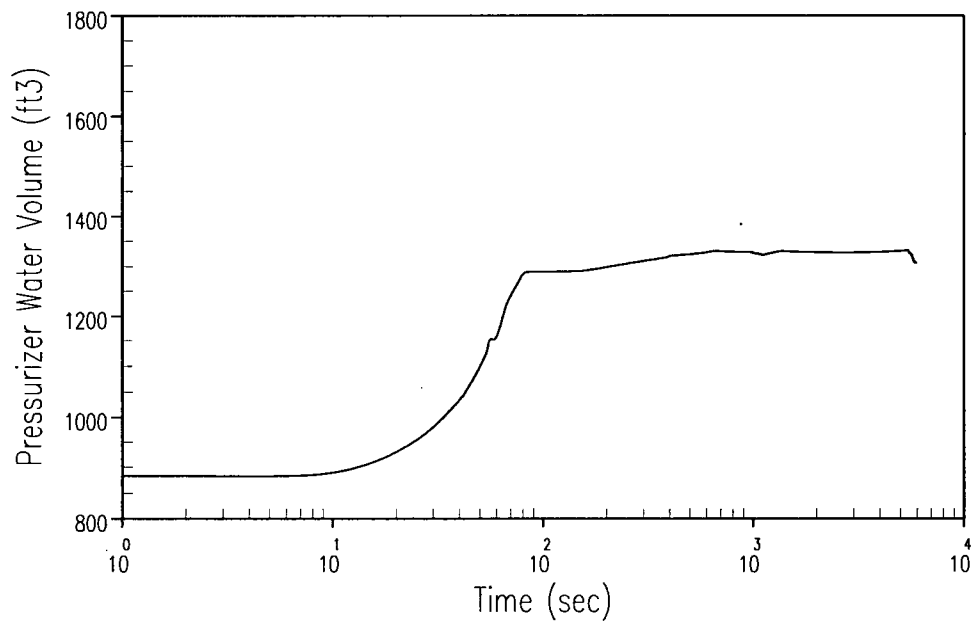
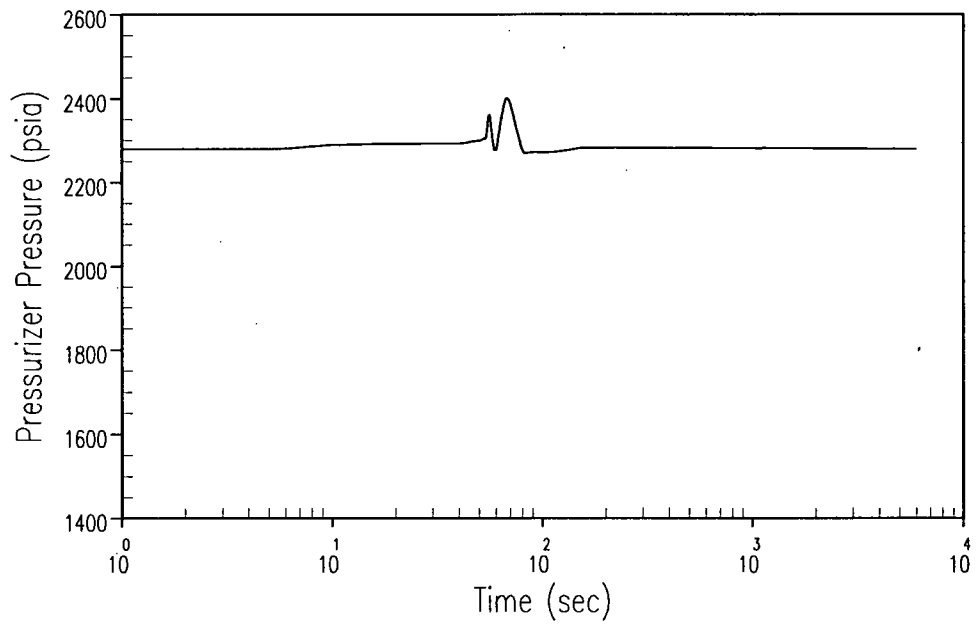


Figure 2.8.5.2.3-3 Unit 1 LONF with Offsite Power– Pressurizer Pressure and Water Volume Versus Time

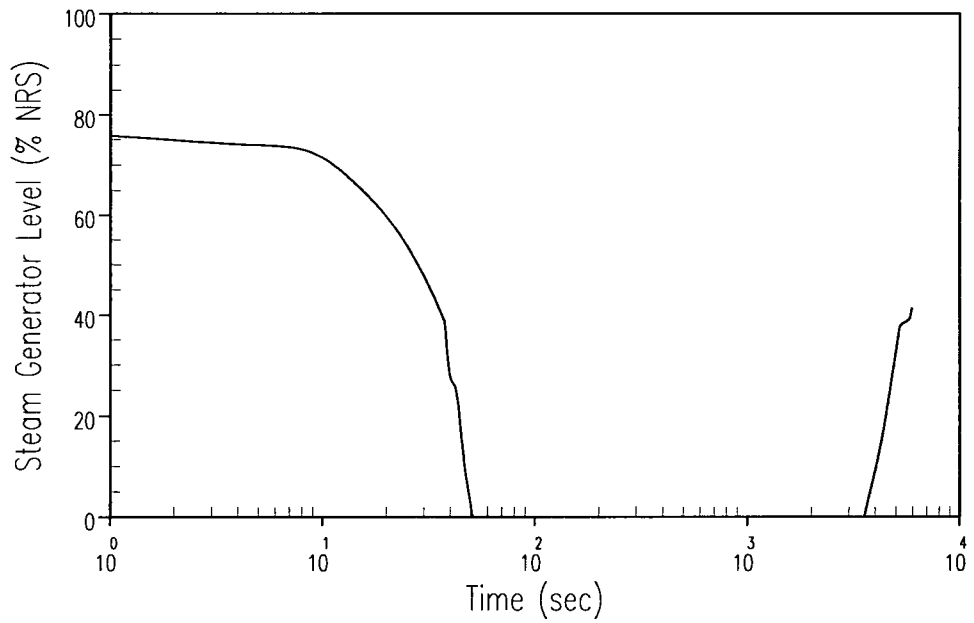
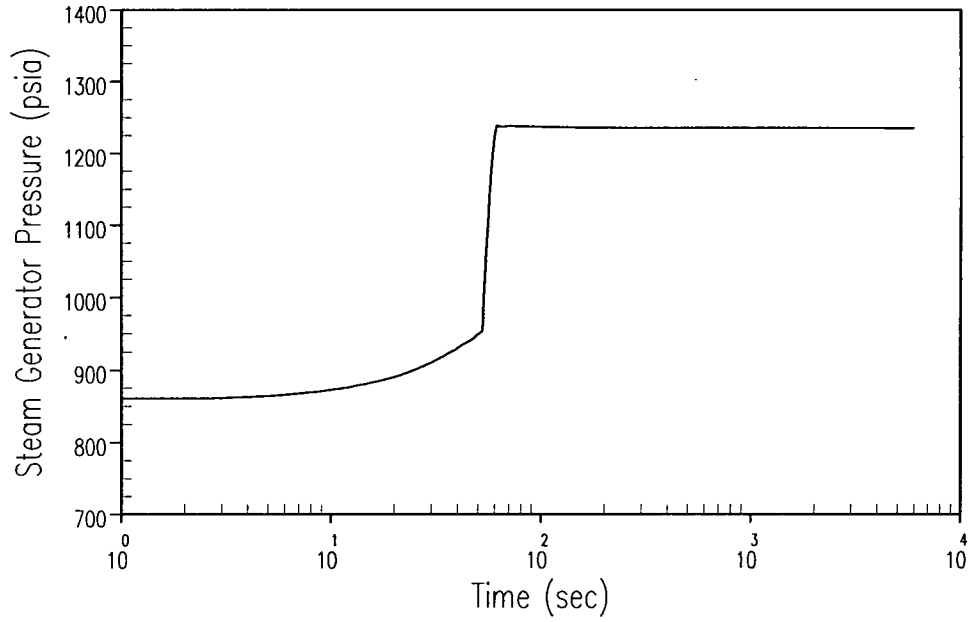


Figure 2.8.5.2.3-4 Unit 1 LONF with Offsite Power – Steam Generator Pressure and Level Versus Time

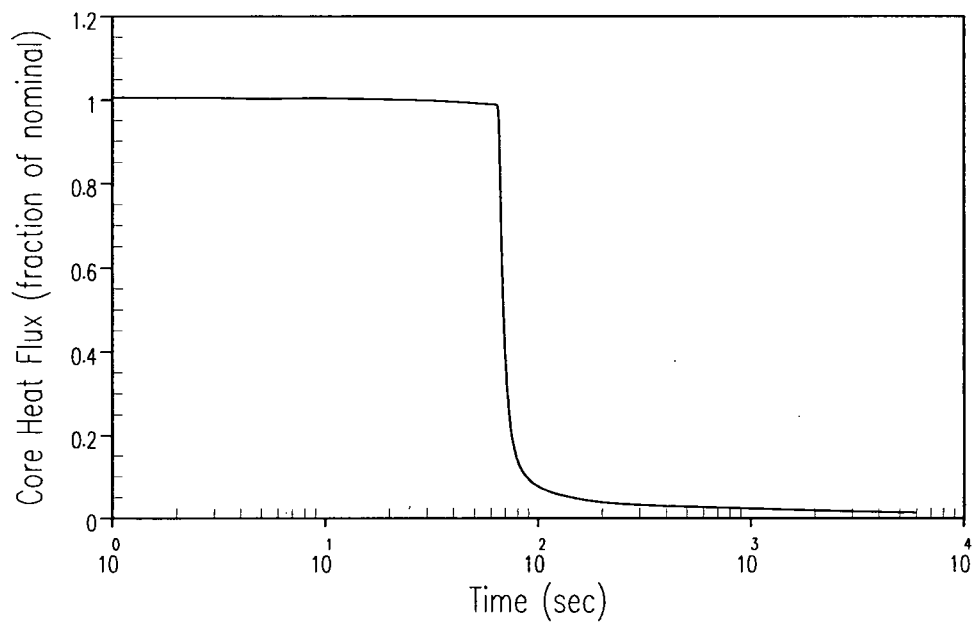
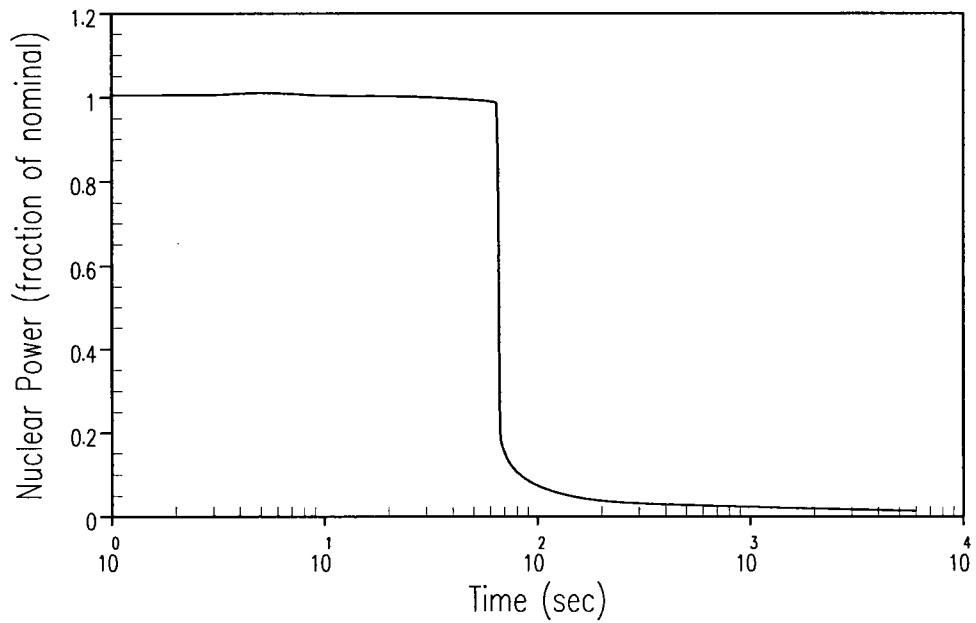


Figure 2.8.5.2.3-5 Unit 2 LONF with Offsite Power – Nuclear Power and Core Average Heat Flux Versus Time

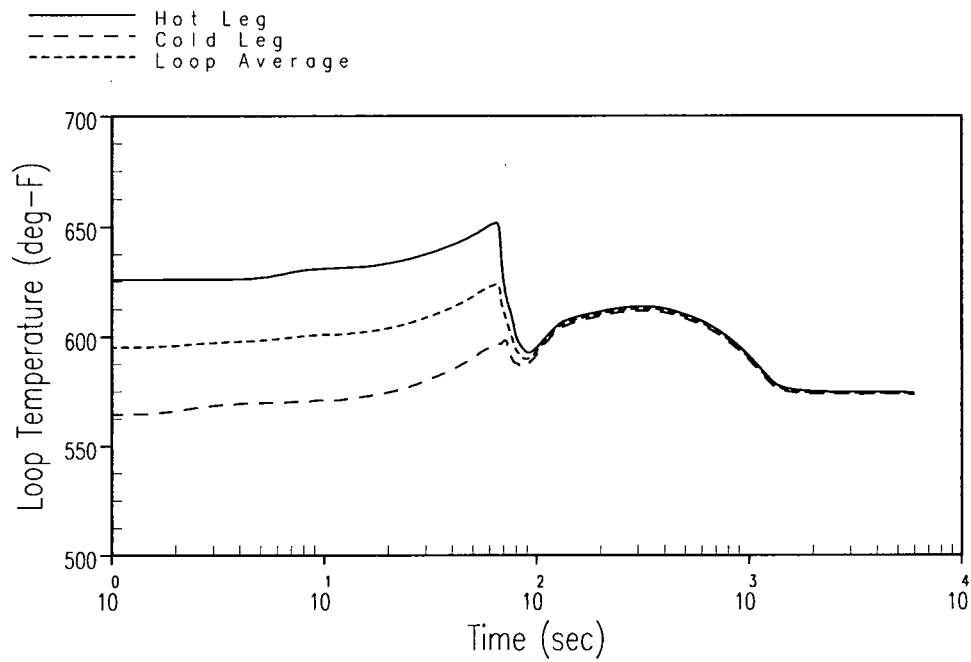
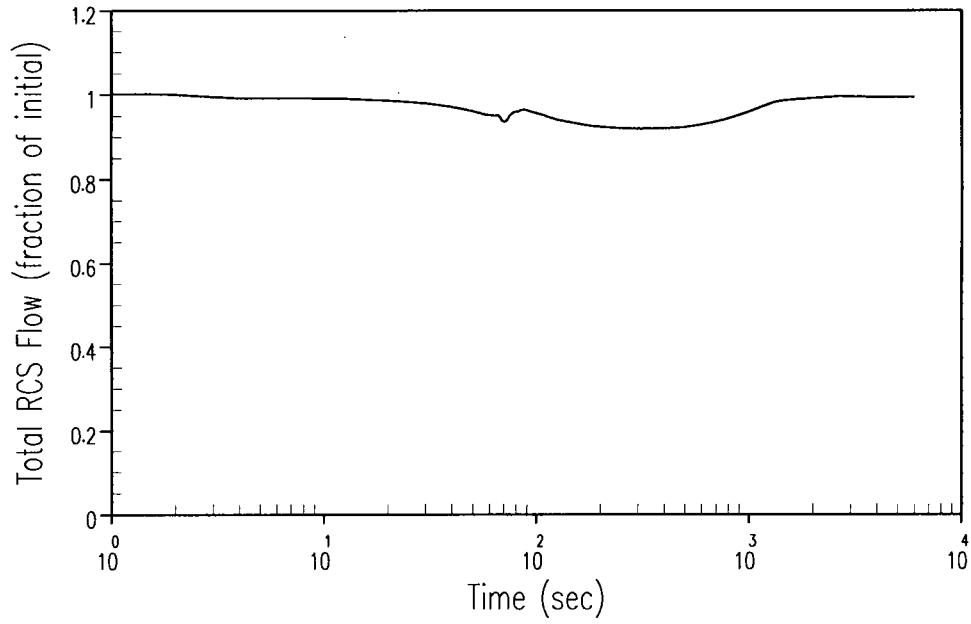


Figure 2.8.5.2.3-6 Unit 2 LONF with Offsite Power – Reactor Coolant Flow Rate and Loop Temperature Versus Time

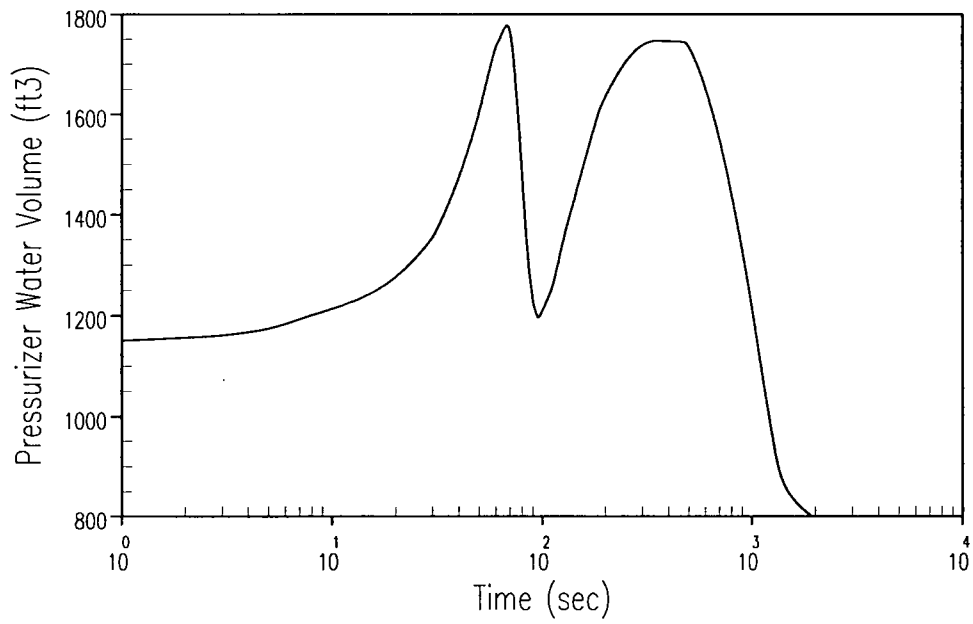
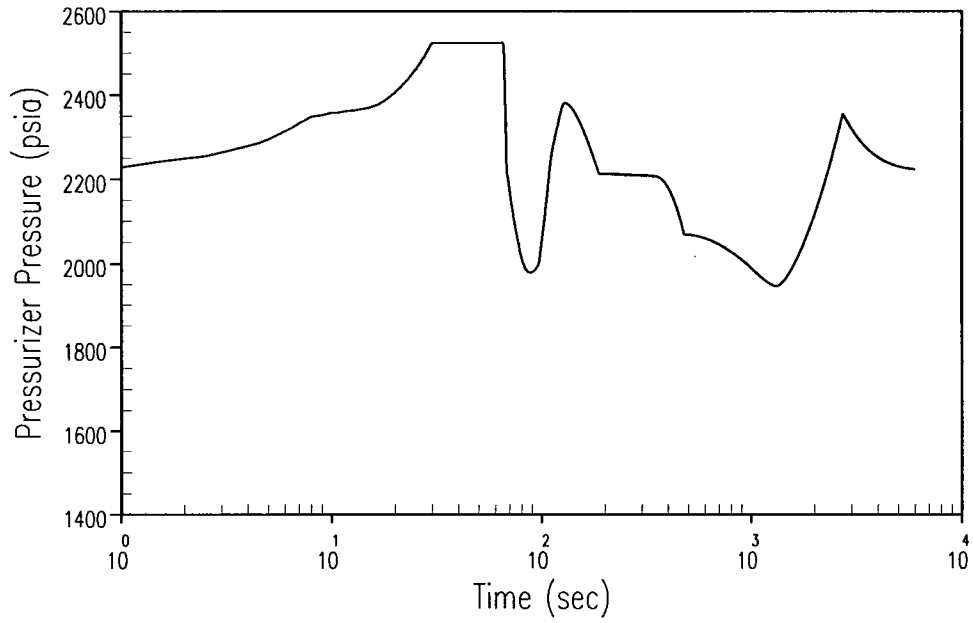


Figure 2.8.5.2.3-7 Unit 2 LONF with Offsite Power – Pressurizer Pressure and Water Volume Versus Time

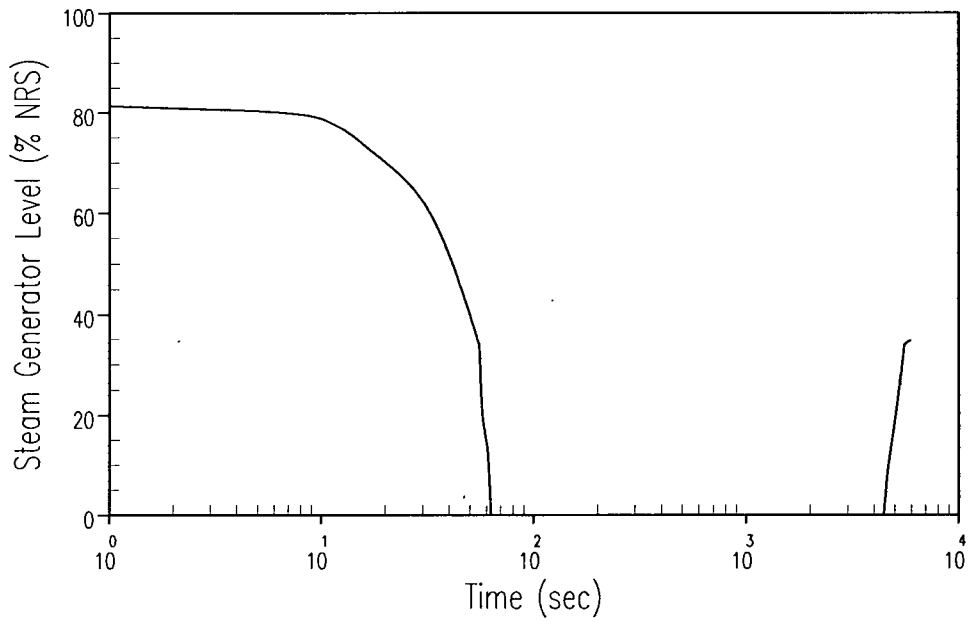
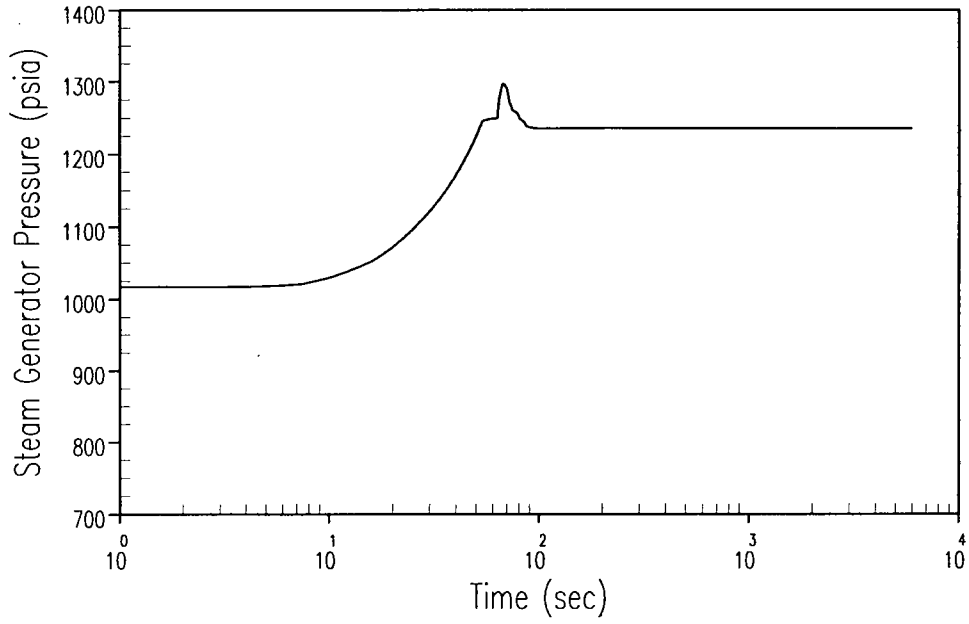


Figure 2.8.5.2.3-8 Unit 2 LONF with Offsite Power – Steam Generator Pressure and Level Versus Time

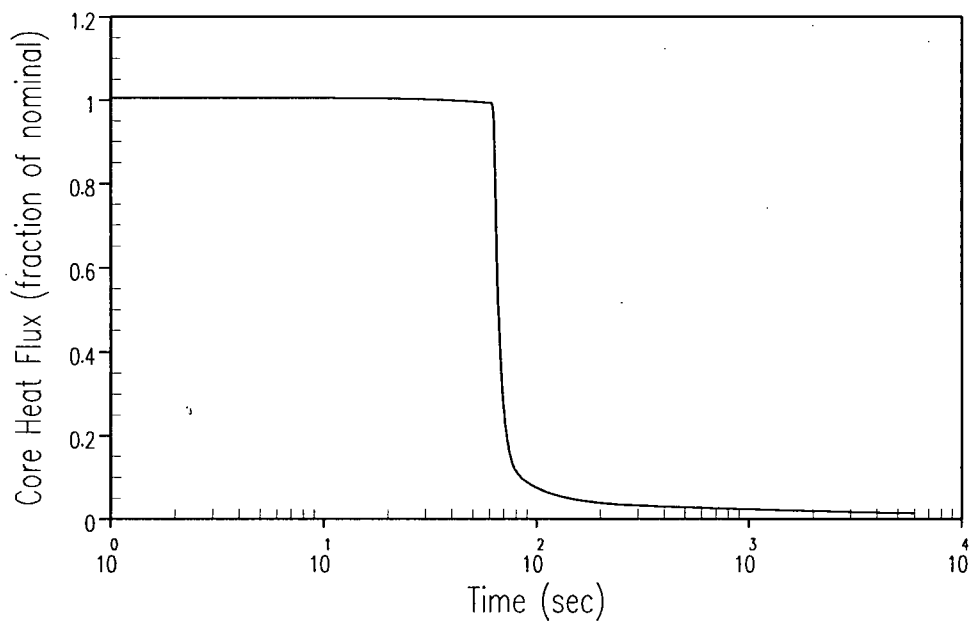
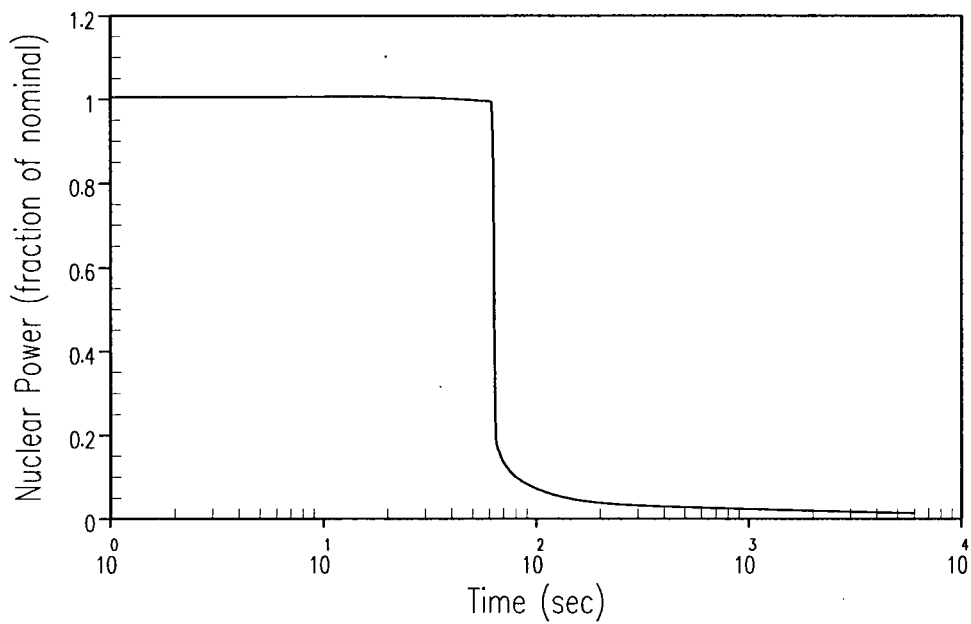
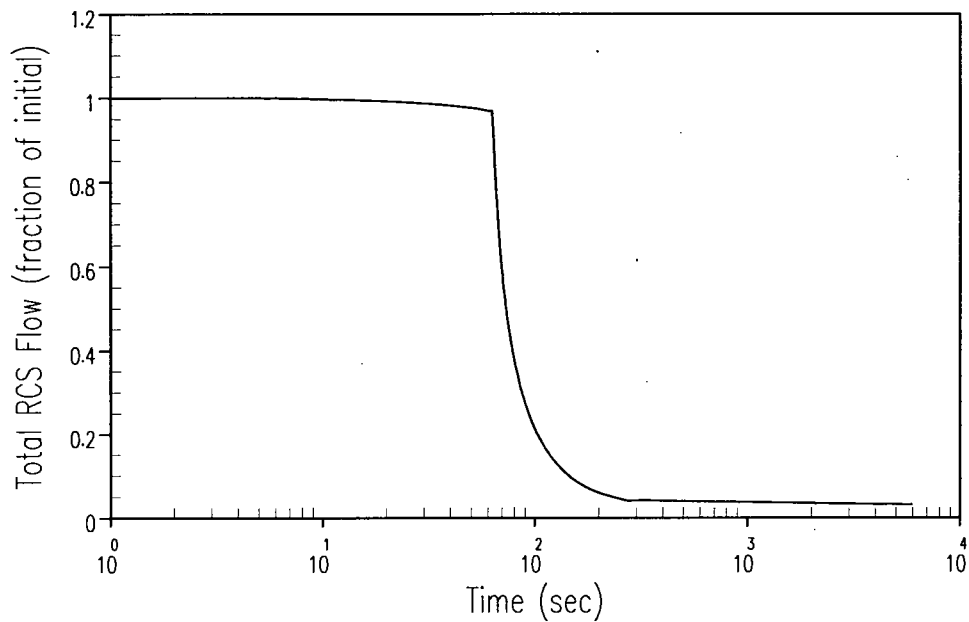


Figure 2.8.5.2.3-9 Unit 1 LONF without Offsite Power – Nuclear Power and Core Average Heat Flux Versus Time



— Hot Leg
 - - - Cold Leg
 - · - · Loop Average

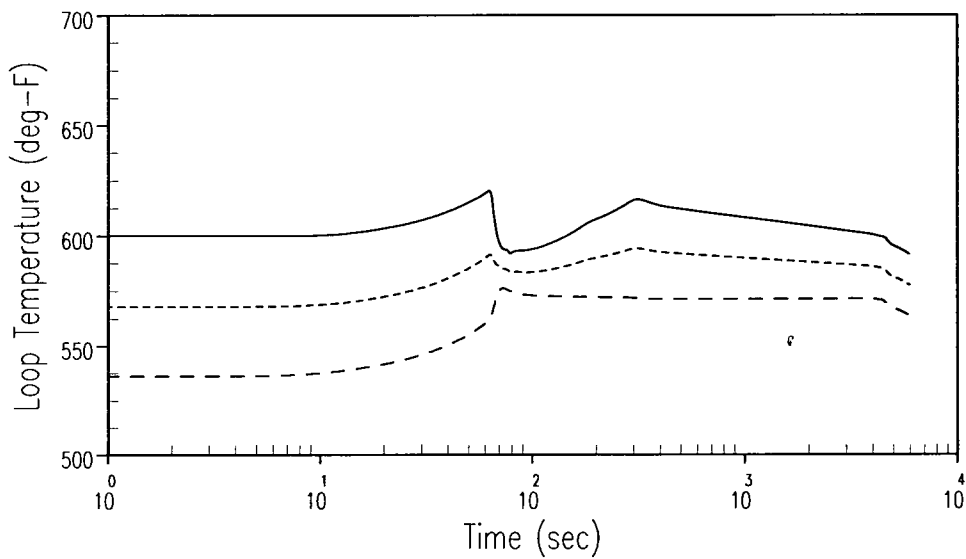


Figure 2.8.5.2.3-10 Unit 1 LONF without Offsite Power – Reactor Coolant Flow Rate and Loop Temperature Versus Time

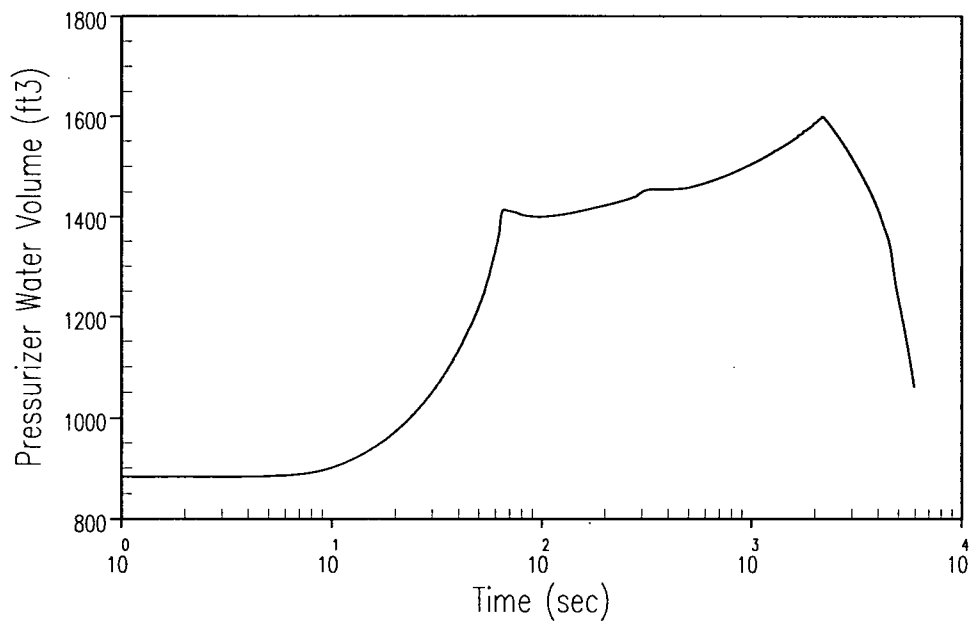
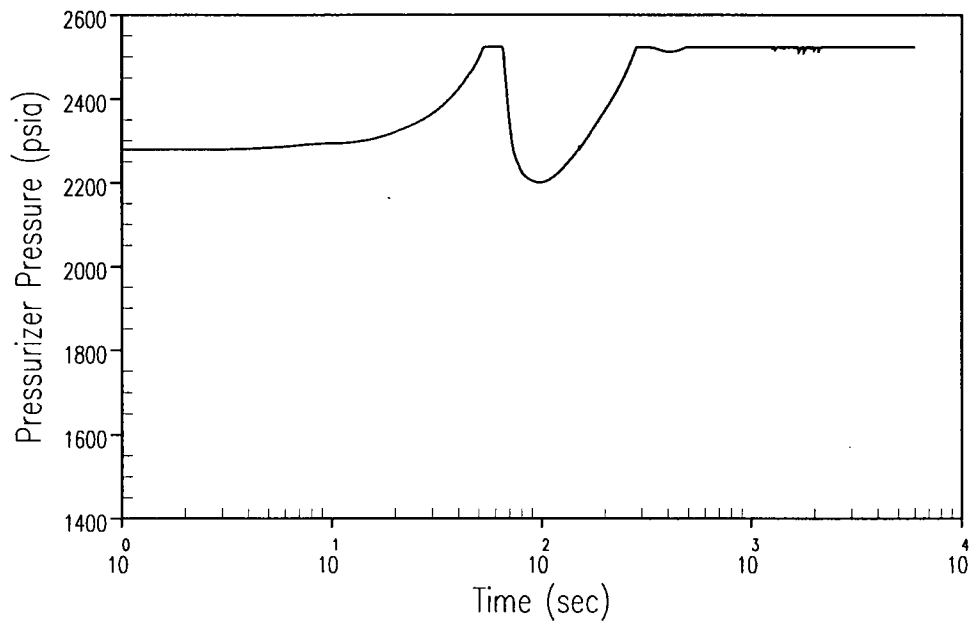


Figure 2.8.5.2.3-11 Unit 1 LONF without Offsite Power – Pressurizer Pressure and Water Volume Versus Time

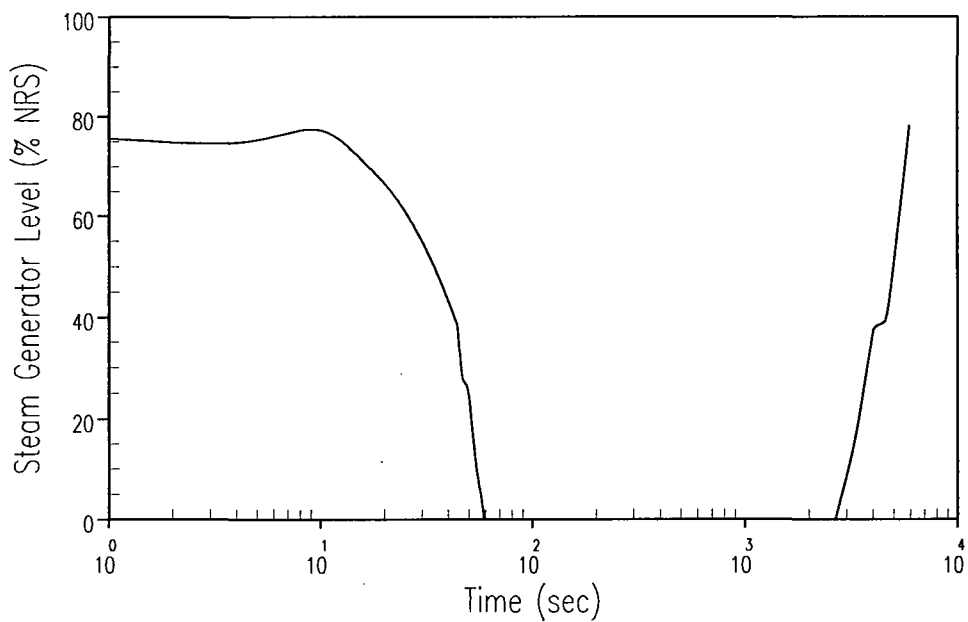
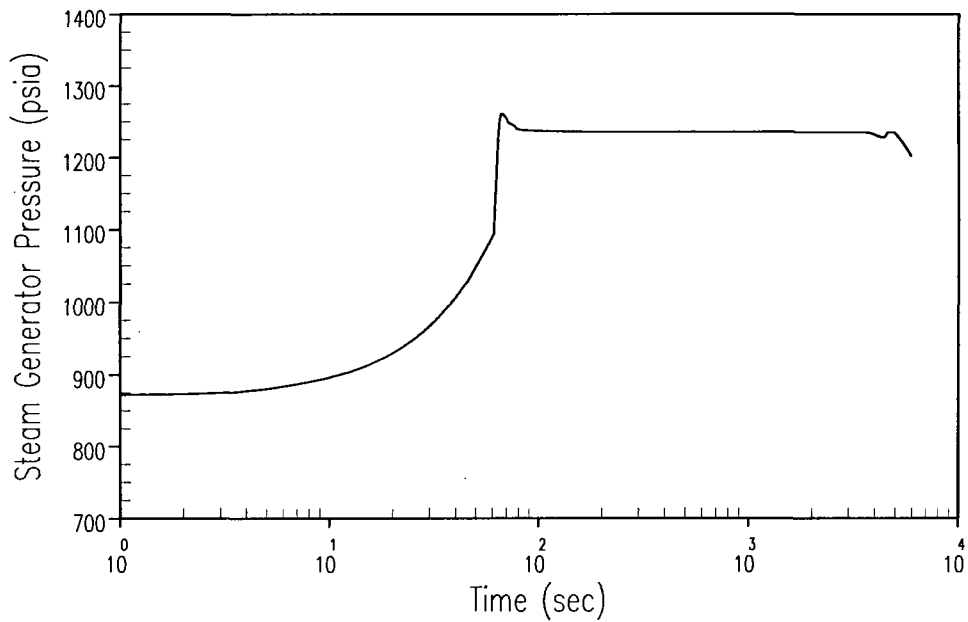


Figure 2.8.5.2.3-12 Unit 1 LONF without Offsite Power – Steam Generator Pressure and Level Versus Time

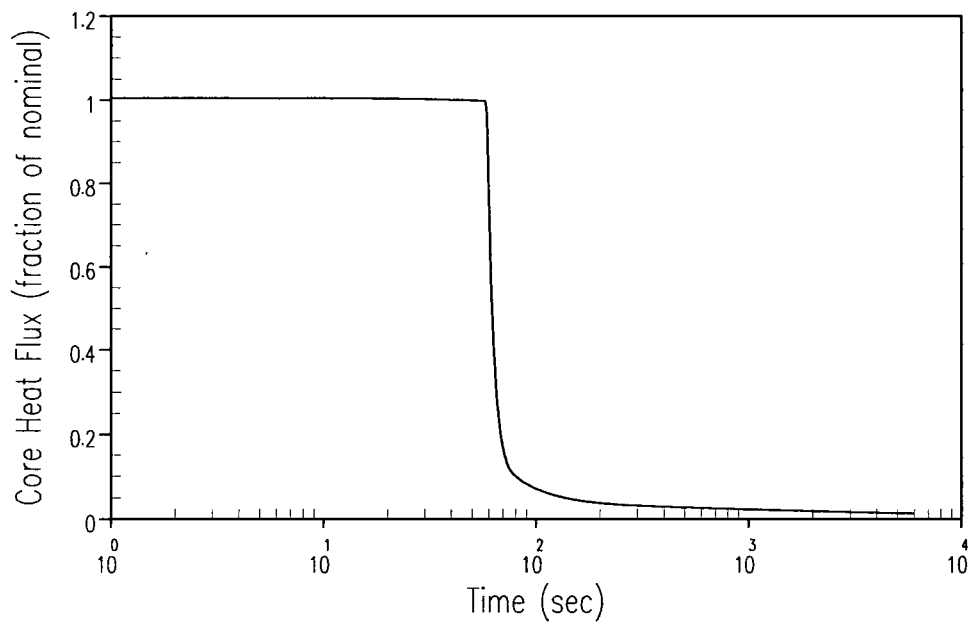
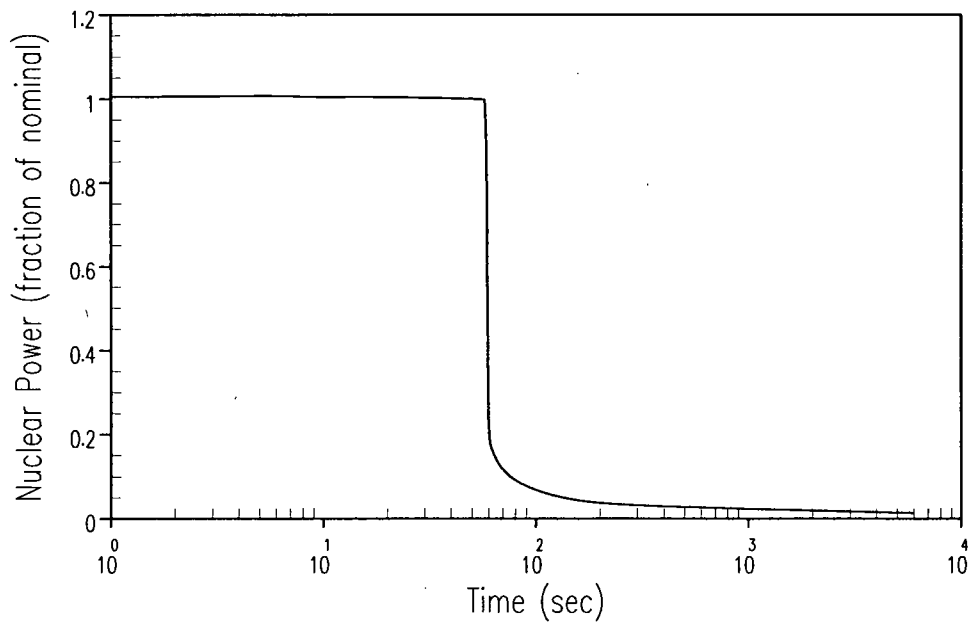
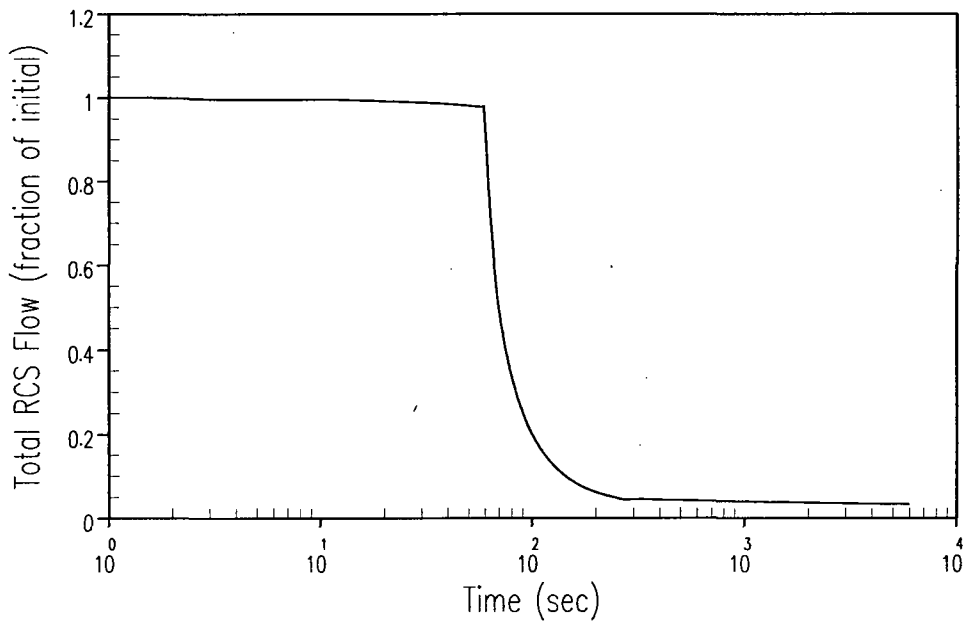


Figure 2.8.5.2.3-13 Unit 2 LONF without Offsite Power – Nuclear Power and Core Average Heat Flux Versus Time



— Hot Leg
 - - - Cold Leg
 - · - · Loop Average

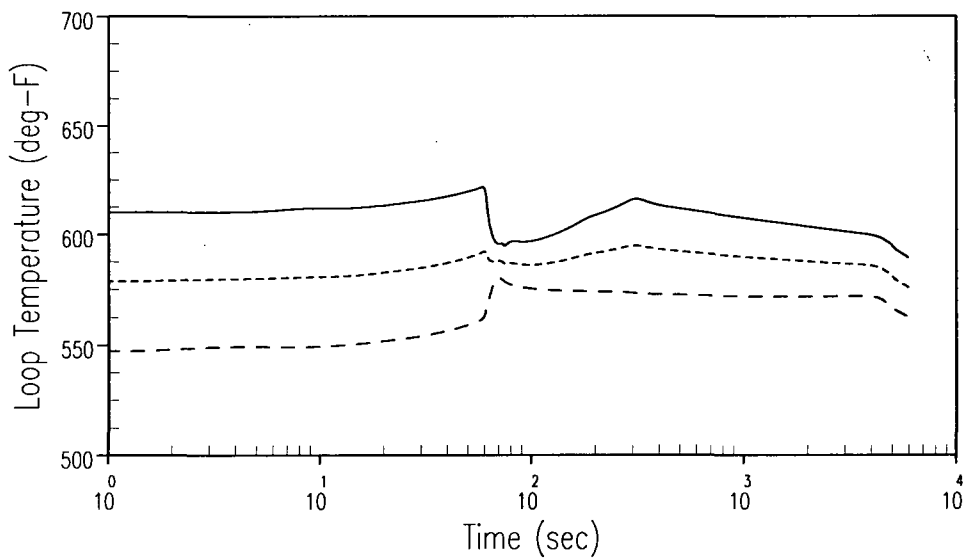


Figure 2.8.5.2.3-14 Unit 2 LONF without Offsite Power – Reactor Coolant Flow Rate and Loop Temperature Versus Time

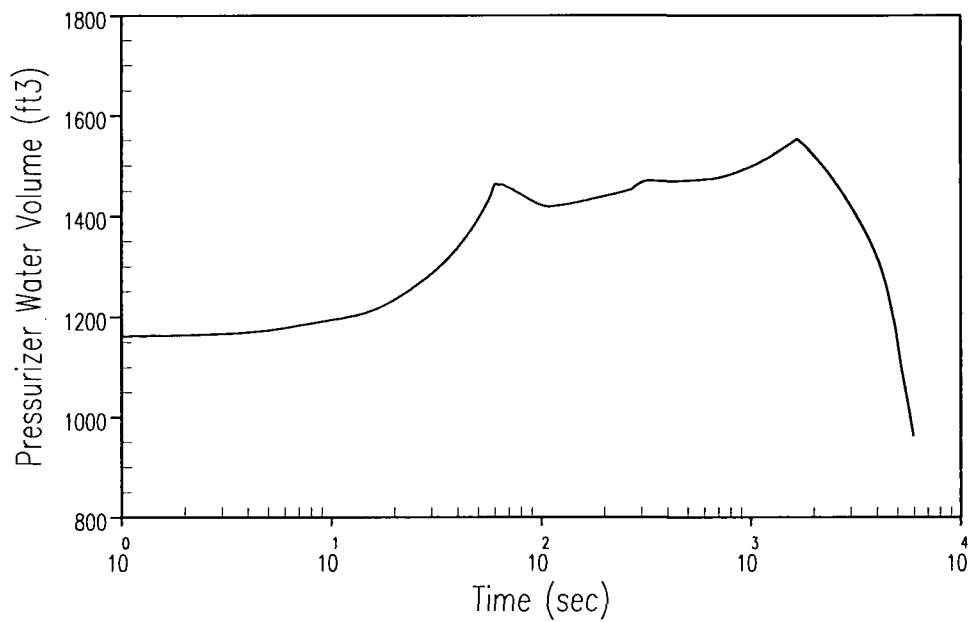
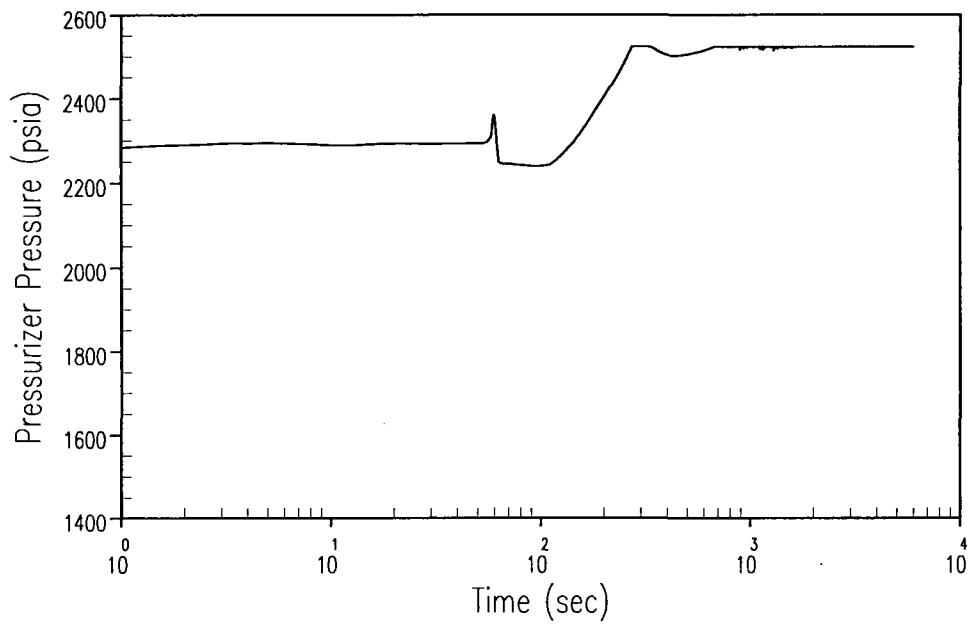


Figure 2.8.5.2.3-15 Unit 2 LONF without Offsite Power – Pressurizer Pressure and Water Volume Versus Time

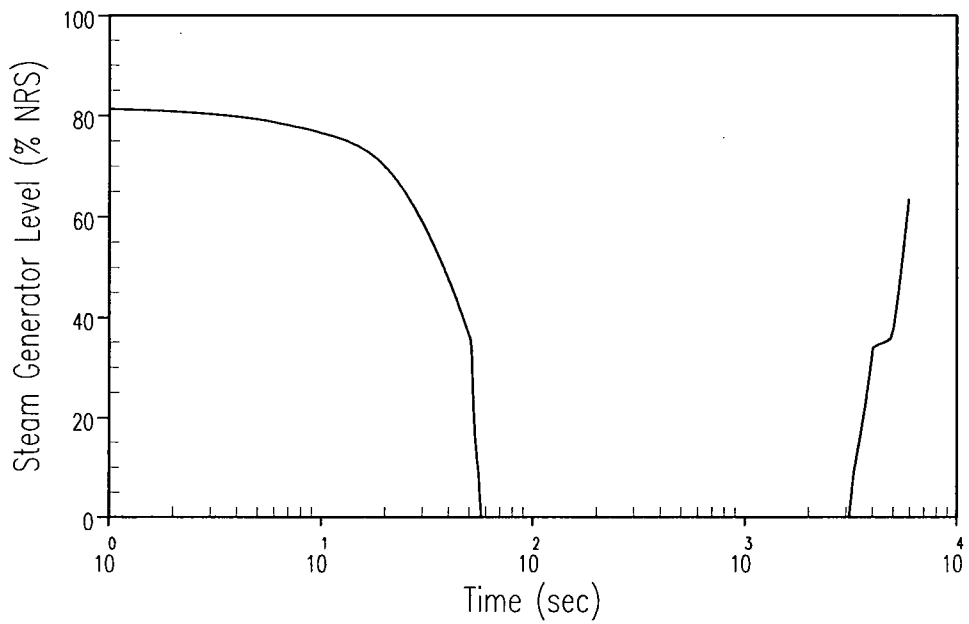
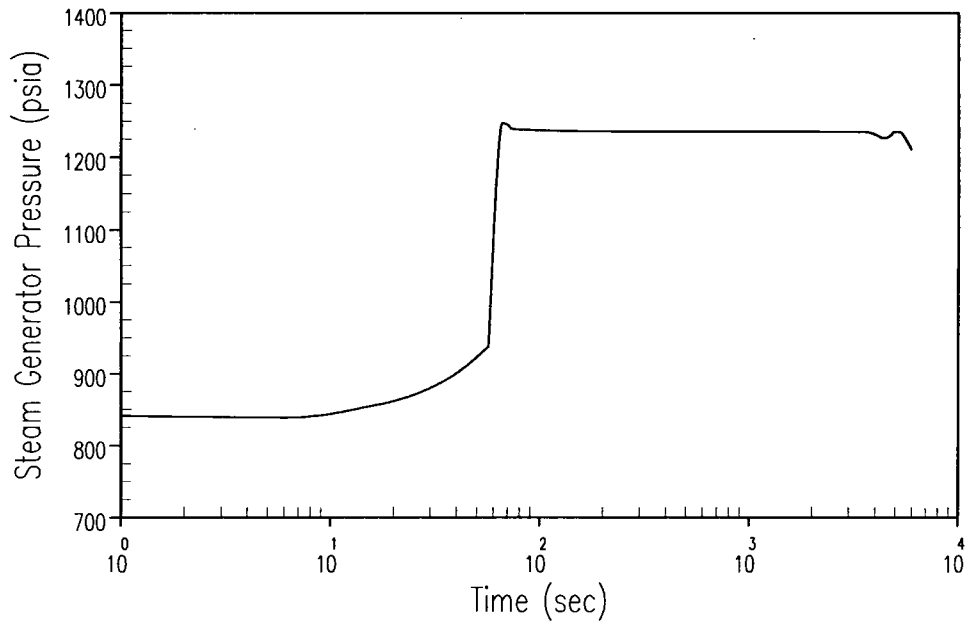


Figure 2.8.5.2.3-16 Unit 2 LONF without Offsite Power – Steam Generator Pressure and Level Versus Time

2.8.5.2.4 Feedwater System Pipe Breaks Inside and Outside Containment

2.8.5.2.4.1 Regulatory Evaluation

Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system (RCS) cooldown (by excessive energy discharge through the break) or an RCS heatup (by reducing feedwater flow to the affected loop). In either case, reactor protection and safety systems are actuated to mitigate the transient.

The review covered:

- The postulated initial core and reactor conditions
- The methods of thermal-hydraulic analyses
- The sequence of events
- The assumed response of the reactor coolant and auxiliary systems
- The functional and operational characteristics of the reactor protection system
- The results of the transient analyses

The Nuclear Regulatory Commission's (NRC's) acceptance criteria are based on:

- General Design Criterion (GDC)-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, thus ensuring that core cooling capability is maintained.
- GDC-28, insofar as it requires that the reactivity control systems be designed to ensure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.
- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to ensure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized.
- GDC-35, insofar as it requires the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP Units 1 and 2 safety related structures, systems, and components with respect to nuclear design relative to conformance to:

- GDC-27, Combined Reactivity Control Systems Capability, is described in FSAR Section 3.1.3.8.

CPNPP Units 1 and 2 are provided with a means of making the core subcritical and maintaining it at that level under any anticipated conditions and with an appropriate margin for contingencies. Chapters 4 and 9 of the FSAR discuss these means in detail. Combined use of the rod cluster control system assemblies (RCCAs) and the chemical shim permit the necessary shutdown margin to be maintained during long term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full-out upon trip for this determination. FSAR Chapter 15 describes accident assumptions in detail.

- GDC-28, Reactivity limits, is described in FSAR Section 3.1.3.9

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values that prevent rupture of the RCS boundary or disruption of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The maximum positive reactivity insertion rates for the withdrawal of RCCAs and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCAs and of the chemical and volume control system (CVCS). Technical Specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of FSAR Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in FSAR Section 4.3. The capability of the CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in FSAR Chapter 15.

Assurance of core cooling capability following Condition IV accidents, such as rod ejections, steam line breaks, and similar accidents, is given by keeping the RCPB stresses within faulted condition limits as specified by applicable American Society of Mechanical Engineers (ASME) codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of necessary safety features.

- GDC-31, Fracture Prevention of Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.4.2.

Close control is maintained over material selection and fabrication for the RCS to ensure that the boundary behaves in a non-brittle manner. The RCS materials exposed to the coolant are corrosion resistant, stainless steel, or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel (RV) structural steel is established by Charpy V-notch and drop weight tests, in accordance with 10 CFR Part 50, Appendix G.

As part of the RV specification, certain requirements that are not specified by the applicable ASME Codes are performed, as follows:

- Ultrasonic Testing – In addition to code requirements, the performance of a 100-percent volumetric ultrasonic test of RV plate for shear wave and a post-hydro-test ultrasonic map of all welds in the pressure vessel are required. Cladding bond ultrasonic inspections to more restrictive requirements than those specified in the codes are also required to preclude interpretation problems during in-service inspection.
- Radiation Surveillance Program – In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens. These programs evaluate the effect of radiation on the fracture toughness of RV steels based on the measured change in reference transition temperature and fracture mechanics measurements. These measurements are performed in accordance with American Society Testing Material (ASTM) E-185-1982, “Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,” and the requirements of 10 CFR Part 50, Appendix H.
- Material chemistry (copper, phosphorous, sulfur, and vanadium) of the reactor vessel core region is controlled to reduce sensitivity to embrittlement, which is caused by irradiation over the life of the plant.

The fabrication and quality control techniques used in the fabrication of the RCS are equivalent to those used for the RV. The inspections of the RV, pressurizer, piping, pumps, and steam generators are governed by requirements of the ASME Codes (Refer to FSAR Chapter 5).

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Appendix G, “Protection Against Non-Ductile Failure.” The approach specifies that allowed stress intensity factors for all vessel operating conditions shall not exceed the referenced stress intensity factor (KIR) for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material reference temperature (RT_{NDT}) due to irradiation.

- GDC-35, Emergency Core Cooling System (ECCS), is addressed in FSAR Section 3.1.4.6.

An ECCS is provided to cope with any LOCA in the plant design basis. Abundant cooling water is available in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to ensure that cladding metal-water reaction is limited to less than 1 percent. Adequate design provisions are made to ensure performance of the required safety functions even with a single failure.

FSAR Section 6.3 includes details of the capability of the systems. FSAR Chapter 15 includes an evaluation of the adequacy of the system functions. Performance evaluations are conducted in accordance with 10 CFR Part 50.46 and Appendix K.

Chapter 15 of the FSAR and its subsections discuss postulated events that include decreases in RCS inventory, as well as decreases in secondary-side heat removal capability. Section 10 of the FSAR describes the main steam system.

2.8.5.2.4.2 Technical Evaluation

The specific acceptance criterion applied for this event is that no boiling occur in the hot or cold legs prior to the point in the transient where the heat removal capacity of the auxiliary feedwater (AFW) system exceeds the heat generation. This conservatively ensures that the core remains covered and geometrically intact for the duration of the event. Furthermore, the analysis ensures that appropriate margin for malfunctions, such as stuck rods, were accounted for in the safety analysis assumptions. This conservatively satisfies the CPNPP Units 1 and 2 current licensing basis with respect to the requirements of GDC-27, GDC-28, GDC-31, and GDC-35.

The discussion below demonstrates that all applicable acceptance criteria are met for this event at CPNPP Units 1 and 2 at uprated power conditions.

2.8.5.2.4.2.1 Introduction

A major feedwater line break (FSAR Section 15.2.8) is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator can also be discharged through the break. Furthermore, a break in this location could preclude the subsequent addition of AFW to the affected steam generator. A break upstream of the feedline check valve would affect the nuclear steam supply system (NSSS) only as a loss of feedwater. This case is covered by the loss-of-normal-feedwater (LONF) analysis presented in Licensing Report (LR) subsection 2.8.5.2.3.

Depending upon the size of the break and the plant operating conditions at the time of the rupture, the break could either cause an RCS heatup or cooldown. The potential RCS cooldown resulting from a secondary pipe break is evaluated in the steam line break analysis presented in LR subsection 2.8.5.1.2.2.1. Only the RCS heatup effects of a feedline break are presented in this section.

A feedline break reduces the ability to remove heat generated by the core from the RCS. The AFW system is provided to ensure that adequate feedwater is available to provide decay heat removal.

2.8.5.2.4.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The following key assumptions were made in the analysis:

- NSSS power of 3,628 MWt plus 0.6-percent power uncertainty was assumed.
- The initial RCS average temperature was set to 595.2°F, the nominal high T_{avg} value of 589.2°F plus a T_{avg} uncertainty of 6.0°F.
- The initial pressurizer pressure was 30 psid below its nominal value of 2,250 psia to account for initial condition uncertainties.
- The initial pressurizer level was set to the nominal full-power programmed value of 60-percent span plus 5.0-percent span to account for initial condition uncertainties.
- The initial steam generator water level for Unit 1 was set to the nominal value (67-percent narrow-range span (NRS)) plus 10-percent NRS in the faulted steam generator, and the nominal value minus 10-percent NRS in the intact steam generators to account for initial condition uncertainties.
- The initial steam generator water level for Unit 2 was set to the nominal value (64-percent NRS) plus 18-percent NRS in the faulted steam generator, and the nominal value minus 7-percent NRS in the intact steam generators to account for initial condition uncertainties.
- The main feedwater flow to all steam generators was assumed to be lost at the time the break occurred (all main feedwater spilled out through the break).
- The full double-ended main feedwater pipe break was assumed. An effective break size of 1.11844 ft² was analyzed for CPNPP Unit 1 with $\Delta 76$ SGs and 0.2234 ft² for Unit 2 with D-5 SGs.
- The single-failure assumption was conservatively set as the loss of the motor driven AFW pump feeding two intact steam generators; all the flow from the second motor driven AFW pump is assumed to flow out the break along with a portion of the turbine driven AFW flow until the faulted steam generator is isolated.
- For the first 30 minutes following reactor trip, a total of 430 gpm of AFW flow from the turbine-driven AFW pump was split equally among the three intact steam generators. Following isolation of the faulted steam generator, an additional 370 gpm was made available to split among the 3 intact steam generators. No AFW flow is assumed to reach the faulted steam generator. AFW flow from the second motor-driven pump was conservatively not modeled.
- Although it is expected that the actuation of the safety injection system would occur during this event, the analysis conservatively did not model the safety injection flow.

-
- Pressurizer power-operated relief valves (PORVs) were assumed operable as they minimize RCS pressure, which results in a lower saturation temperature. The pressurizer sprays and heaters were assumed to be unavailable.
 - Reactor trip was assumed to be actuated when the steam generator low-low water level trip setpoint was reached in the ruptured steam generator. A setpoint of 10-percent NRS was modeled for Unit 1 and 7.5-percent NRS was modeled for Unit 2. A description of the method used by RETRAN to calculate steam generator level is provided in Section 3.8.2 of WCAP-14882 (Reference 1).
 - The main steam line isolation valves serve to isolate the intact steam generators from the faulted steam generator.
 - No credit was taken for heat energy deposited in portions of the RCS metal during the RCS heatup.
 - No credit was taken for charging or letdown.
 - Maximum steam generator tube plugging of 10 percent was assumed to minimize primary-to-secondary-side heat transfer.
 - Steam generator heat transfer across the tubes was adjusted as the shell-side liquid inventory decreased. Specifically, the heat transfer correlation for the steam generator tubes (heat conductors) is automatically adjusted by the RETRAN code for the changing conditions as the tubes uncover.
 - Core residual heat generation was based on the 1979 version of American Nuclear Society (ANS) 5.1 (Reference 2). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip was assumed.
 - No credit was taken for the following potential protection logic signals to mitigate the consequences of the accident:
 - High-pressurizer pressure
 - High-pressurizer level
 - High-containment pressure
 - Overtemperature N-16

The feedline break accident is a Condition IV occurrence as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. Condition IV events are faults that are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material. The following items summarize the acceptance criteria associated with this event:

-
- Pressures in the RCS and main steam system (MSS) are maintained below 110 percent of the respective design pressures.
 - Any activity release is such that the calculated doses are within acceptable limits.
 - Any fuel damage that can occur during the transient is of a sufficiently limited extent that the core will remain in place and geometrically intact with no loss of core cooling capability.

With respect to overpressurization, the feedline break event, both with and without offsite power, is bounded by the loss-of-load/turbine trip event discussed in LR subsection 2.8.5.2.1 in which assumptions are made to conservatively calculate the RCS and MSS pressure transients. For the feedline break event, turbine trip occurs after reactor trip, whereas for the loss-of-load/turbine trip event, the turbine trip is the initiating fault. Therefore, the primary/secondary power mismatch and resultant RCS and MSS heatup and pressurization transients are always more severe for the loss-of-load/turbine trip than for the feedline break event. For this reason, no attempt is made to calculate the maximum RCS or MSS pressures for the feedline break event.

Although no explicit dose calculations are performed for the feedline break event, by demonstrating there is no fuel cladding damage as a result of this event, the radiological consequences can be shown to be bounded by those of the steam line rupture event, discussed in LR subsection 2.9.2.

With respect to fuel cladding damage due to "dryout," where the water level in the vessel drops below the top of the core, Westinghouse has established an internal criterion that no bulk boiling occurs in the primary coolant system prior to event turnaround. Turnaround occurs when the heat removal capability of the steam generators being fed auxiliary feedwater exceeds NSSS heat generation. This conservatively ensures that the core remains covered with water and thereby will remain in place and geometrically intact with no loss of core cooling capability. This single criterion is conservative and was chosen for convenience in interpreting the transient results.

With respect to fuel cladding damage due to departure from nucleate boiling, the pre-trip aspects of a feedline break event would be bounded by the loss-of-load/turbine trip heatup event, discussed in LR subsection 2.8.5.2.1, while the post-trip aspects of a feedline break event would be bounded by the steam line rupture event, discussed in LR subsection 2.8.5.1.2.2.1.

2.8.5.2.4.2.3 Description of Analyses and Evaluations

The transient response following a feedline break event was calculated by a detailed digital simulation of the plant. The analysis modeled a simultaneous loss of main feedwater to all steam generators and subsequent reverse blowdown of the faulted steam generator. The analysis was performed using the RETRAN code (Reference 1), which simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam

generators, and steam generator safety valves. The code computed pertinent plant variables including temperatures, pressures, and power level. Also, RETRAN calculates the transient break flow quality as a function of temperature and pressure with respect to time.

The following two cases were analyzed for CPNPP Units 1 and 2:

- Unit 1: Maximum reactivity feedback, with offsite power, 1.11844 ft² break
Maximum reactivity feedback, without offsite power, 1.11844 ft² break
- Unit 2: Minimum reactivity feedback, with offsite power, 0.2234 ft² break
Minimum reactivity feedback, without offsite power, 0.2234 ft² break

2.8.5.2.4.2.4 Results

The results of the feedline break cases analyzed showed that no bulk boiling occurred in the primary coolant system following a feedline break prior to the time that the heat removal capability of the steam generators, being fed AFW, exceeded NSSS residual heat generation.

The limiting case for each unit was the case where offsite power was assumed to be available. The transient results for the limiting cases are presented in Figures 2.8.5.2.4-1 through 2.8.5.2.4-6 for Unit 1, and Figures 2.8.5.2.4-7 through 2.8.5.2.4-12 for Unit 2. The time sequence of events for these cases are presented in Table 2.8.5.2.4-1 and 2.8.5.2.4-2 for Units 1 and 2, respectively.

The results of the analyses performed for CPNPP Units 1 and 2 at uprated power conditions showed that for the postulated feedwater line rupture, AFW system capacity was adequate to remove decay heat, thus, ensuring that the applicable acceptance criteria are met.

2.8.5.2.4.3 Conclusion

The analyses of feedwater system pipe breaks have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a non-brittle manner, the probability of propagating fracture of the RCPB will be minimized, and abundant core cooling will be provided. Based on this, it is concluded that the plant will continue to meet the CPNPP current licensing basis with respect to the requirements of GDC-27, GDC-28, GDC-31, and GDC-35 following implementation of the proposed SPU. Therefore, the proposed SPU is acceptable with respect to the RCS heatup effects of a feedwater system pipe break.

2.8.5.2.4.4 References

1. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
2. ANSI/ANS-5.1 – 1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

Table 2.8.5.2.4-1

Unit 1 - Time Sequence of Events – Major Rupture of a Main Feedwater Pipe

Case	Event	Time (sec)⁽¹⁾
Feedline Rupture with Maximum Reactivity Feedback, Offsite Power Available, Break Size of 1.11844 ft ²	Main feedline rupture occurs	20.0
	Low-low steam generator water level reactor trip setpoint reached in ruptured steam generator	26.0
	Low-low steam generator water level trip signal is generated in ruptured steam generator and rods begin to fall	28.0
	Low-low steam generator water level reactor trip setpoint reached in second steam generator	37.2
	Flow from the turbine-driven AFW pump is initiated	122.2
	Low steamline pressure setpoint reached in ruptured steam generator	156.0
	All main steamline isolation valves close	161.0
	Pressurizer PORV setpoint reached (first occurrence)	338.6
	First steam generator safety valve setpoint reached in intact steam generators	662.1
	Minimum margin to hot leg saturation occurs (Hot and cold leg temperatures begin to decrease)	1,837.5

Note:

1. Time includes 20 seconds of steady-state.

Table 2.8.5.2.4-2

Unit 2 - Time Sequence of Events – Major Rupture of a Main Feedwater Pipe

Case	Event	Time (sec)⁽¹⁾
Feedline Rupture with Minimum Reactivity Feedback, Offsite Power Available, Break Size of 0.2234 ft ²	Main feedline rupture occurs	20.0
	Pressurizer PORV setpoint reached (first occurrence)	31.5
	Low-low steam generator water level reactor trip setpoint reached in ruptured steam generator	41.5
	Low-low steam generator water level trip signal is generated in ruptured steam generator and rods begin to fall	43.5
	Low-low steam generator water level reactor trip setpoint reached in second steam generator	47.2
	First steam generator safety valve setpoint reached in intact steam generators	48.9
	Flow from the turbine-driven AFW pump is initiated	132.2
	Low steamline pressure setpoint reached in ruptured steam generator	418.5
	All main steamline isolation valves close	423.5
	Minimum margin to hot leg saturation occurs (Hot and cold leg temperatures begin to decrease)	1,848.5

Note:

1. Time includes 20 seconds of steady-state.

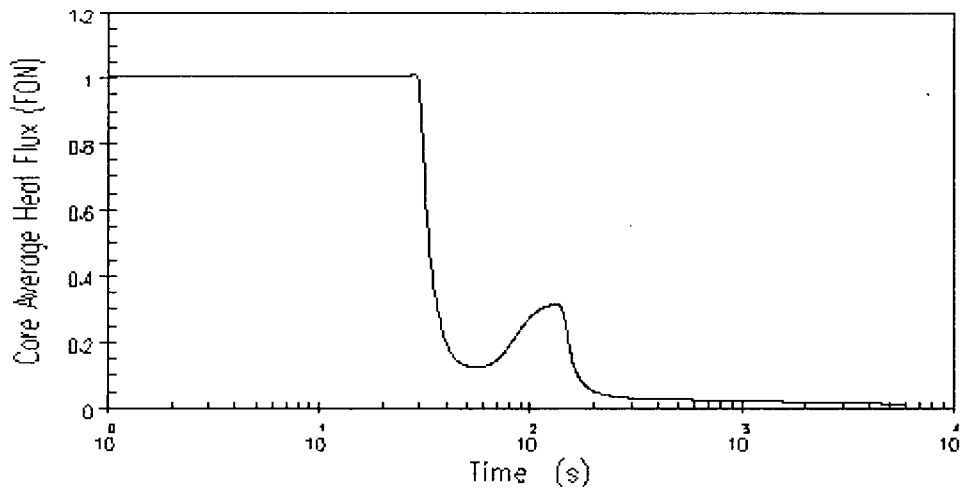
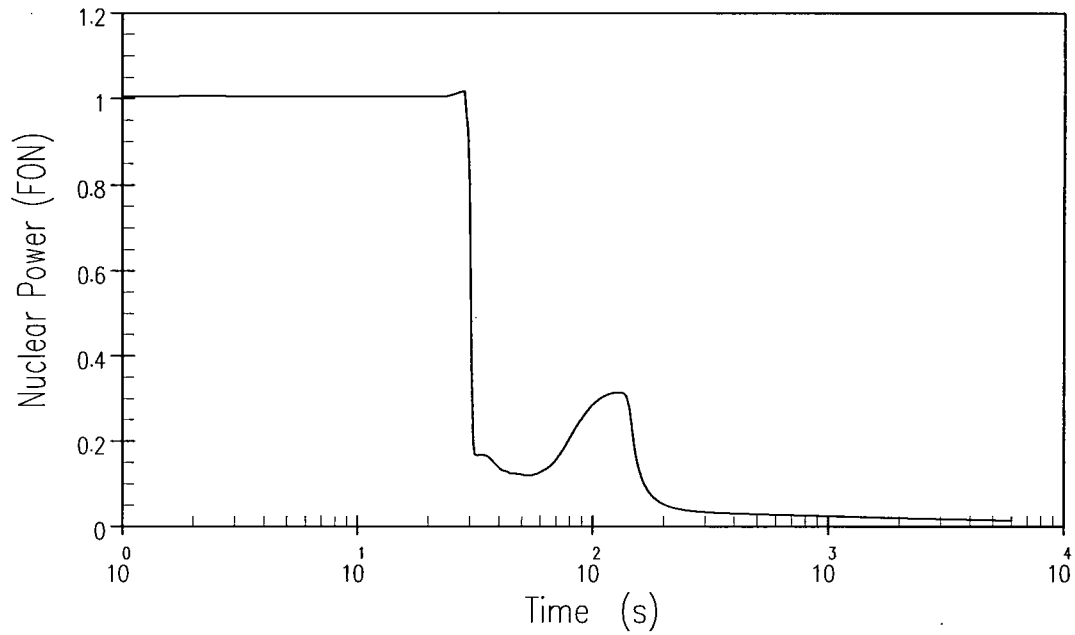


Figure 2.8.5.2.4-1 CPNPP Unit 1 – Feedline Break with Offsite Power – Nuclear Power and Core Average Heat Flux Versus Time

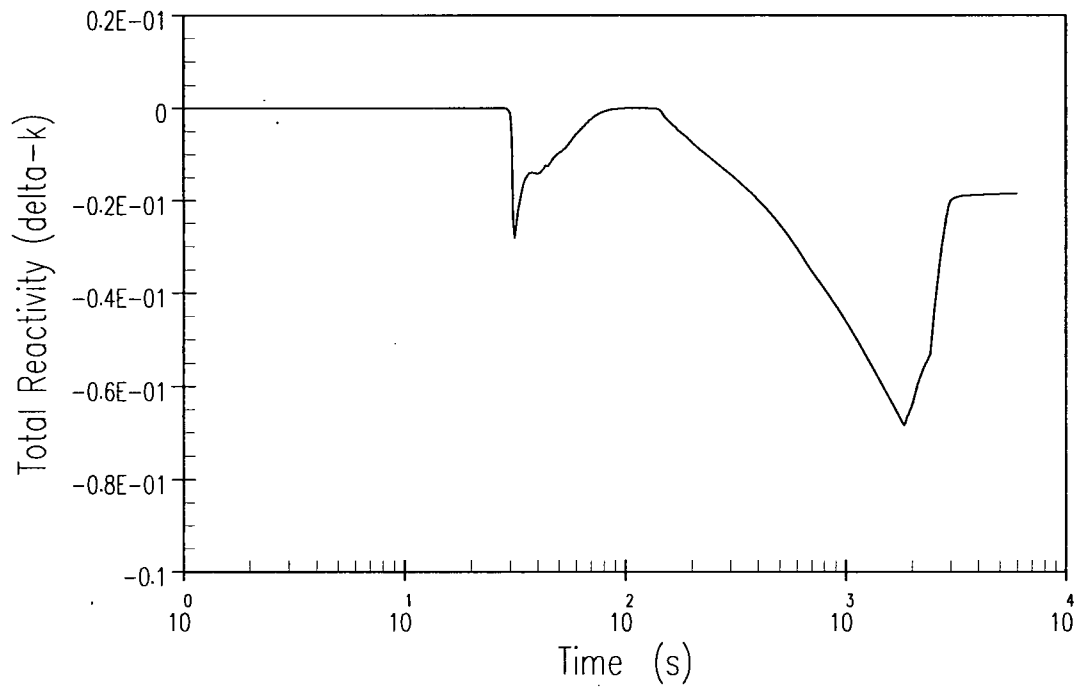


Figure 2.8.5.2.4-2 CPNPP Unit 1 – Feedline Break with Offsite Power – Total Reactivity Versus Time

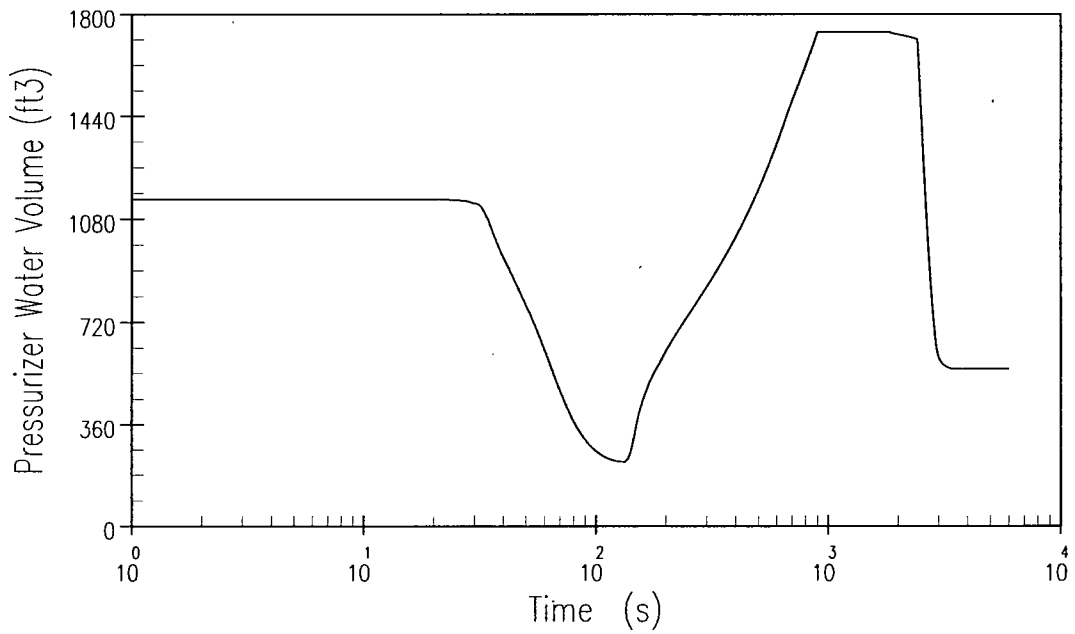
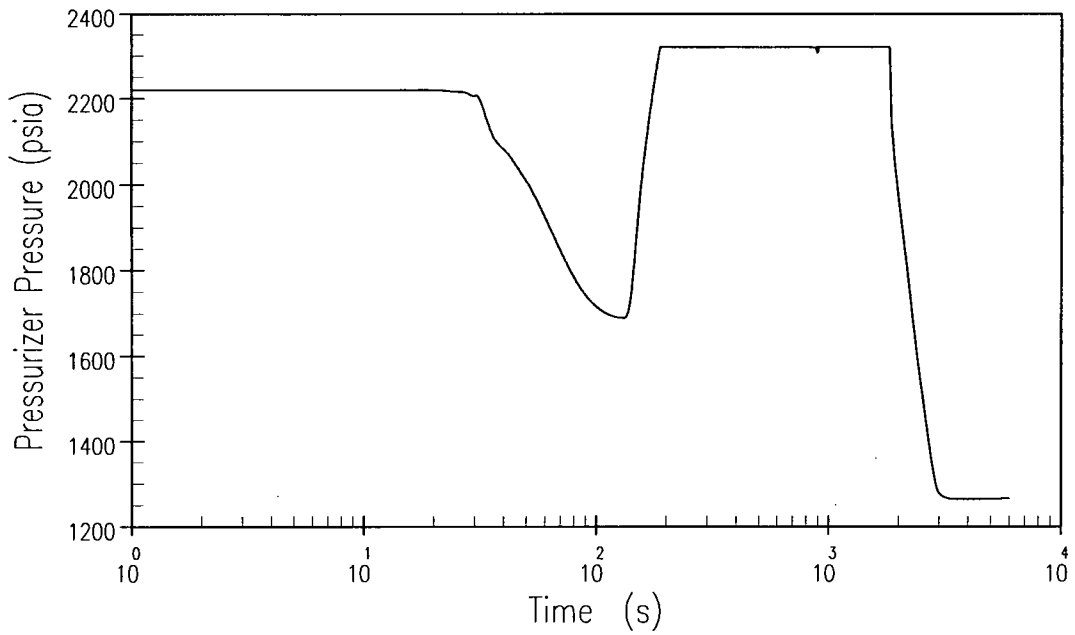


Figure 2.8.5.2.4-3 CPNPP Unit 1 – Feedline Break with Offsite Power – Pressurizer Pressure and Water Volume Versus Time

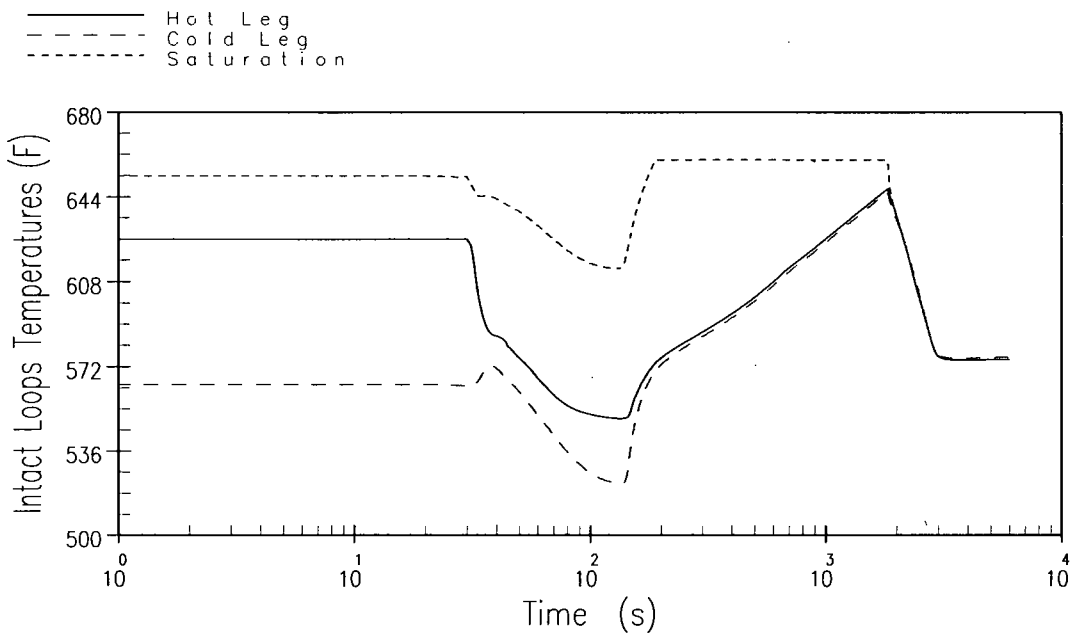
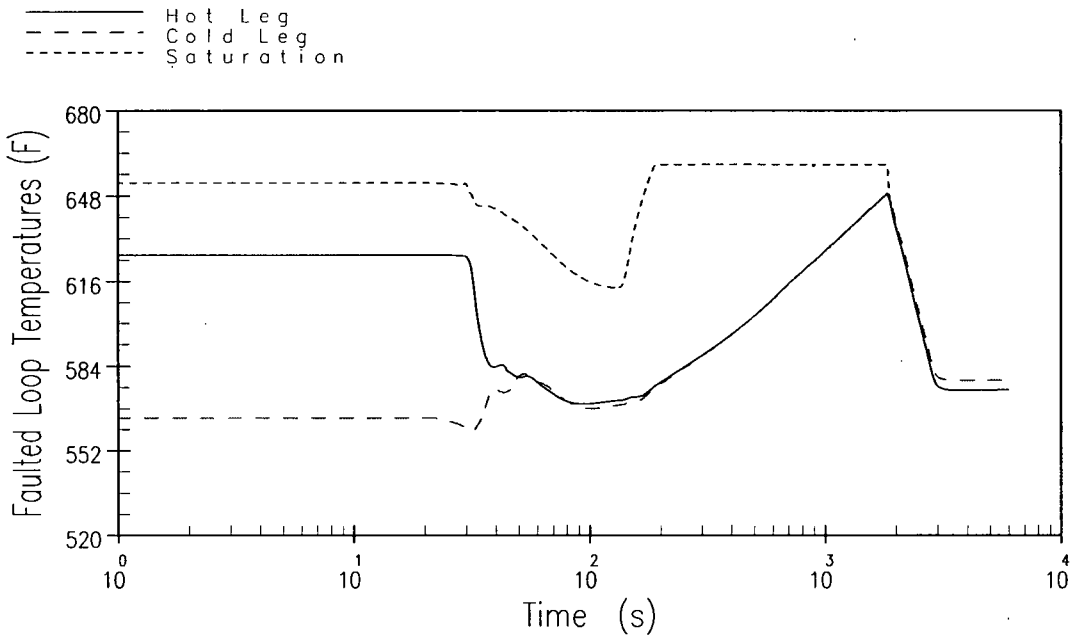


Figure 2.8.5.2.4-4 CPNPP Unit 1 – Feedline Break with Offsite Power – Reactor Coolant Temperatures Versus Time for the Faulted and Intact Loops

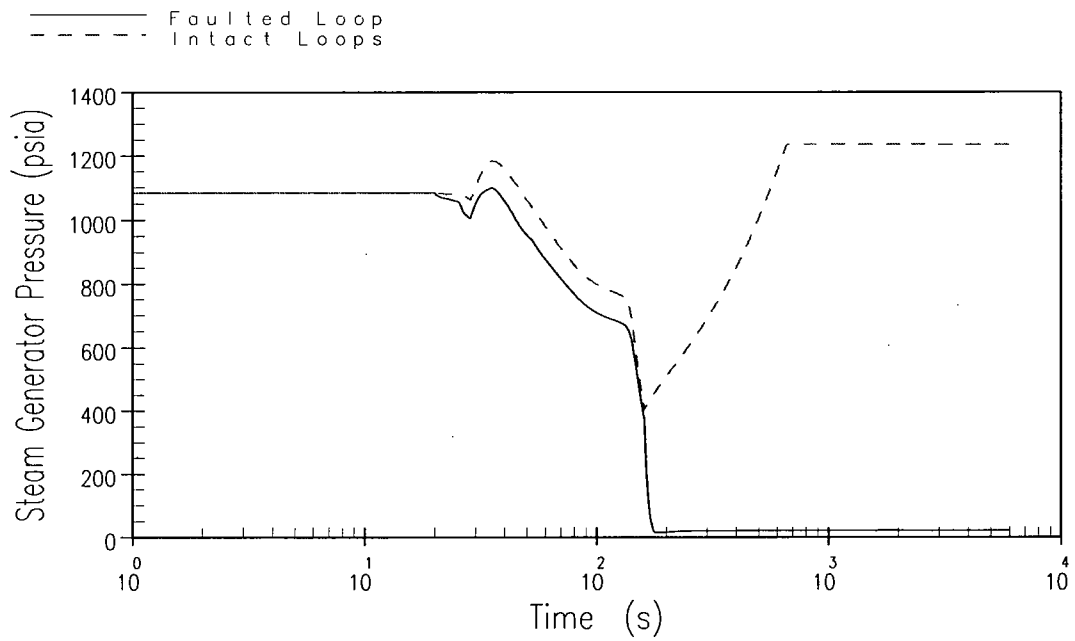
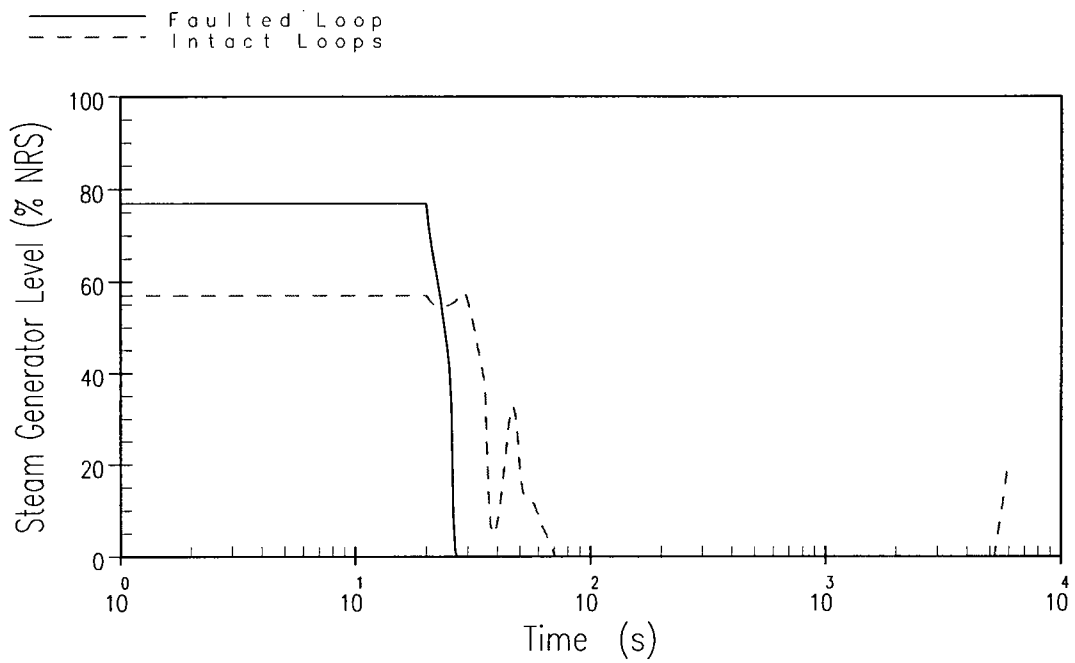


Figure 2.8.5.2.4-5 CPNPP Unit 1 – Feedline Break with Offsite Power – Steam Generator Level and Pressure Versus Time

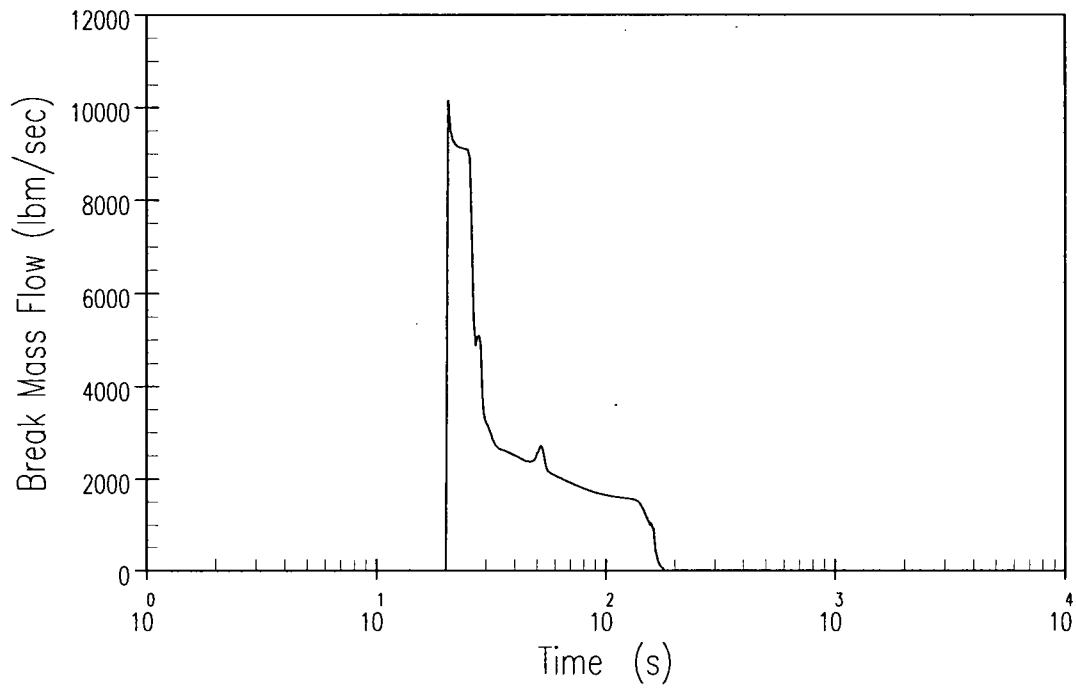


Figure 2.8.5.2.4-6 CPNPP Unit 1 – Feedline Break with Offsite Power – Feedline Break Flow Versus Time

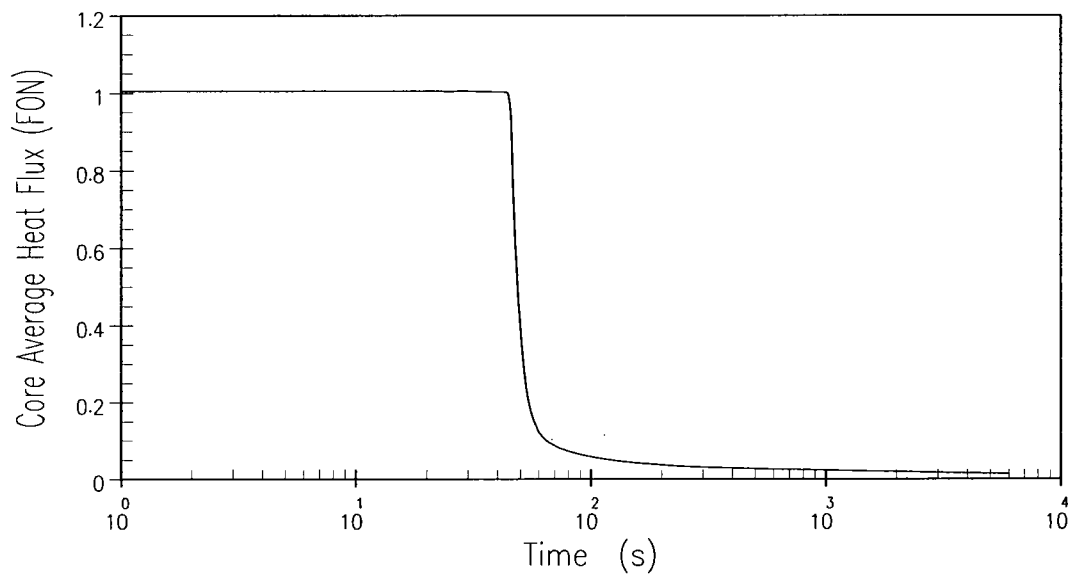
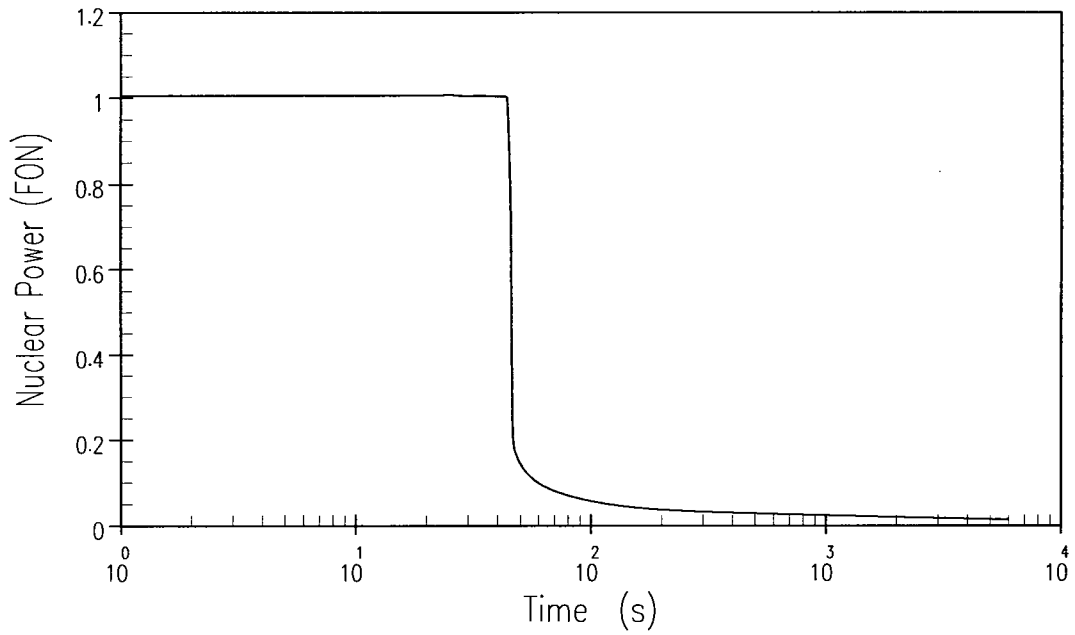


Figure 2.8.5.2.4-7 CPNPP Unit 2 – Feedline Break with Offsite Power – Nuclear Power and Core Average Heat Flux Versus Time

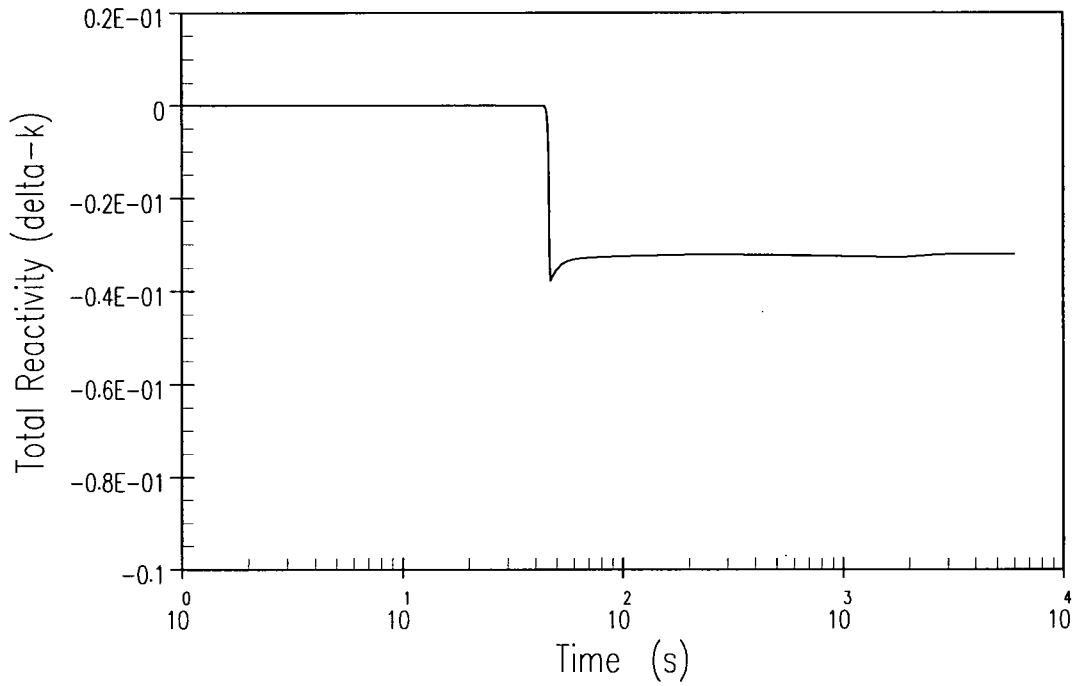


Figure 2.8.5.2.4-8 CPNPP Unit 2 – Feedline Break with Offsite Power – Total Reactivity Versus Time

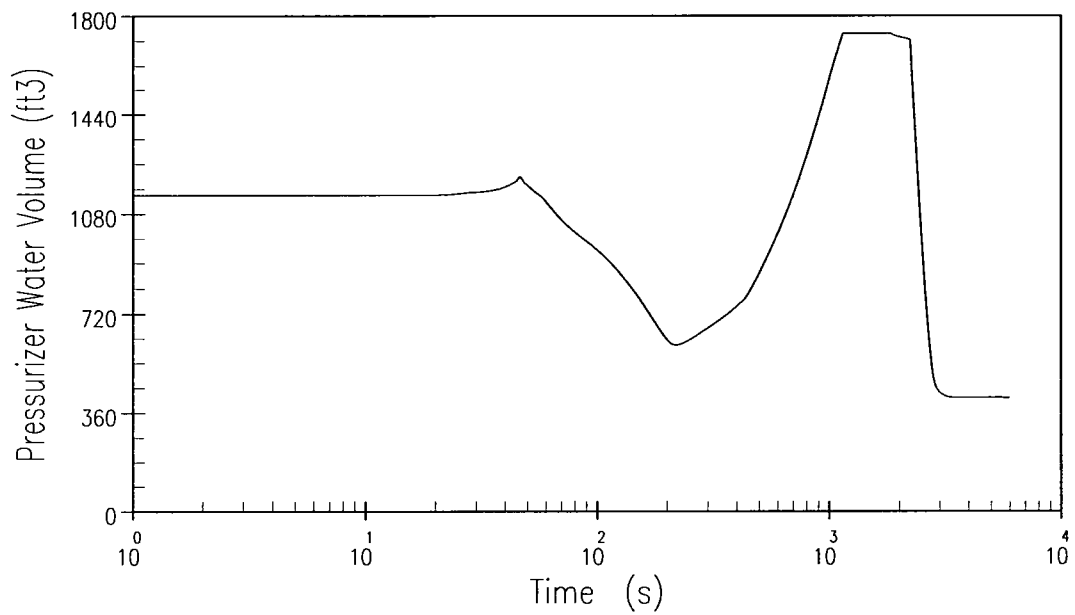
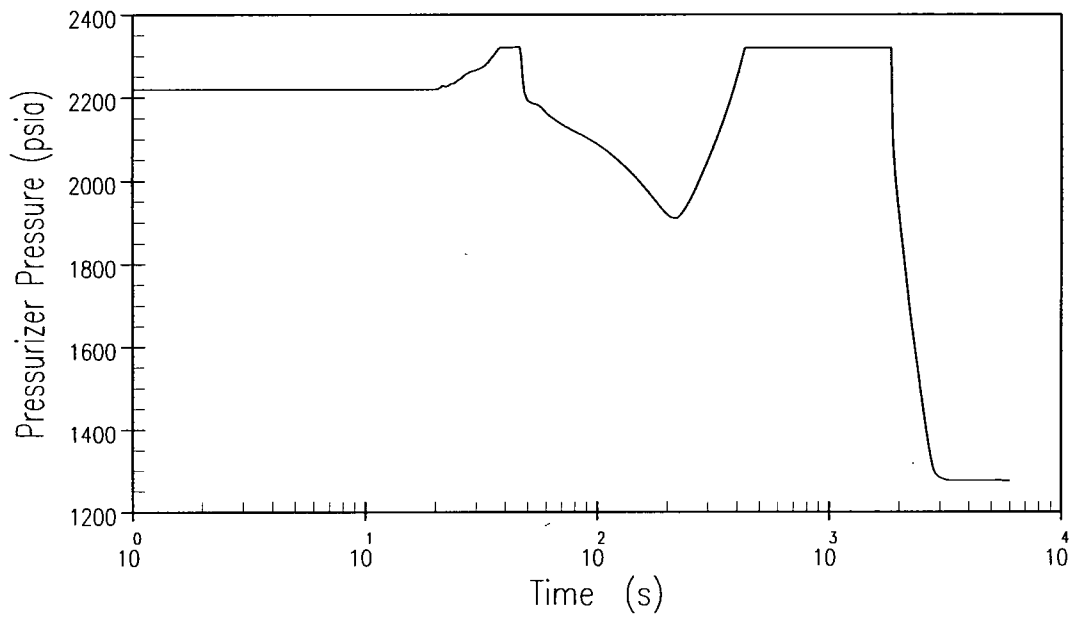


Figure 2.8.5.2.4-9 CPNPP Unit 2 – Feedline Break with Offsite Power – Pressurizer Pressure and Water Volume Versus Time

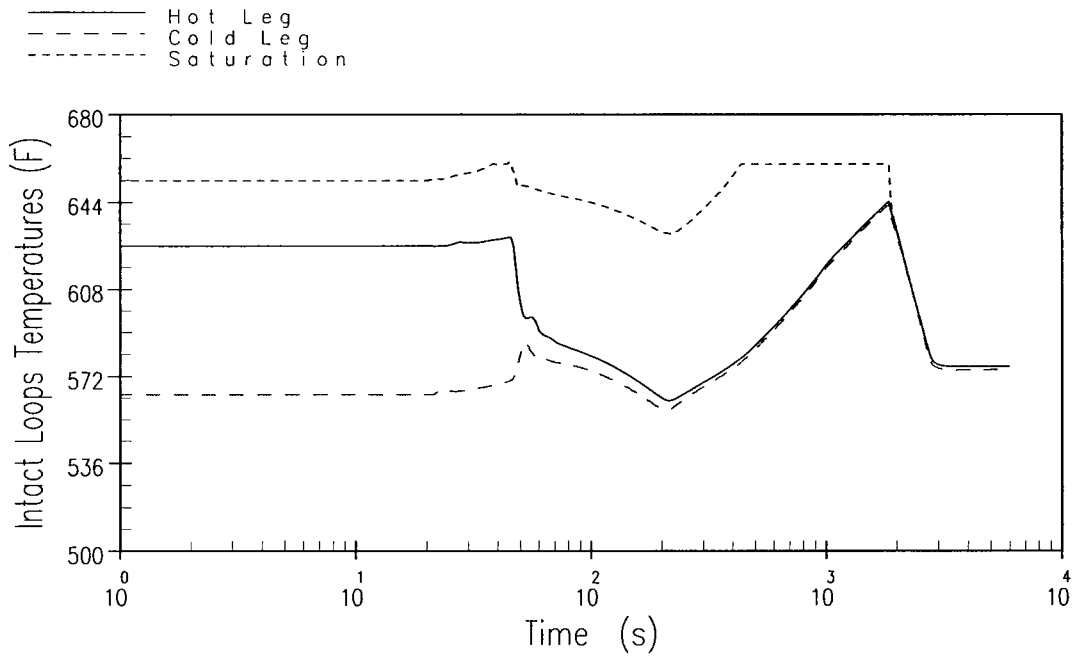
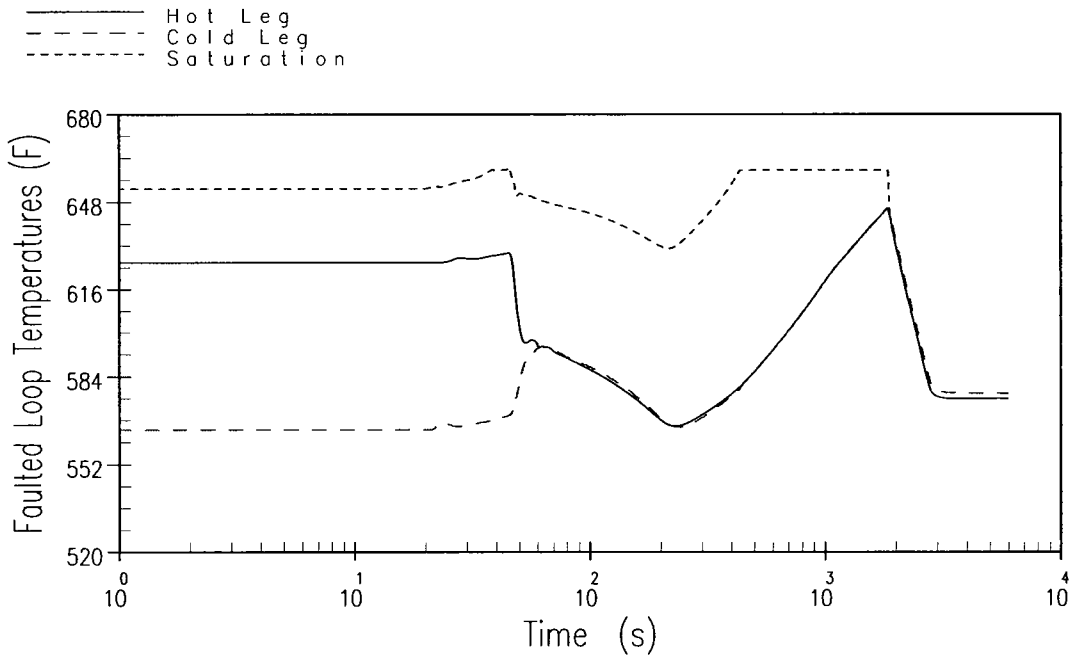


Figure 2.8.5.2.4-10 CPNPP Unit 2 – Feedline Break with Offsite Power – Reactor Coolant Temperatures Versus Time for the Faulted and Intact Loops

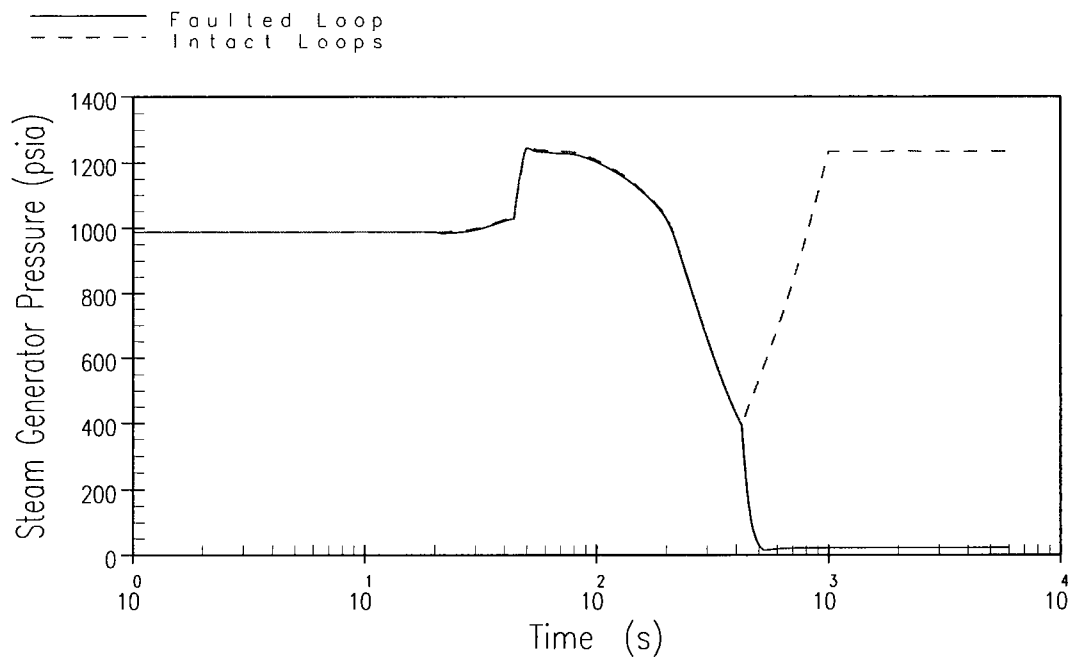
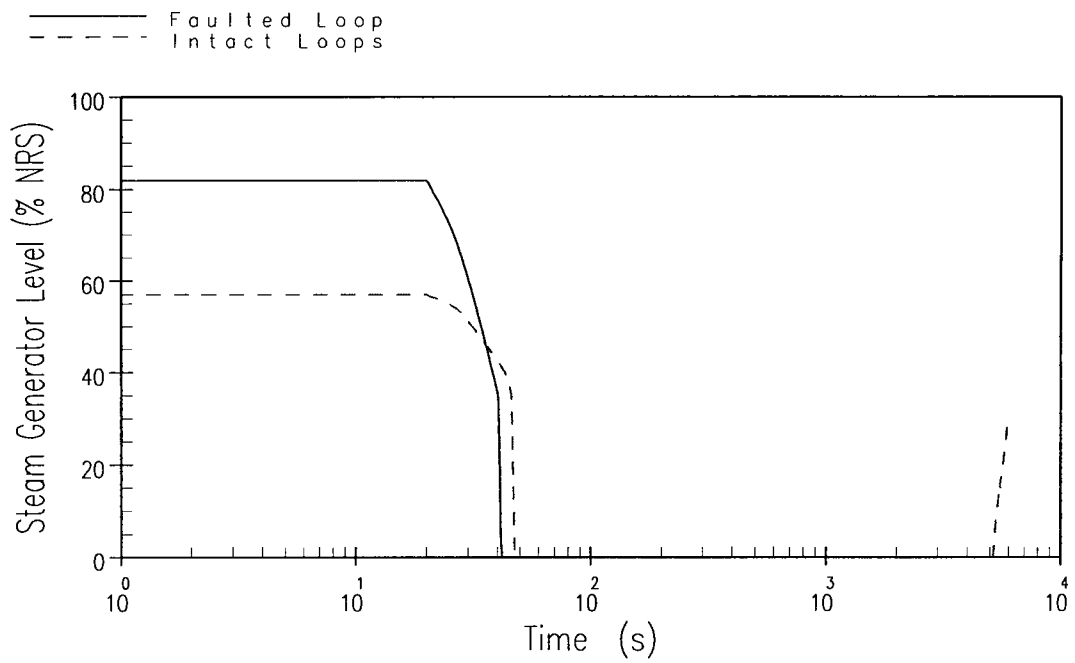


Figure 2.8.5.2.4-11 CPNPP Unit 2 – Feedline Break with Offsite Power – Steam Generator Level and Pressure Versus Time

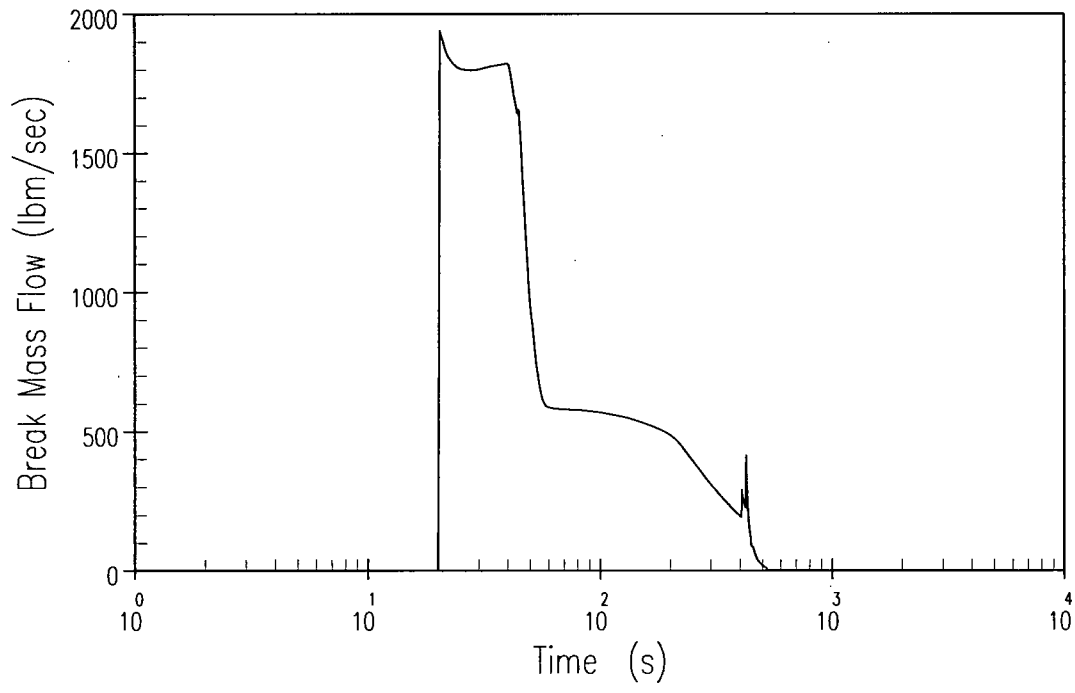


Figure 2.8.5.2.4-12 CPNPP Unit 2 – Feedline Break with Offsite Power – Feedline Break Flow Versus Time

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

2.8.5.3.1.1 Regulatory Evaluation

A decrease in reactor coolant system (RCS) flow occurring while the plant is at power will result in a degradation of core heat transfer. This will lead to increased fuel temperatures that may cause fuel damage if specified acceptable fuel design limits (SAFDLs) are exceeded. Reactor trip and safety systems will actuate to mitigate the transient. The following transient characteristics have been reviewed:

- The postulated initial core and reactor conditions
- The methods of thermal and hydraulic analyses
- The sequence of events
- The assumed reactions of reactor system components
- The functional and operational characteristics of the reactor trip system (RTS)
- The results of the transient analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including anticipated operational occurrences (AOOs).
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCS pressure boundary are not exceeded during any condition of normal operation.
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

- To preclude significant fuel damage during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II).
- To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.
- To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and design evaluation of reactor components, including the fuel, reactor vessel internals, and reactivity control systems. Details of the control and protection systems instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the accident analyses of FSAR Chapter 15, which show that the acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, reactor coolant pressure boundary (RCPB) components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and of integrated hydrostatic testing of assembled components.

FSAR Chapter 5 discusses the RCS design.

- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control methods are provided. These are rod control cluster assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the full-length control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and AOOs, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCA are presented in FSAR Chapter 4. The operation is discussed in FSAR Chapter 7. The means of controlling the boric acid concentration are described in FSAR Chapter 9. Performance analyses under accident conditions are included in FSAR Chapter 15.

The current licensing basis confirms that all applicable acceptance criteria are met for the loss of forced reactor coolant flow transients. Specifically, as noted in the FSAR Sections 15.3.1.3 and 15.3.2.3, the departure from nucleate boiling ratio (DNBR) will not decrease below the limit value at any time during the transients. Therefore, no fuel or cladding damage is predicted. Also, the peak RCS and main steam system (MSS) pressures remained below their respective limits at all times.

2.8.5.3.1.2 Technical Evaluation

The specific acceptance criteria for this event are as follows:

- The DNBR remains above the 95/95 DNBR limit at all times during the transient. Demonstrating that the DNBR limit is met satisfies the requirements of GDC-10.
- Primary and secondary pressures remain below 110 percent of their respective design pressures at all times during the transient. Demonstrating that the primary and secondary pressure limits are met satisfies the requirements of GDC-15.
- GDC-26 requires reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including AOOs. This is accomplished by ensuring that appropriate margin for malfunctions, such as stuck RCCAs, are accounted for in the safety analysis assumptions. Demonstrating that the fuel design limits (that is, DNBR) are met satisfies the requirements of GDC-26.

The discussion below demonstrates that all applicable acceptance criteria are met for this event at CPNPP Units 1 and 2 at uprated conditions.

2.8.5.3.1.2.1 Introduction

A loss of forced reactor coolant flow accident (FSAR Sections 15.3.1 and 15.3.2) can result from a mechanical or electrical failure in a reactor coolant pump (RCP), from an interruption in the power supplying one or more of these pumps, or from a reduction in RCP motor supply frequency. If the reactor is at power at the time of the event, the immediate effect from the loss of forced coolant flow is a rapid increase in the coolant temperature. This increase in coolant temperature could result in departure from nucleate boiling (DNB), with subsequent fuel damage, if the reactor is not promptly tripped.

The following signals provide protection against a loss of forced reactor coolant flow incident:

- Low reactor coolant loop flow reactor trip
- Undervoltage on RCP power supply busses reactor trip
- Underfrequency on RCP power supply busses reactor trip

The reactor trip on low reactor coolant loop flow provides primary protection against partial loss-of-flow conditions. This function is generated by two-out-of-three low-flow signals in any reactor coolant loop. Above Permissive P-8, low flow in any loop will actuate a reactor trip. Between approximately 10-percent power (Permissive P-7) and the power level corresponding to Permissive P-8, low flow in two loops will actuate a reactor trip. Reactor trip on low flow is blocked below Permissive P-7 since there is insufficient heat production to be concerned about DNB.

The reactor trip on RCP undervoltage is provided to protect against conditions that can cause a loss of voltage to all RCPs, that is, loss of offsite power (LOOP). An undervoltage reactor trip serves as an anticipatory backup to the low reactor coolant loop flow trip. The undervoltage trip function is blocked below approximately 10-percent power (Permissive P-7).

The RCP underfrequency reactor trip is provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid. The RCP underfrequency reactor trip function is blocked below Permissive P-7. This trip function also serves as an anticipatory backup to the low reactor coolant loop flow trip.

2.8.5.3.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

This accident was analyzed using the Revised Thermal Design Procedure (RTDP) (Reference 1). Initial core power was assumed to be at its nominal value consistent with steady-state, full-power operation. The RCS pressure and vessel average temperature were assumed to be at their nominal values. Minimum measured flow was also assumed. Uncertainties in initial conditions were accounted for in the DNBR limit value as described in the RTDP.

A conservatively large absolute value of the Doppler-only power coefficient was used. The analysis also assumed a conservative moderator temperature coefficient (MTC) of 0 pcm/°F at

hot full-power (HFP) conditions. This resulted in the maximum core power and hot spot heat flux during the initial part of the transient when the minimum DNBR is reached.

Engineered safety systems (such as safety injection) are not required to function. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

A partial loss of forced reactor coolant flow incident is classified as a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. A complete loss of forced reactor coolant flow incident is classified by the ANS as a Condition III event. However, for conservatism, the incident was analyzed to Condition II criteria. The immediate effect from a complete loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature and subsequent increase in RCS pressure. The following three items identify the acceptance criteria associated with the analysis of the loss of flow events:

- The critical heat flux is not to be exceeded. This is met by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressures in the RCS and MSS are maintained below 110 percent of their respective design pressures.
- The peak linear heat generation rate does not exceed a value that would cause fuel centerline melt.

2.8.5.3.1.2.3 Description of Analyses and Evaluations

With each being applicable to both CPNPP Units 1 and 2 the following loss of forced reactor coolant flow cases were analyzed:

- Loss of power to one RCP (partial loss of flow)
- Loss of power to all RCPs (complete loss of flow)
- 5 Hz/second frequency decay of the RCPs power supply (complete loss of flow)

The transients were analyzed with two computer codes. First, the RETRAN computer code (Reference 2) was used to calculate the loop and core flows during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE computer code (Reference 3) was then used to calculate the heat flux and DNBR based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from the output of the RETRAN transient run. The DNBR transients presented represent the minimum of the typical or thimble cell for the fuel.

2.8.5.3.1.2.4 Results

The partial loss of flow case resulted in a low reactor coolant loop flow reactor trip signal and the complete loss of flow case resulted in an undervoltage reactor trip signal. The frequency decay

complete loss of flow case resulted in an underfrequency reactor trip signal. The VIPRE (Reference 3) analysis for these scenarios confirmed that the minimum DNBR acceptance criterion was met. Fuel cladding damage criteria were not challenged in any of the loss of forced reactor coolant flow cases since the DNB criterion was met.

The analyses of the loss of flow events also demonstrated that the peak RCS and MSS pressures were well below their respective limits.

The most limiting of these cases in terms of the minimum calculated DNBR was the complete loss of flow undervoltage case. The transient results for each case are presented in Figures 2.8.5.3.1-1 through 2.8.5.3.1-21. The sequence of events for each case is presented in Table 2.8.5.3.1-1. Numerical results for the analyses are shown in Table 2.8.5.3.1-2.

The analysis demonstrates that, for the aforementioned loss of flow cases, the DNBR did not decrease below the safety analysis limit value at any time during the transients. Therefore, no fuel or cladding damage is predicted. Also, the peak RCS and MSS pressures remained below their respective limits at all times. All applicable acceptance criteria were therefore met.

The protection features identified in Licensing Report (LR) subsection 2.8.5.3.1.2.1 provide mitigation for the loss of forced reactor coolant flow transients such that the above criteria are satisfied. Furthermore, the results and conclusions of the loss of flow analysis will be confirmed on a cycle-specific basis as part of the normal reload safety evaluation process.

2.8.5.3.1.3 Conclusion

The analyses of the decrease in reactor coolant flow event have been reviewed and it is concluded that the analyses have adequately accounted for plant operations at the proposed uprated power level and were performed using acceptable analytical models. The review further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCS and MSS pressure limits will not be exceeded as a result of this event. Based on this, it is concluded that the plant will continue to meet the requirements of GDCs -10, -15, and -26 following implementation of the uprate. Therefore, the uprate is acceptable with respect to the decrease in reactor coolant flow event.

2.8.5.3.1.4 References

1. WCAP-11397, "Revised Thermal Design Procedure," April 1989.
2. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
3. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

Table 2.8.5.3.1-1		
Time Sequence of Events – Loss of Forced Reactor Coolant Flow		
Case	Event	Time (sec)
Loss of Power to One RCP	Flow Coastdown Begins	0.0
	Reactor Coolant Low-Flow Trip Setpoint Reached	1.4
	Rods Begin to Drop	2.4
	Minimum DNBR Occurs	3.2
Loss of Power to All RCPs	Flow Coastdown Begins	0.0
	Rods Begin to Drop ⁽¹⁾	1.5
	Minimum DNBR Occurs	3.1
5 Hz/sec Frequency Decay of the RCPs Power Supply	Frequency Decay Begins	0.0
	Underfrequency Trip Setpoint Reached	0.6
	Rods Begin to Drop	1.2
	Minimum DNBR Occurs	3.0
Note:		
1. Undervoltage reactor trip is assumed to occur 1.5 seconds following loss of bus voltage		

Table 2.8.5.3.1-2		
Results – Loss of Forced Reactor Coolant Flow		
	Analysis Value	Limit Value
Minimum DNBR – Loss of Power to One RCP	2.173	1.61
Minimum DNBR – Loss of Power to All RCPs	1.901	1.61
Minimum DNBR – Frequency Decay of the RCPs Power Supply	1.921	1.61

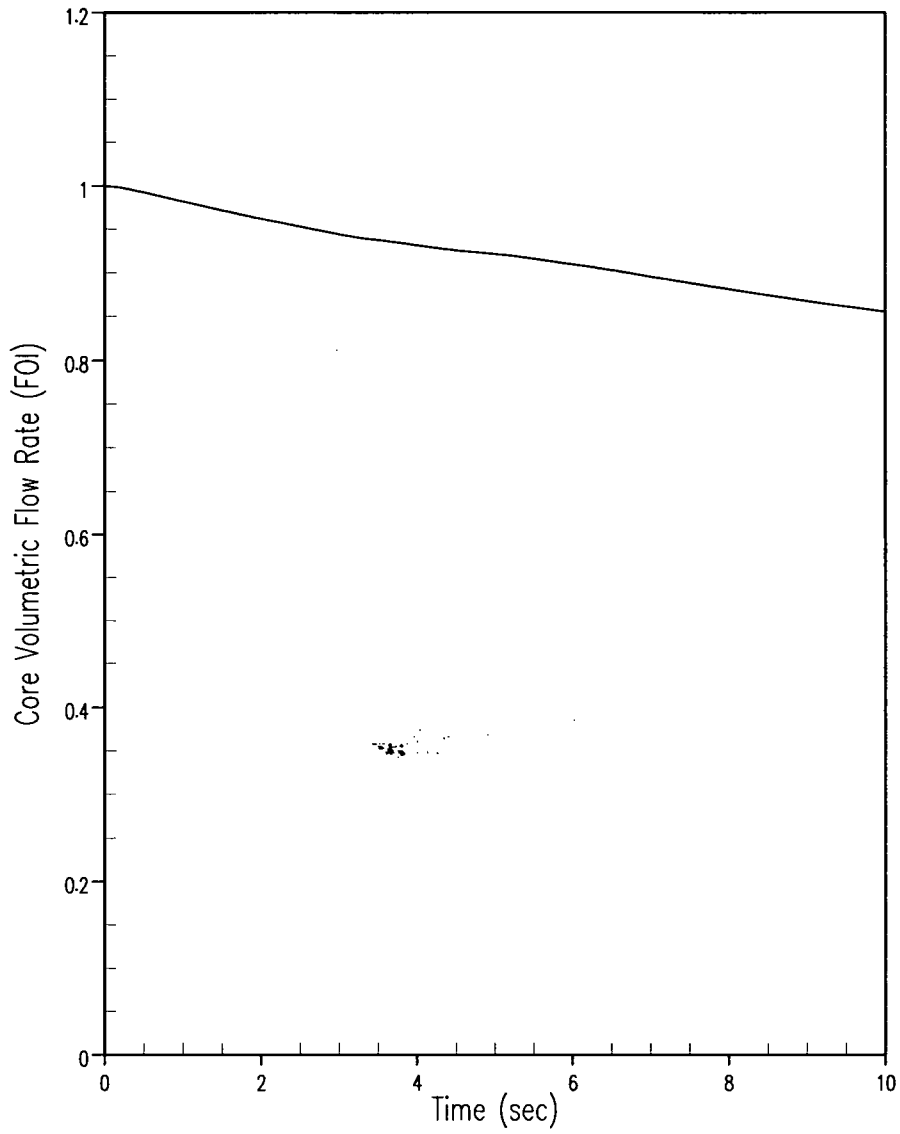


Figure 2.8.5.3.1-1 Partial Loss of Flow – Core Volumetric Flow Rate Versus Time

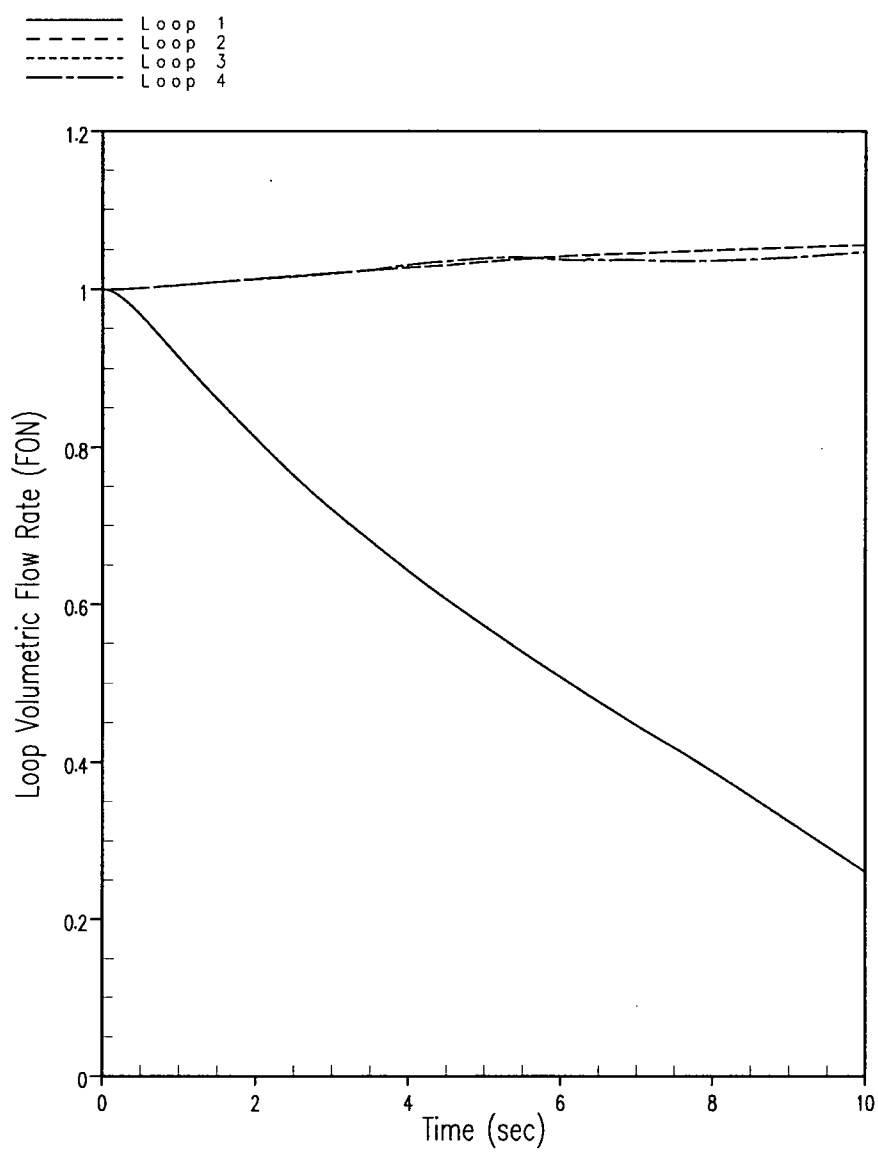


Figure 2.8.5.3.1-2 Partial Loss of Flow – Loop Volumetric Flow Rate Versus Time

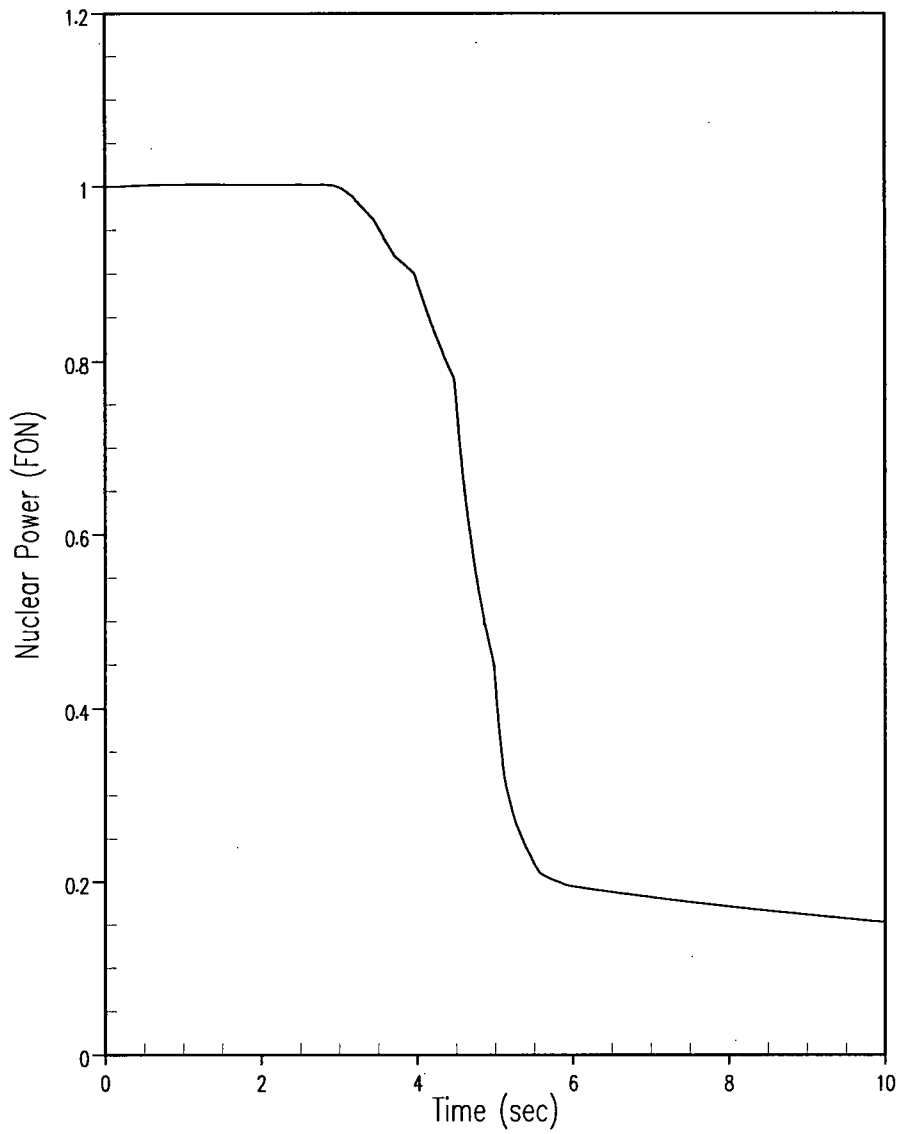


Figure 2.8.5.3.1-3 Partial Loss of Flow – Nuclear Power Versus Time

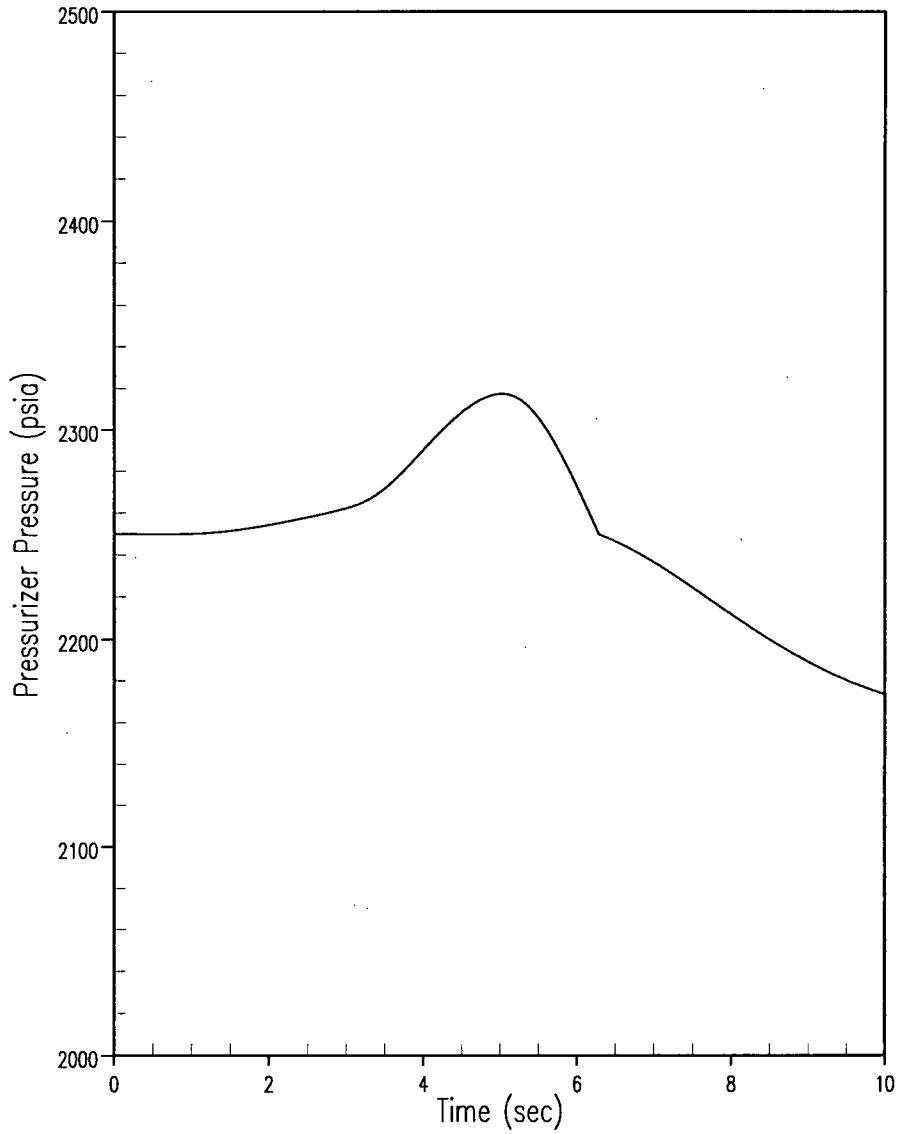


Figure 2.8.5.3.1-4 Partial Loss of Flow – Pressurizer Pressure Versus Time

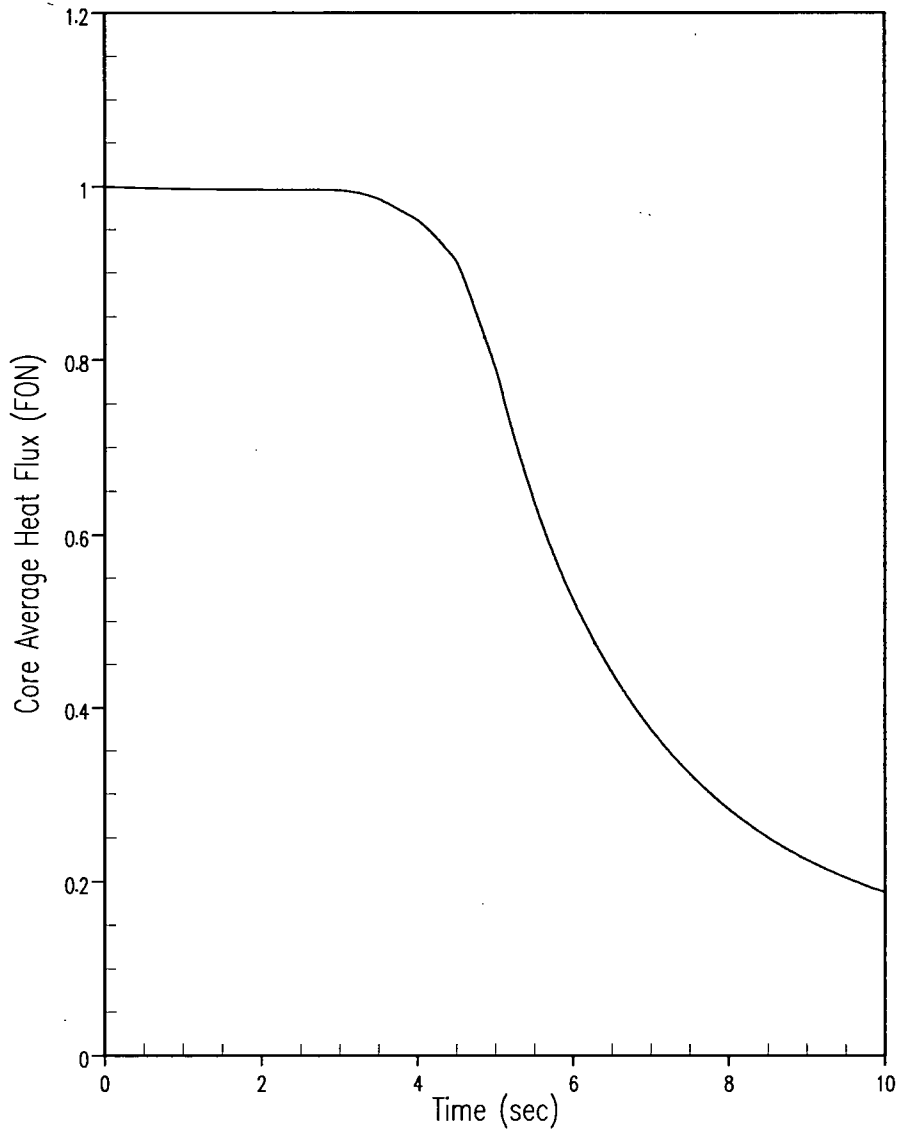


Figure 2.8.5.3.1-5 Partial Loss of Flow – Core Average Heat Flux Versus Time

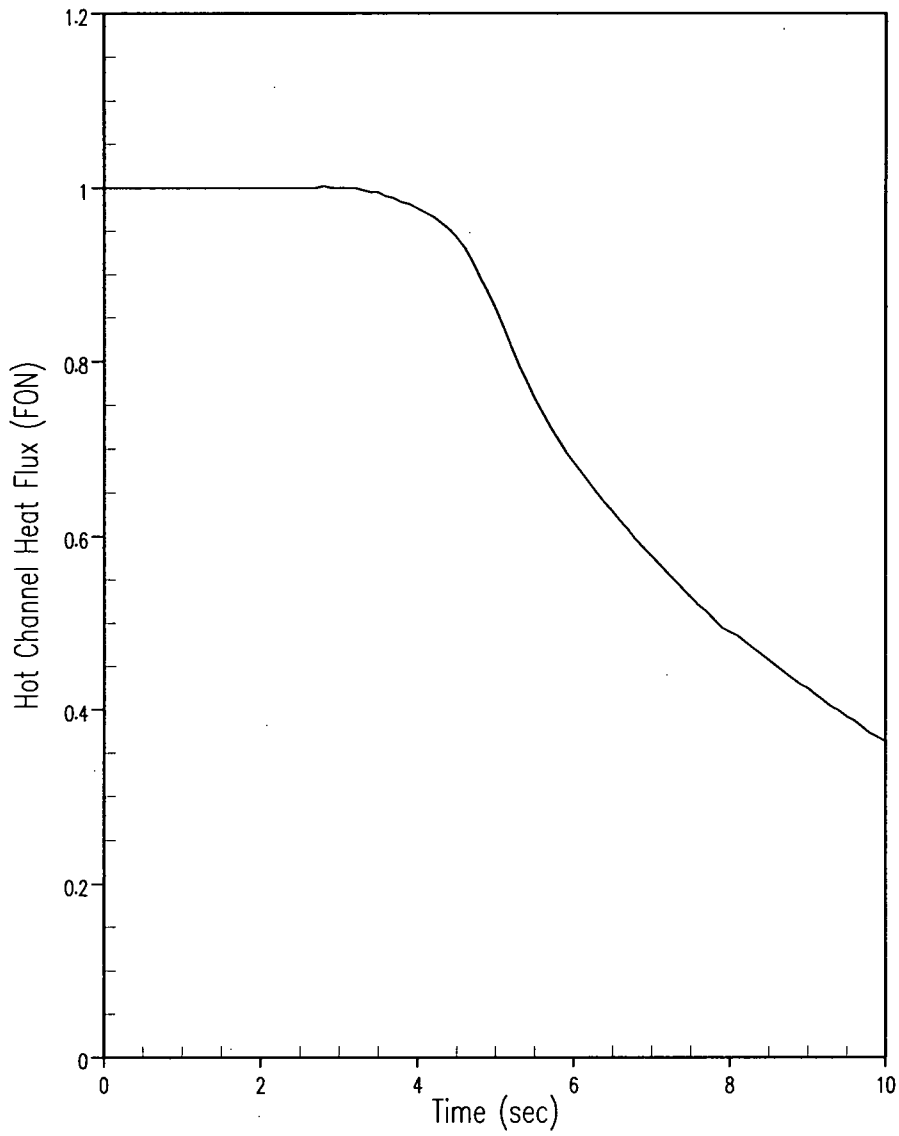


Figure 2.8.5.3.1-6 Partial Loss of Flow – Hot Channel Heat Flux Versus Time

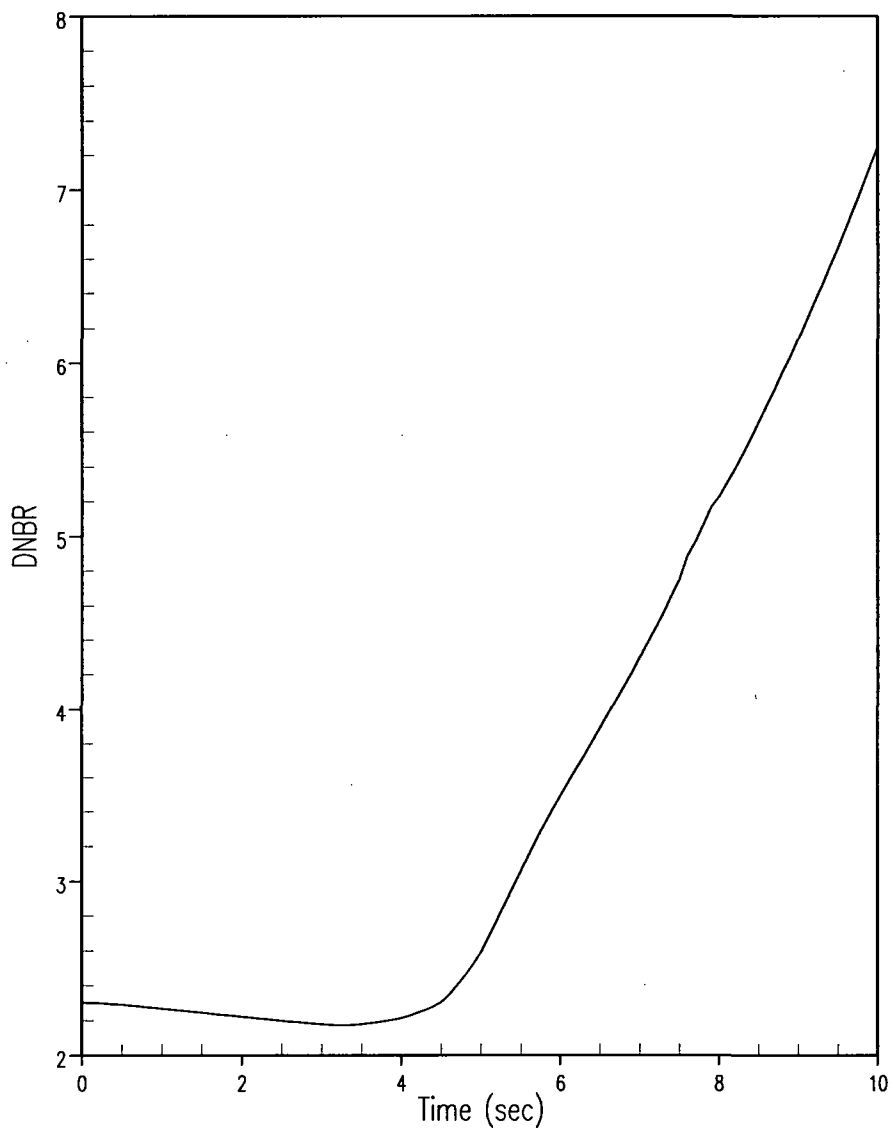


Figure 2.8.5.3.1-7 Partial Loss of Flow – DNBR Versus Time

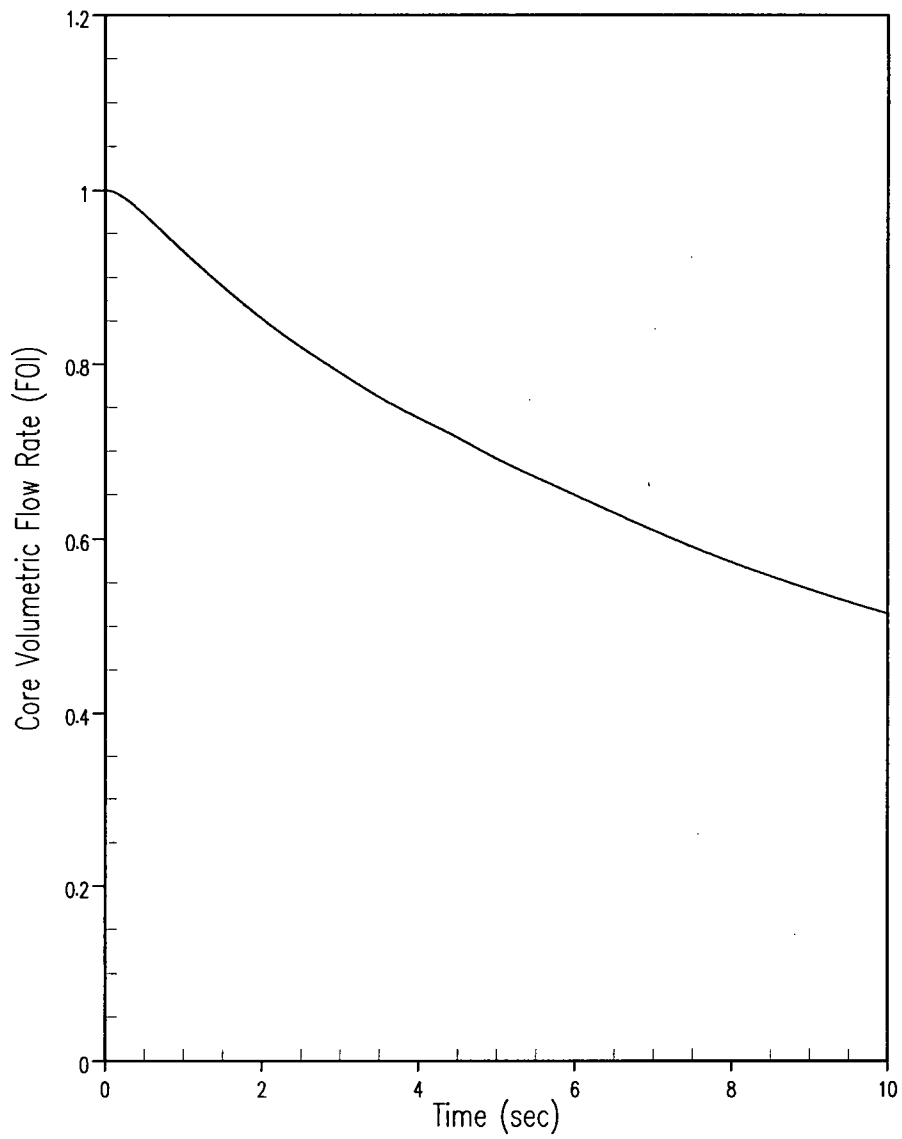


Figure 2.8.5.3.1-8 Complete Loss of Flow Undervoltage – Core Volumetric Flow Rate Versus Time

— Loop 1
- - - Loop 2
- · - Loop 3
- · - Loop 4

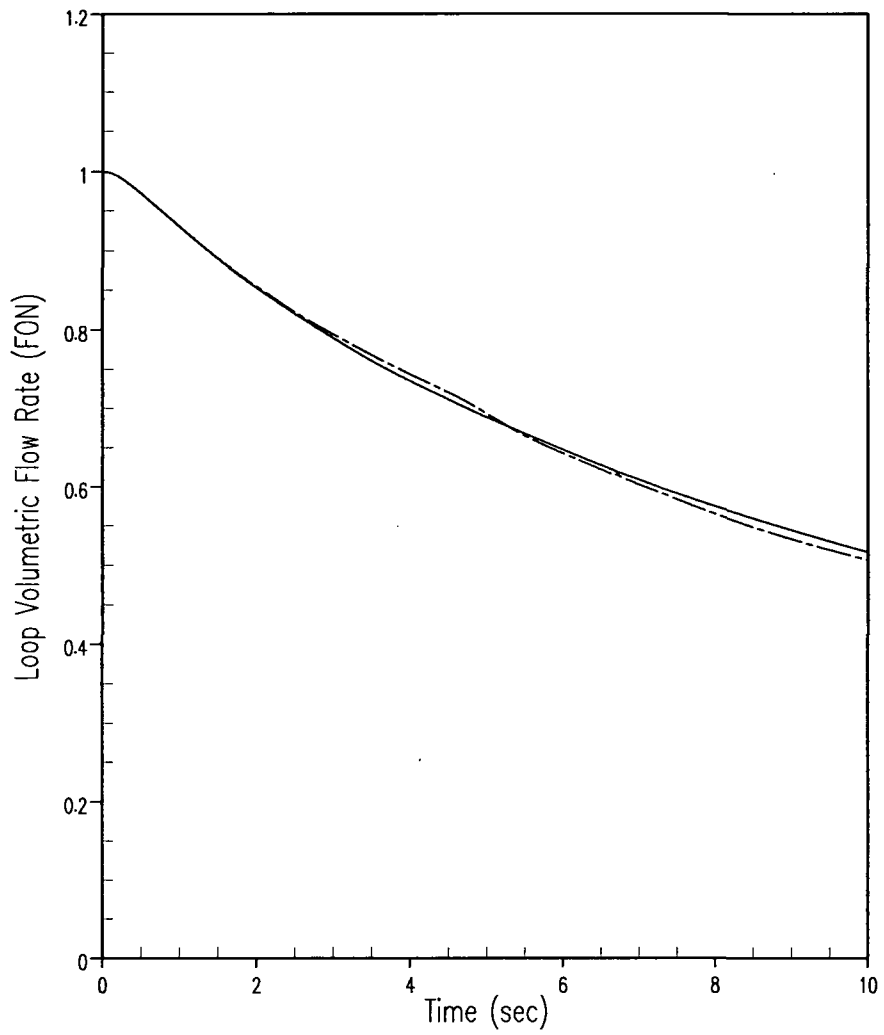


Figure 2.8.5.3.1-9 Complete Loss of Flow Undervoltage – Loop Volumetric Flow Rate Versus Time

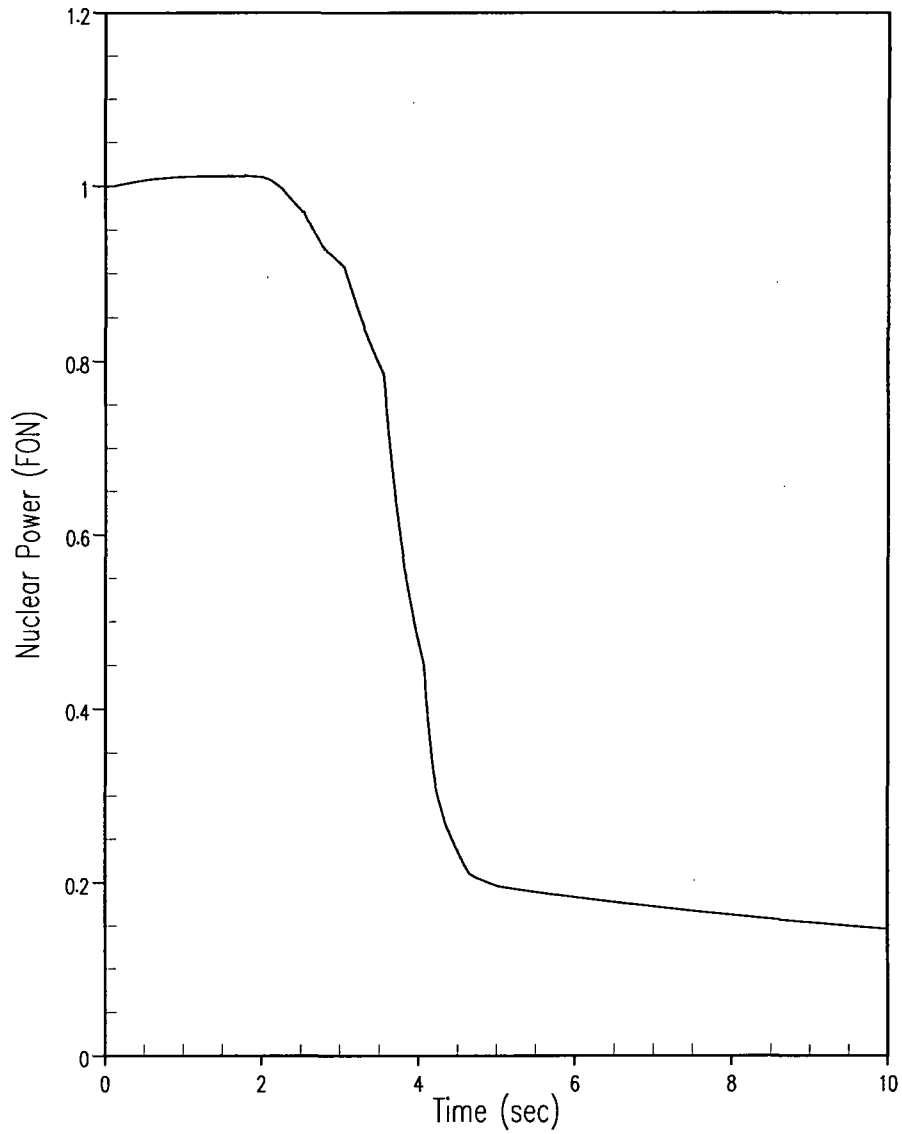


Figure 2.8.5.3.1-10 Complete Loss of Flow Undervoltage – Nuclear Power Versus Time

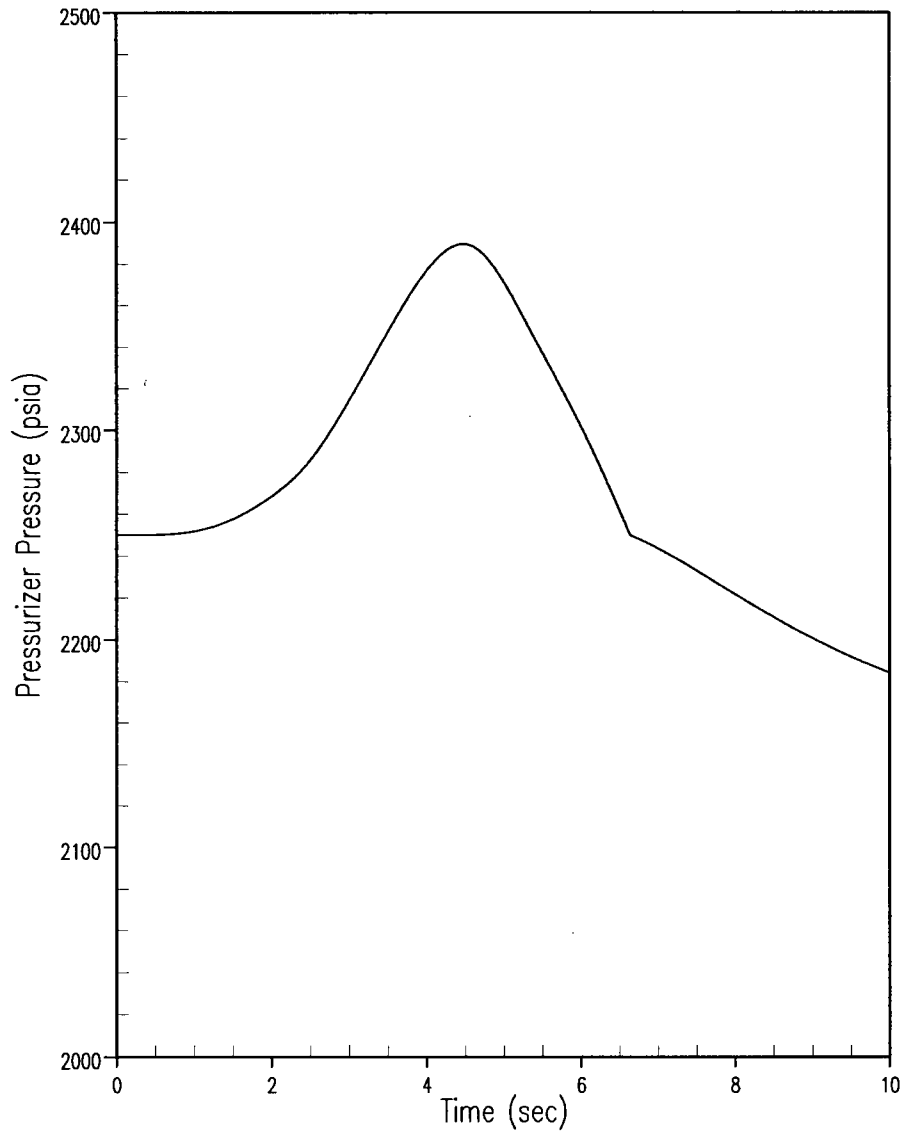


Figure 2.8.5.3.1-11 Complete Loss of Flow Undervoltage – Pressurizer Pressure Versus Time

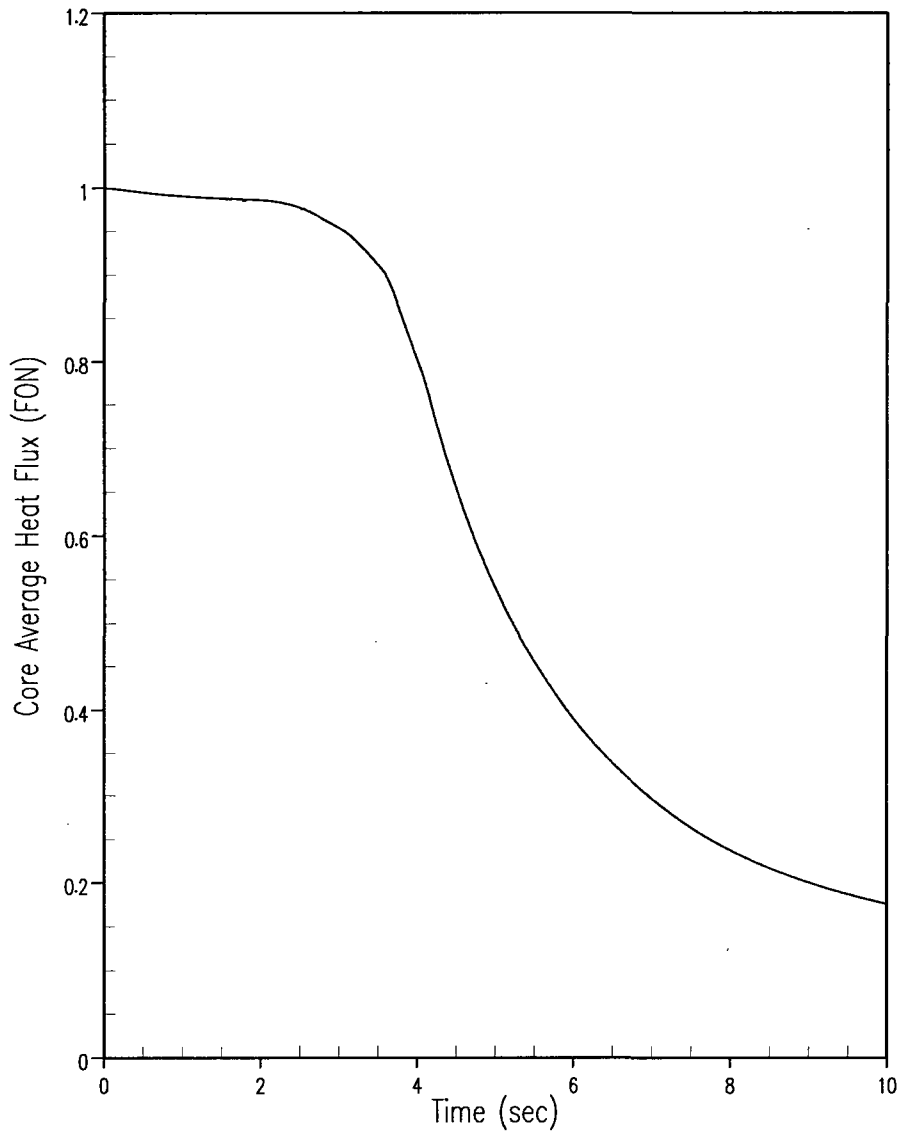


Figure 2.8.5.3.1-12 Complete Loss of Flow Undervoltage – Core Average Heat Flux Versus Time

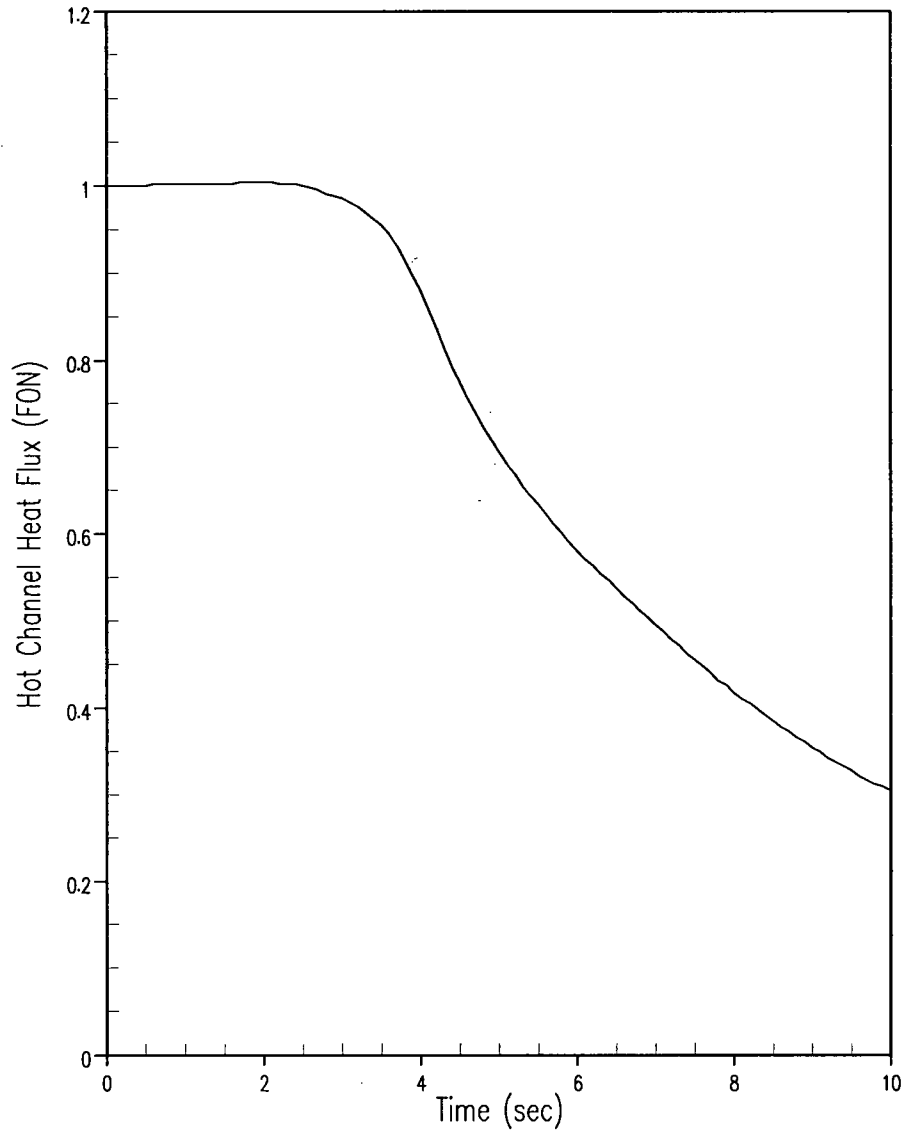


Figure 2.8.5.3.1-13 Complete Loss of Flow Undervoltage – Hot Channel Heat Flux Versus Time

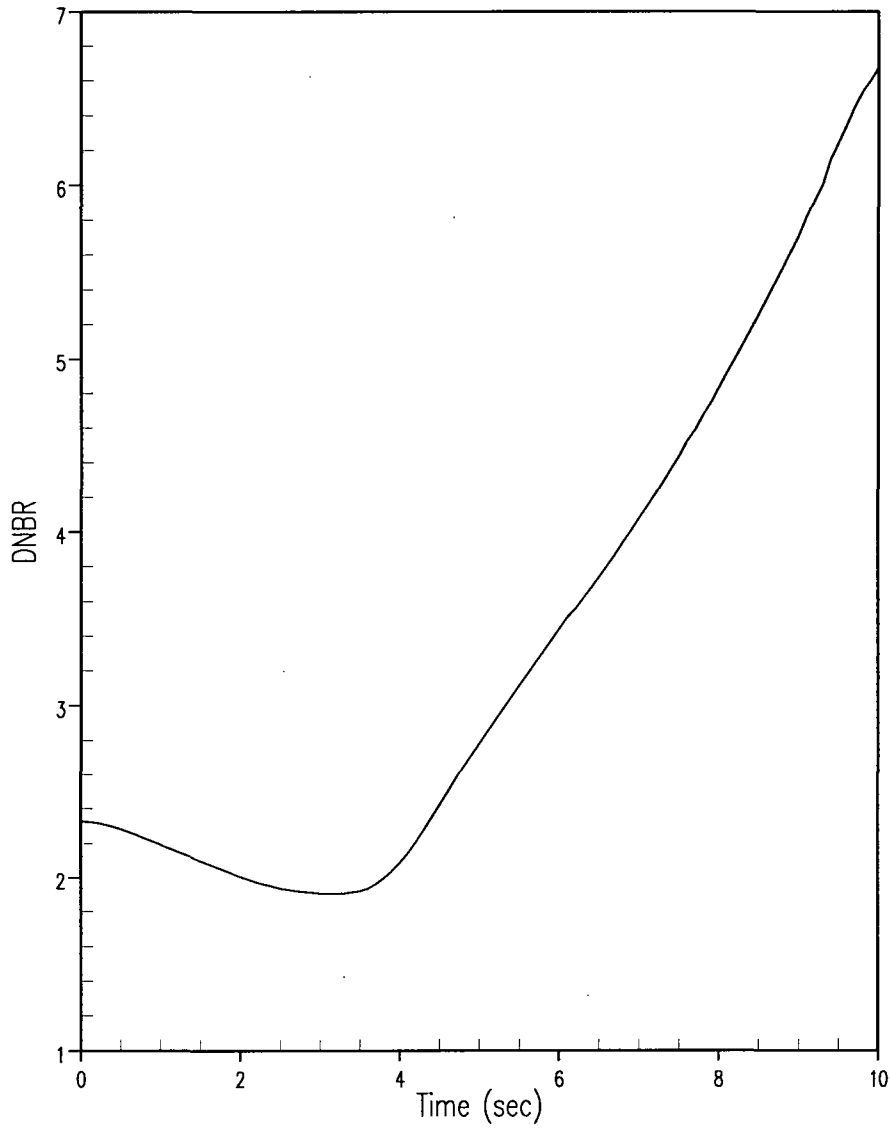


Figure 2.8.5.3.1-14 Complete Loss of Flow Undervoltage – DNBR Versus Time

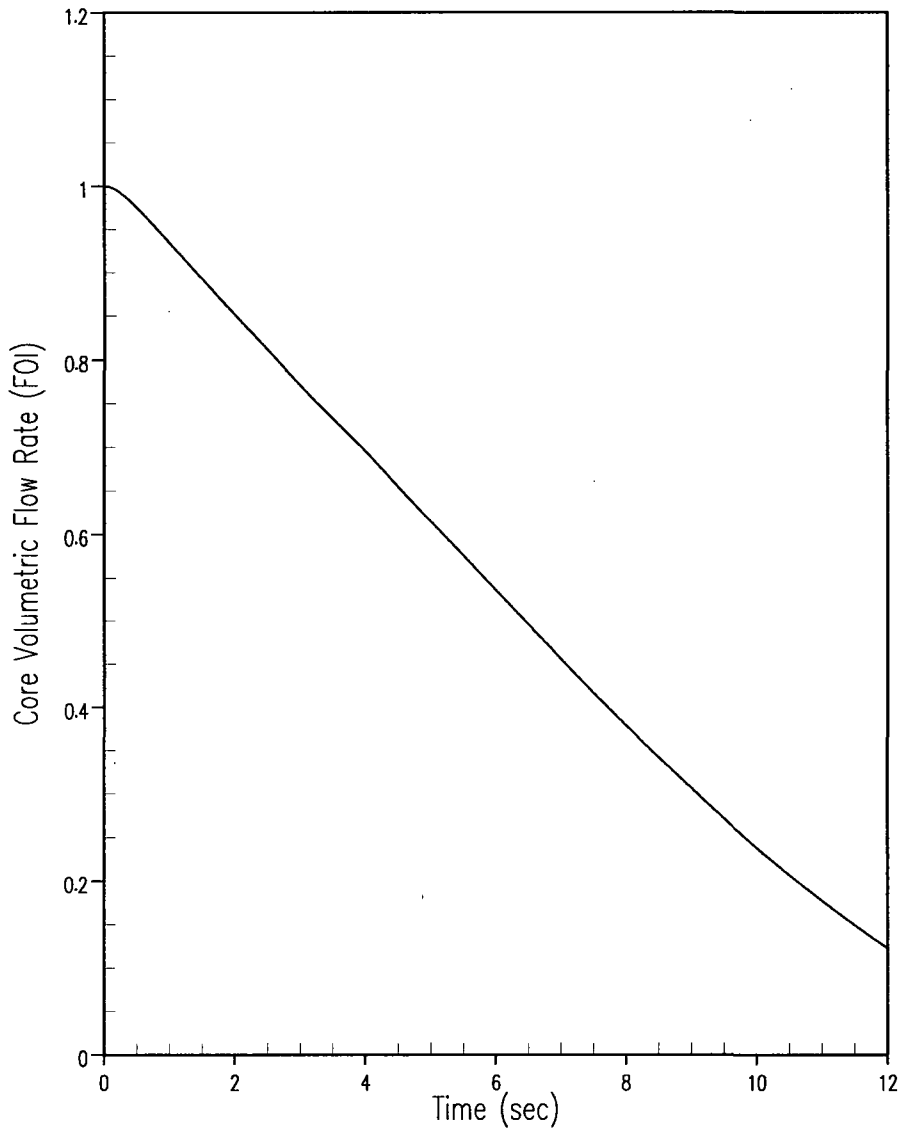


Figure 2.8.5.3.1-15 Complete Loss of Flow Frequency Decay – Core Volumetric Flow Rate Versus Time

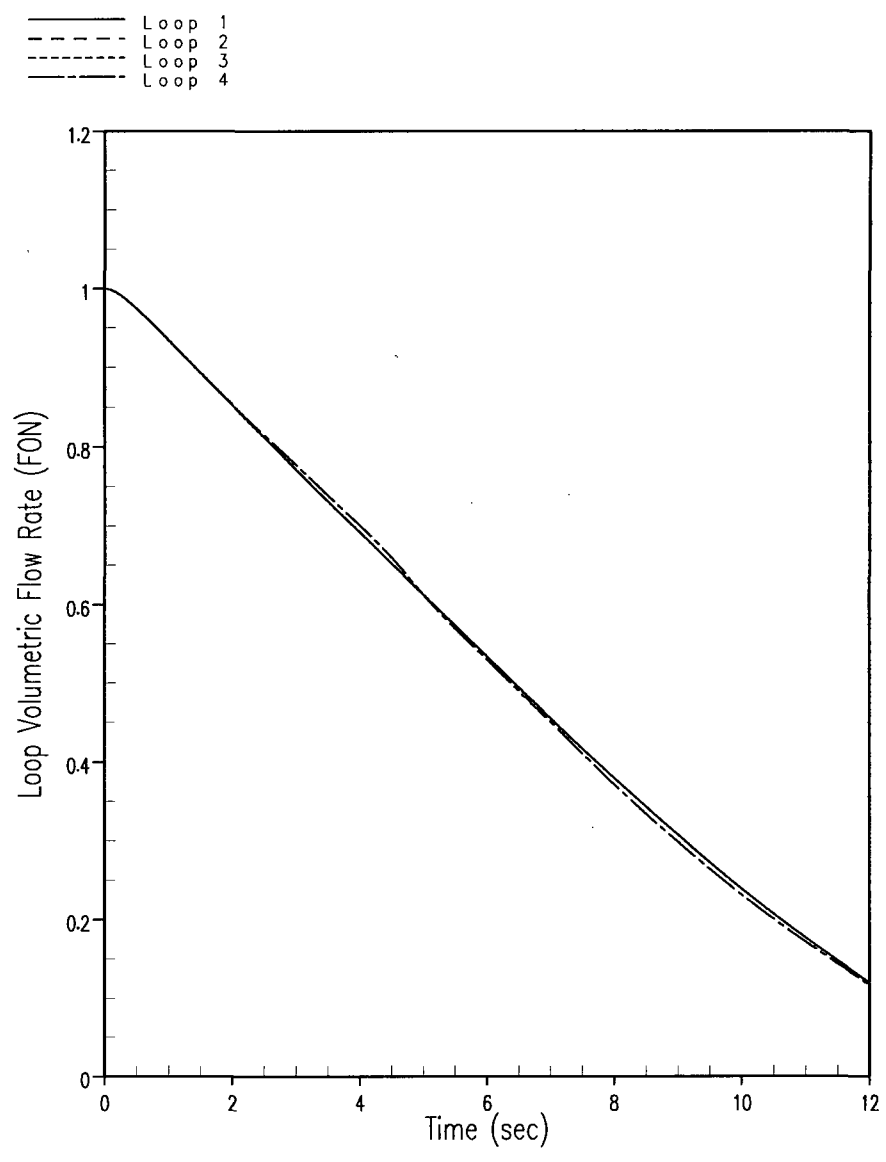


Figure 2.8.5.3.1-16 Complete Loss of Flow Frequency Decay – Loop Volumetric Flow Rate Versus Time

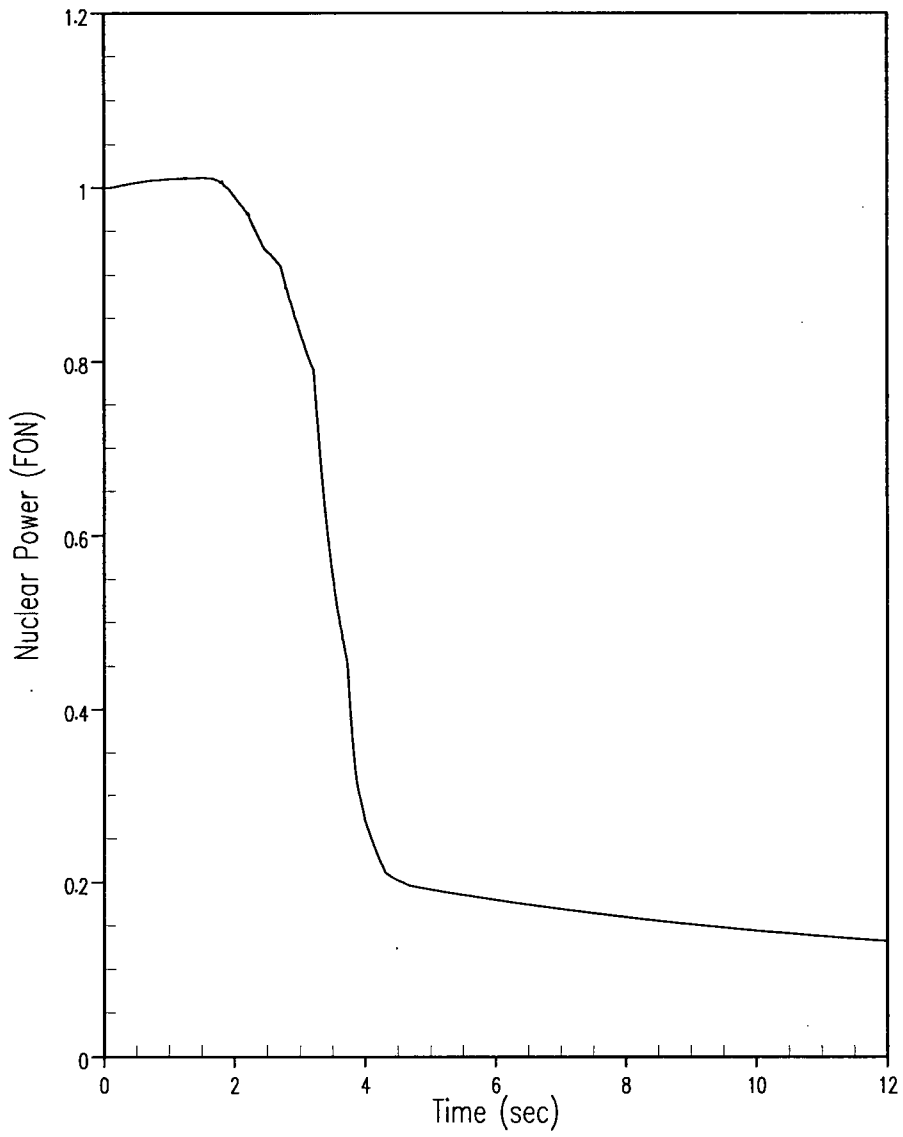


Figure 2.8.5.3.1-17 Complete Loss of Flow Frequency Decay – Nuclear Power Versus Time

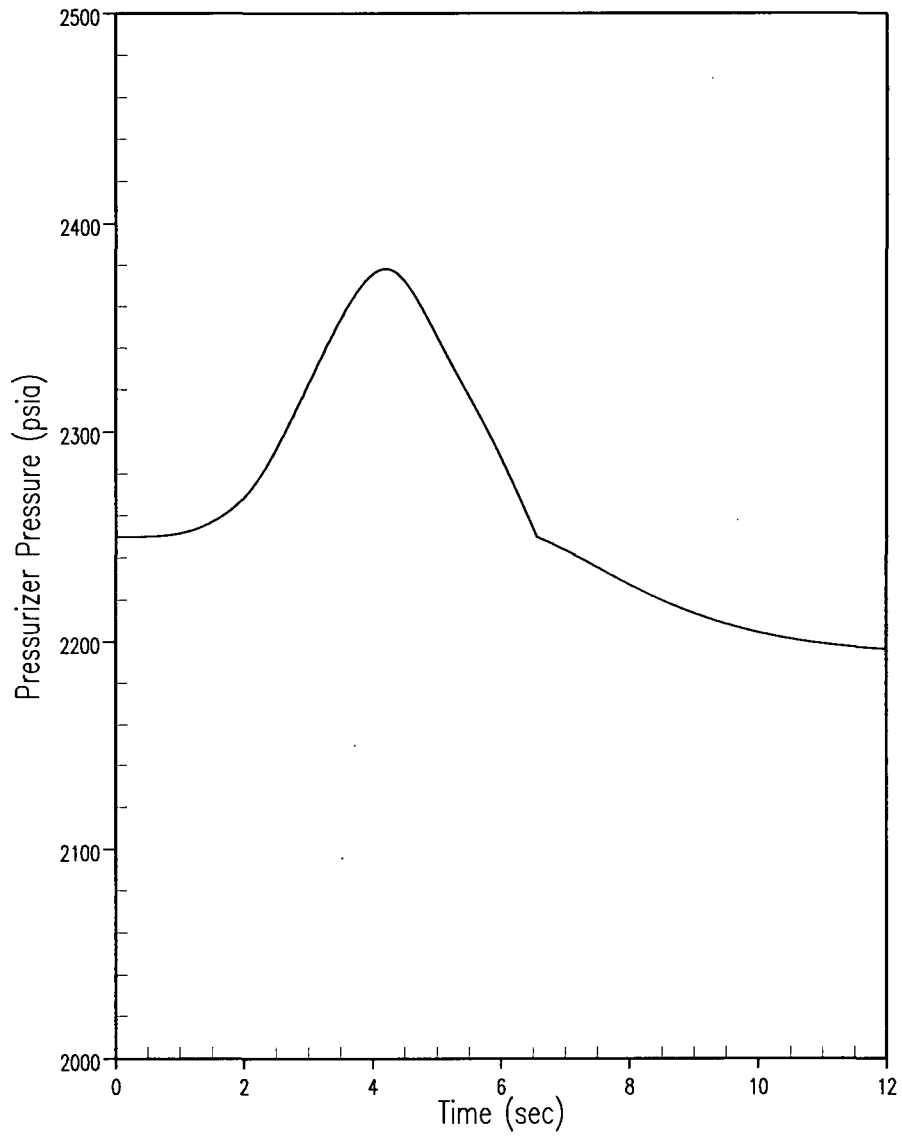


Figure 2.8.5.3.1-18 Complete Loss of Flow Frequency Decay – Pressurizer Pressure Versus Time

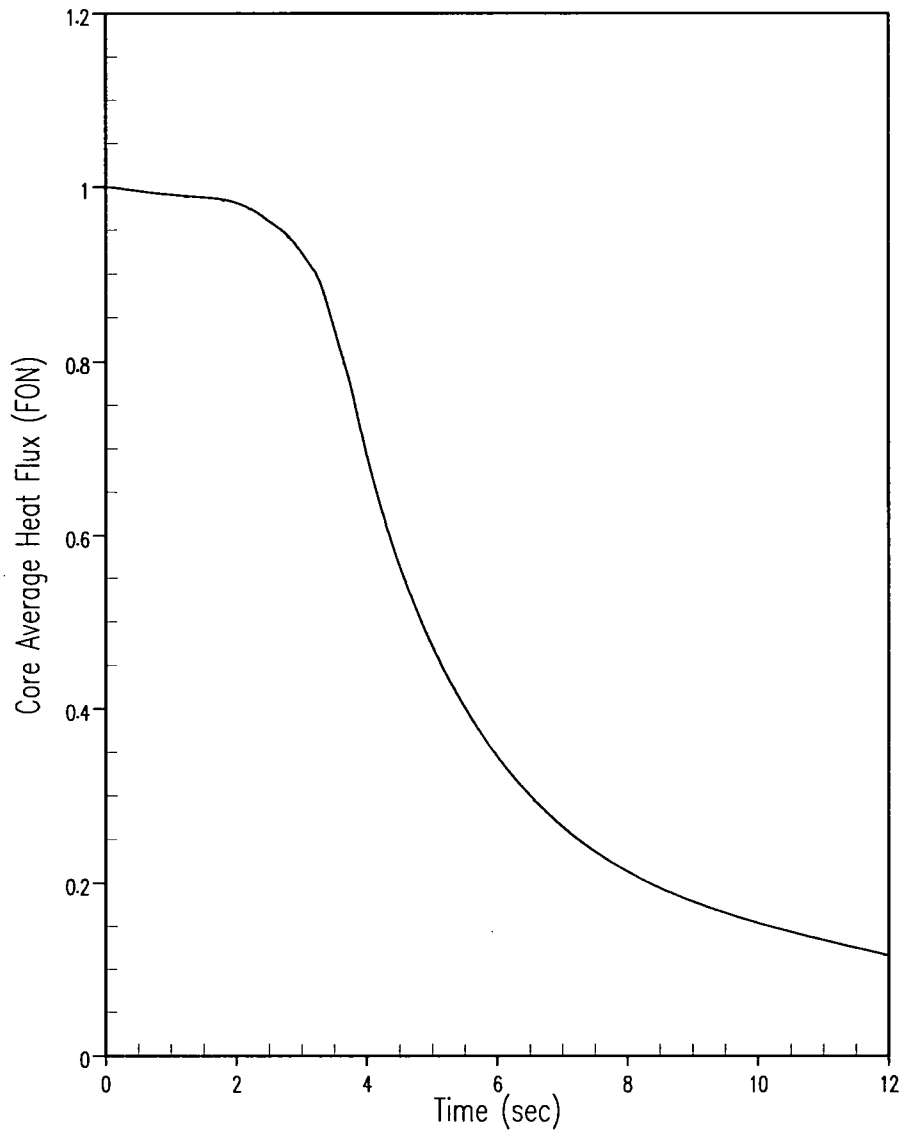


Figure 2.8.5.3.1-19 Complete Loss of Flow Frequency Decay – Core Average Heat Flux Versus Time

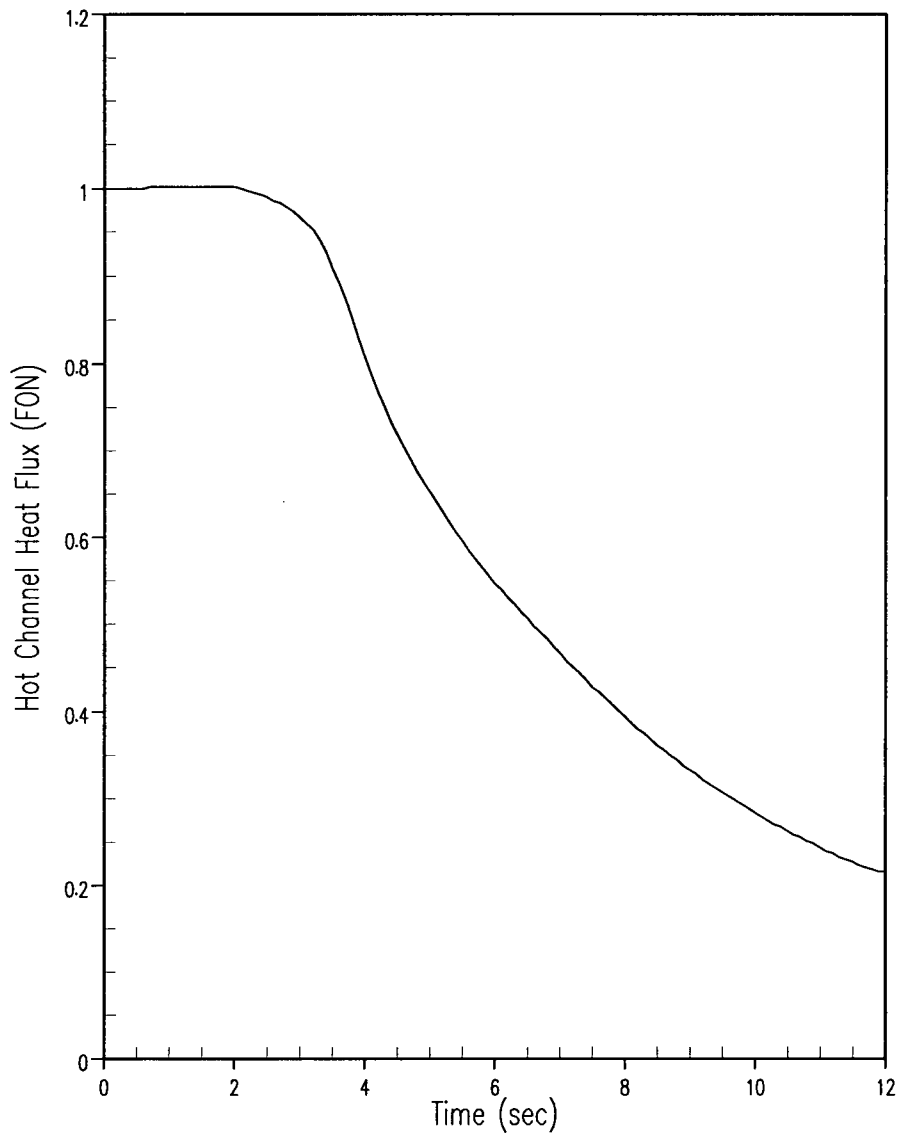


Figure 2.8.5.3.1-20 Complete Loss of Flow Frequency Decay – Hot Channel Heat Flux Versus Time

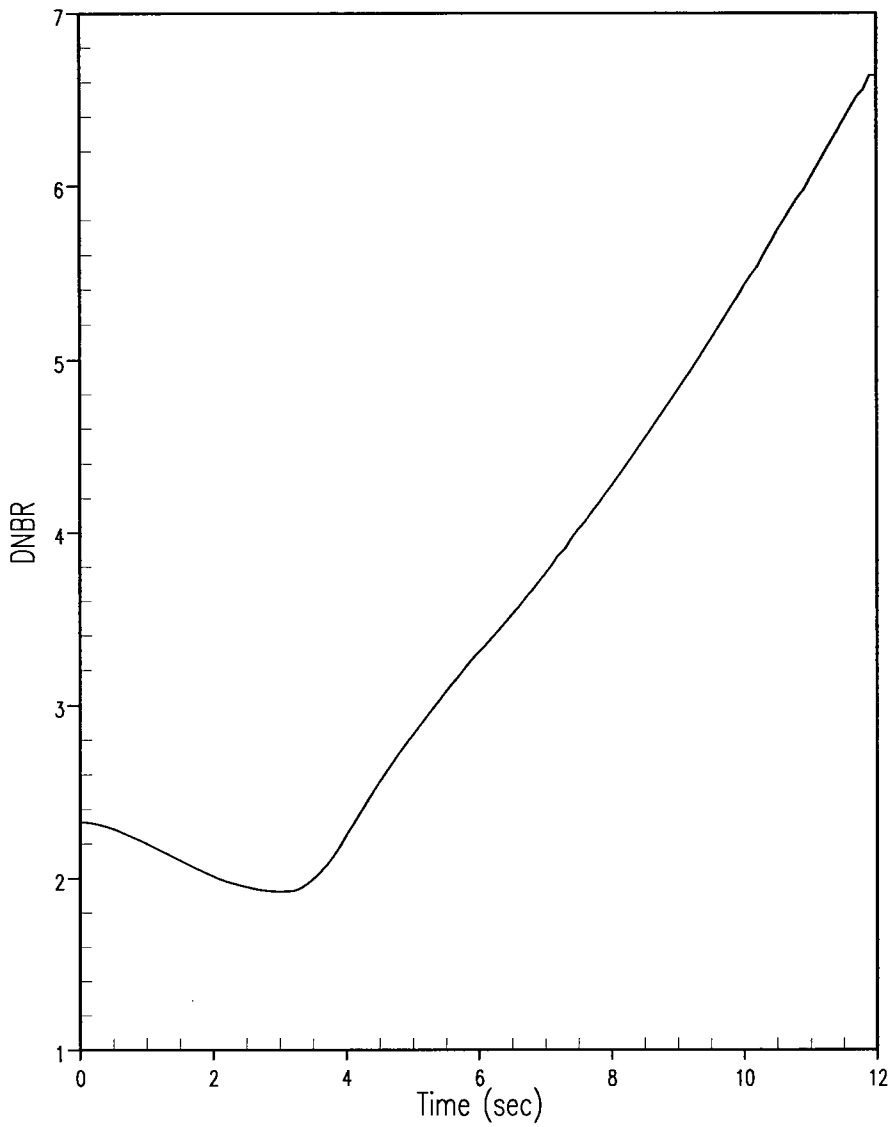


Figure 2.8.5.3.1-21 Complete Loss of Flow Frequency Decay – DNBR Versus Time

2.8.5.3.2 Reactor Coolant Pump Rotor Seizure and RCP Shaft Break

2.8.5.3.2.1 Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump (RCP). Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the plant is at power results in a degradation of core heat transfer, which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor trip and safety systems are actuated to mitigate the transient. The following transient characteristics have been reviewed:

- The postulated initial and long-term core and reactor conditions
- The methods of thermal and hydraulic analyses
- The sequence of events
- The assumed reactions of reactor system components
- The functional and operational characteristics of the reactor trip system (RTS)
- The results of the transient analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained.
- GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.
- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to ensure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-27, Combined Reactivity Control Systems Capability, is described in FSAR Section 3.1.3.8.

The facility is provided with means of making the core subcritical and maintaining it at that level under any anticipated conditions and with an appropriate margin for contingencies. These means are discussed in detail in FSAR Chapters 4 and 9. Combined use of the rod cluster control assemblies (RCCAs) and the chemical shim permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. Upon trip for this determination, the single highest worth control cluster is assumed to be stuck full-out upon trip.

- GDC-28, Reactivity Limits, is described in FSAR Section 3.1.3.9.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values that prevent rupture of the reactor coolant system (RCS) boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The maximum positive reactivity insertion rates for the withdrawal of rod cluster control assembly (RCCA) and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCA and of the chemical and volume control system (CVCS). Technical Specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of FSAR Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in FSAR Section 4.3. The capability of the CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in FSAR Chapter 15.

Assurance of core cooling capability following Condition IV accidents, such as rod ejections, steam line break, and similar accidents, is given by keeping the RCPB stresses within faulted condition limits as specified by applicable American Society of Mechanical Engineering (ASME) codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of necessary safety features.

- GDC 31, Fracture Prevention of Reactor Coolant Pressure Boundary, is described in FSAR Section 3.1.4.2.

Close control is maintained over material selection and fabrication for the RCS to ensure that the boundary behaves in a nonbrittle manner. Those RCS materials which are exposed to the coolant are corrosion resistant, stainless steel, or Inconel. The reference temperature of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR Part 50, Appendix G.

As part of the reactor vessel specification, certain requirements which are not specified by the applicable ASME codes are performed, as follows:

1. Ultrasonic Testing

In addition to code requirements, the performance of a 100-percent volumetric ultrasonic test of reactor vessel plate for shear wave and a post-hydro-test ultrasonic map of all welds in the pressure vessel are required. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the codes is also required to preclude interpretation problems during in-service inspection.

2. Radiation Surveillance Program

In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens. These programs evaluate the effect of radiation on the fracture toughness of reactor vessel steels based on the measured change in reference transition temperature and fracture mechanics measurements. These measurements are performed in accordance with American Society Testing Material (ASTM) E 185-1982, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, and the requirements of 10 CFR Part 50, Appendix H.

3. Material chemistry (copper, nickel, phosphorous, sulfur, and vanadium) of the reactor vessel core region is controlled to reduce sensitivity to embrittlement, which is caused by irradiation over the life of the plant.

The fabrication and quality control techniques used in the fabrication of the RCS are equivalent to those used for the reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generator are governed by the requirements of ASME Codes (see FSAR Chapter 5).

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Appendix G, Protection Against Non-Ductile Failure. The approach specifies that allowed stress intensity factors for all vessel operating conditions shall not exceed the reference stress intensity factor (KIR) for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material due to irradiation.

The current licensing basis concludes that all applicable acceptance criteria are met for the postulated instantaneous seizure of the rotor or break of the shaft of an RCP. Specifically, as noted in the FSAR Sections 15.3.3.3 and 15.3.4.2, the maximum RCS pressure reached during the transient is less than that which would cause stresses to exceed the stress limits. Therefore, the integrity of the primary coolant system is not endangered. The peak cladding surface temperature calculated for the hot spot during the worst transient remains considerably

less than 2,700°F, thus ensuring that the core will remain in place and intact with no loss of core cooling capability. Lastly, the number of rods experiencing departure from nucleate boiling (DNB) remains less than the number of rods assumed to experience DNB for the radiological analysis. Hence, the radiological doses are below the dose values set forth in 10 CFR 100.

2.8.5.3.2.2 Technical Evaluation

The specific acceptance criteria for this event are as follows:

- The peak cladding temperature must remain below 2,700°F and the maximum zirconium-water reaction must remain below 16 percent. Appropriate margin for malfunctions, such as stuck rods, were accounted for in the safety analysis assumptions. Demonstrating that these limits are met satisfies the requirements of GDC-27 and GDC-28.
- Pressures in the RCS are to be maintained less than that which would cause stresses to exceed the faulted condition stress limits for very low probability events such as locked rotor. Demonstrating that this limit is met satisfies the requirements of GDC-28.
- The total percentage of rods-in-DNB should be less than that analyzed in the dose analysis. The specific limit for the uprate analysis is 10 percent. Demonstrating that this limit is met satisfies the requirements of GDC-27 and GDC-28.

The discussion below demonstrates that all applicable acceptance criteria were met for this event at CPNPP Units 1 and 2 at uprated conditions.

2.8.5.3.2.2.1 Introduction

The event postulated is an instantaneous seizure of an RCP rotor or the sudden break of the shaft of the RCP (FSAR Sections 15.3.3 and 15.3.4). Flow through the affected reactor coolant loop is rapidly reduced, leading to initiation of a reactor trip on a low reactor coolant flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced; first because the reduced flow results in a decreased tube-side film heat transfer coefficient, and second because the temperature differential between the reactor coolant in the tubes and the shell-side fluid is decreased. The rapid expansion of the coolant in the reactor core causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves (PORVs), and opens the pressurizer safety valves (PSVs), in that sequence. The PORVs are designed for reliable operation and are expected to function properly during the event. However, for conservatism, their pressure-reducing effect, as well as the pressure-reducing effect of the pressurizer spray, were not included in the analysis.

The consequences of a locked rotor (that is, an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is a reduced core flow when compared to the locked rotor scenario. The analysis considers only one scenario; it represents the most limiting combination of conditions for the locked rotor and pump shaft break events.

2.8.5.3.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

There were two locked rotor cases analyzed, with each being applicable to both CPNPP Units 1 and 2: one for peak RCS pressure and peak cladding temperature (PCT) concerns and a second to determine the percentage of rods-in-DNB. The case evaluating peak RCS pressure and PCT assumed one locked rotor and shaft break with all reactor coolant loops in operation. The first case made assumptions designed to maximize the RCS pressure and cladding temperature transients. It was done using the Standard Thermal Design Procedure (STDP). Initial core power, reactor coolant temperature, and pressure were assumed to be at their maximum values consistent with full-power conditions, including allowances for calibration and instrument errors. This assumption resulted in a conservative calculation of the coolant surge into the pressurizer, which in turn resulted in a maximum calculated peak RCS pressure.

The second case was run to confirm that the percentage of rods-in-DNB is less than that assumed in the radiological analysis. As in the peak RCS pressure/PCT case, one locked rotor and shaft break was assumed with all reactor coolant loops in operation. Initial core power was assumed to be at its nominal value consistent with steady-state, full-power operation. The RCS pressure and vessel average temperature were assumed to be at their nominal values. Minimum measured flow was also assumed. Uncertainties in initial conditions were accounted for in the DNBR limit value as described in the Revised Thermal Design Procedure (RTDP) (Reference 1).

A zero moderator temperature coefficient (MTC) and a conservatively large (absolute value) Doppler-only power coefficient were assumed in the analysis. The negative reactivity from control rod insertion/scram was based on 4.0-percent $\Delta k/k$ trip reactivity from hot full power (HFP).

Engineered safety systems (such as safety injection) are not required to function. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

The RCP locked rotor/shaft break accident is classified as a Condition IV event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. An RCP locked rotor/shaft break results in a rapid reduction in forced reactor coolant loop flow that increases the reactor coolant temperature and subsequently causes the fuel cladding temperature and RCS pressure to increase. The following items summarize the criteria associated with this event:

- Fuel cladding damage, including melting, due to increased reactor coolant temperatures must be prevented. This is precluded by demonstrating that the maximum cladding temperature at the core hot spot remains below 2,700°F, and the zirconium-water reaction at the core hot spot is less than 16 percent by weight.
- Pressures in the RCS are to be maintained below 110 percent of the design pressure.
- The total percentage of rods-in-DNB is less than that analyzed in the dose analysis. The specific limit for the uprate analysis is 10 percent.

2.8.5.3.2.2.3 Description of Analyses and Evaluations

The locked-rotor transient was analyzed with two primary computer codes. First, the RETRAN computer code (Reference 2) was used to calculate the loop and core flows during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code (Reference 3) was then used to calculate the PCT using the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN.

For the peak RCS pressure evaluation, the initial pressure was conservatively estimated to be 30 psi above the nominal pressure of 2,250 psia. This was to allow for initial condition uncertainties in the pressurizer pressure measurement and control channels. This was done to obtain the highest possible rise in the coolant pressure during the transient. The pressure response reported in Table 2.8.5.3.2-2 corresponds to the location in the RCS that has the maximum pressure, that is, in the lower plenum of the reactor vessel.

No credit was taken for the pressure-reducing effect of the pressurizer PORVs, pressurizer spray, or steam dump. Although these systems are expected to function and would result in a lower peak pressure, an additional degree of conservatism was provided by not including their effect. The PSV model included a +2-percent valve opening tolerance above the nominal setpoint of 2,485 psig plus a 1-percent set pressure shift due to the water-filled pressurizer loop seals.

The film boiling coefficient was calculated in the VIPRE code (Reference 3) using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties were evaluated at film temperature. The program calculated the film coefficient at every time step based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and RCS flow rate as a function of time were based on the RETRAN results.

The magnitude and time dependence of the heat transfer coefficient between the fuel and cladding (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with the initial fuel temperature to approximately 10,000 Btu/hr-ft²-°F at the initiation of the transient. Therefore, the large amount of energy stored in the fuel because of the small initial value was released to the cladding at the initiation of the transient.

The zirconium-steam reaction can become significant above 1,800°F (cladding temperature). The Baker-Just parabolic rate equation was used to define the rate of zirconium-steam reaction. The effect of the zirconium-steam reaction was included in the calculation of the PCT temperature transient.

2.8.5.3.2.2.4 Results

With respect to the peak RCS pressure, PCT, and zirconium-steam reaction, the analysis demonstrated that all applicable acceptance criteria were met for CPNPP Units 1 and 2. The calculated sequence of events is presented in Table 2.8.5.3.2-1 for the locked rotor/shaft break event. The results of the calculations (peak pressure, PCT, and zirconium-steam reaction) are summarized in Table 2.8.5.3.2-2. The transient results for the peak pressure/PCT case are provided in Figures 2.8.5.3.2-1 through 2.8.5.3.2-6 while the transient results for the rods-in-DNB case are provided in Figures 2.8.5.3.2-7 through 2.8.5.3.2-12.

The analysis performed for the uprate demonstrated that, for the locked rotor event, the PCT calculated for the hot spot during the worst transient remained considerably less than 2,700°F, and the amount of zirconium-water reaction was small. Under such conditions, the core would remain in place and intact with no loss of core cooling capability.

The analysis also confirmed that the peak RCS pressure reached during the transient was less than the acceptance limit, and thereby, the integrity of the primary coolant system was demonstrated. The total number of rods-in-DNB was less than 10 percent. The low reactor coolant flow reactor trip function provided mitigation for the locked rotor/shaft break transient such that the above criteria were satisfied. Furthermore, the results and conclusions of this analysis will be confirmed on a cycle-specific basis as part of the normal reload safety evaluation process.

2.8.5.3.2.3 Conclusion

The analyses of the sudden decrease in core coolant flow events have been reviewed and it is concluded that the analyses have adequately accounted for plant operation at the proposed uprated power level and were performed using acceptable analytical models. The review further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCS pressure limit will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability

of propagating fracture of the reactor coolant pressure boundary is minimized, and adequate core cooling will be provided. Based on this, it is concluded that the plant will continue to meet the requirements of GDCs -27, -28, and -31 following implementation of the uprate. Therefore, the uprate is acceptable with respect to the sudden decrease in core coolant flow events.

2.8.5.3.2.4 References

1. WCAP-11397, "Revised Thermal Design Procedure," April 1989.
2. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
3. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

Table 2.8.5.3.2-1		
Time Sequence of Events – Single RCP Locked Rotor/Shaft Break		
Case	Event	Time (sec)
Locked Rotor (LR) – Overpressurization/Peak Cladding Temperature	Loop 1 RCP Rotor Seizes	0.0
	Reactor Coolant Low-Flow Trip Setpoint Reached	0.04
	Rods Begin to Drop	1.04
	Remaining Pumps Lose Power and Begin to Coast Down	1.04
	Maximum Cladding Temperature Occurs	3.65
	Maximum RCS Pressure Occurs	4.94
LR – Rods-in-DNB	Loop 1 RCP Rotor Seizes	0.0
	Reactor Coolant Low-Flow Trip Setpoint Reached	0.04
	Rods Begin to Drop	1.04
	Remaining Pumps Lose Power and Begin to Coast Down	1.04
	Minimum DNBR Occurs	2.70

Table 2.8.5.3.2-2		
Results – Single RCP Locked Rotor/Shaft Break		
Criteria	Analysis Value	Limit
Peak Cladding Temperature at Core Hot Spot, °F	1,723.6	2,700.
Maximum Zirconium-Water Reaction at Core Hot Spot, %	0.22	16.0
Maximum RCS Pressure, psia	2,574.5	2,748.2
Total number of rods-in-DNB, %	< 10% ⁽¹⁾	10%
Note:		
1. See Licensing Report Section 2.8.3 for detailed results.		

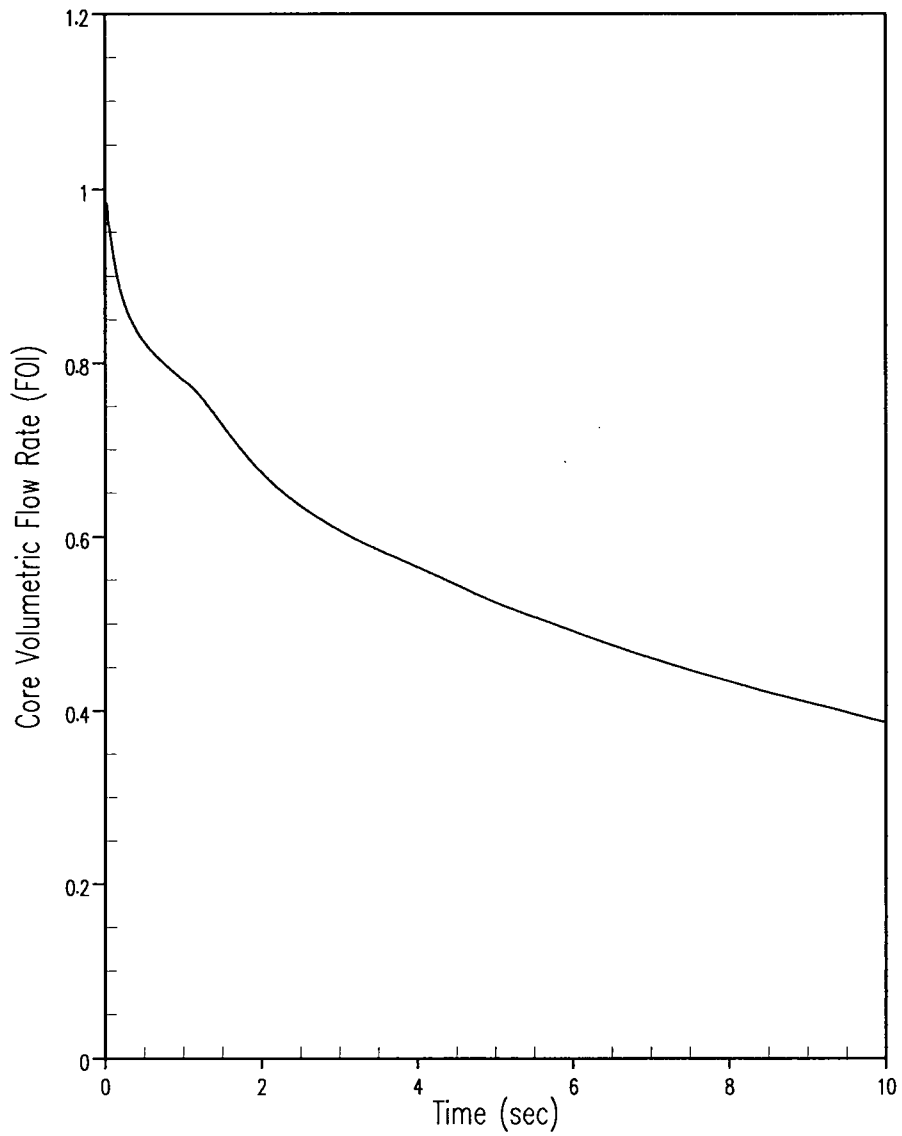


Figure 2.8.5.3.2-1 RCP Locked Rotor/Shaft Break Overpressurization/Peak Clad Temperature – Core Volumetric Flow Rate Versus Time

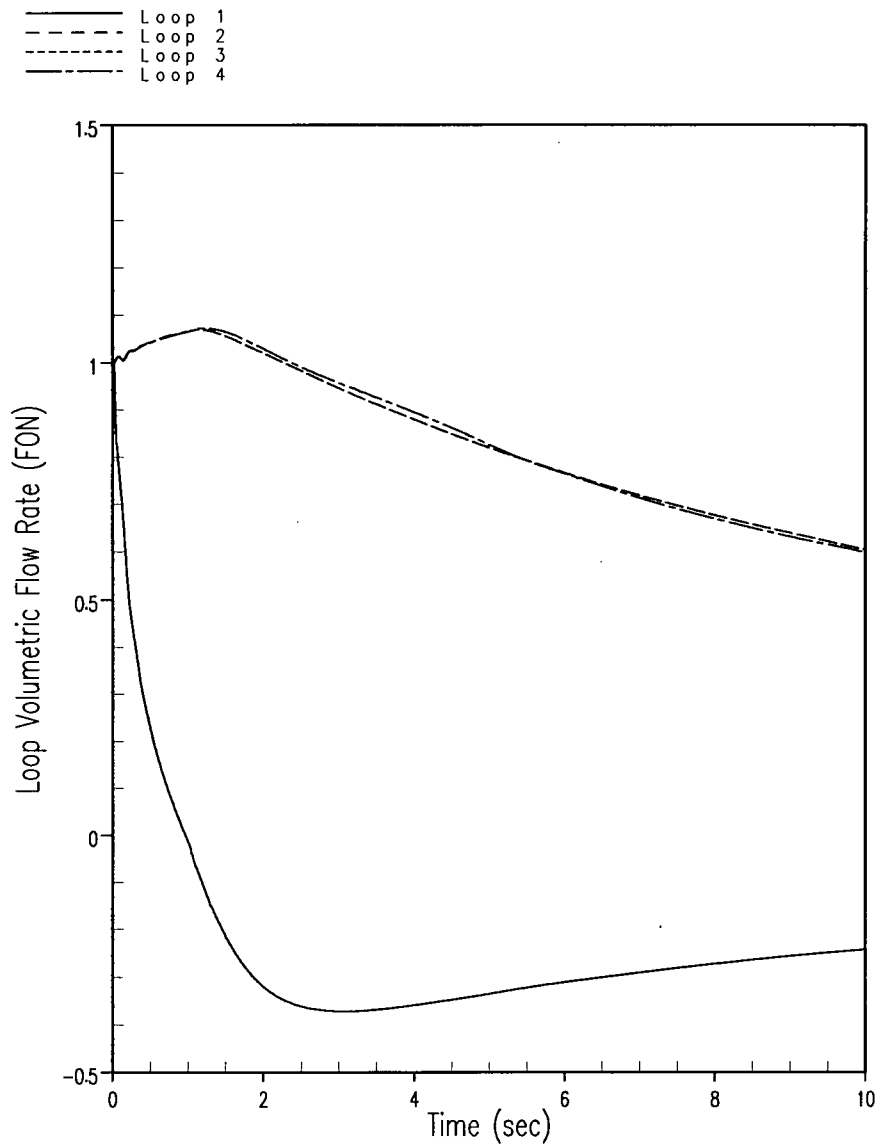


Figure 2.8.5.3.2-2 RCP Locked Rotor/Shaft Break Overpressurization/Peak Clad Temperature – Loop Volumetric Flow Rate Versus Time

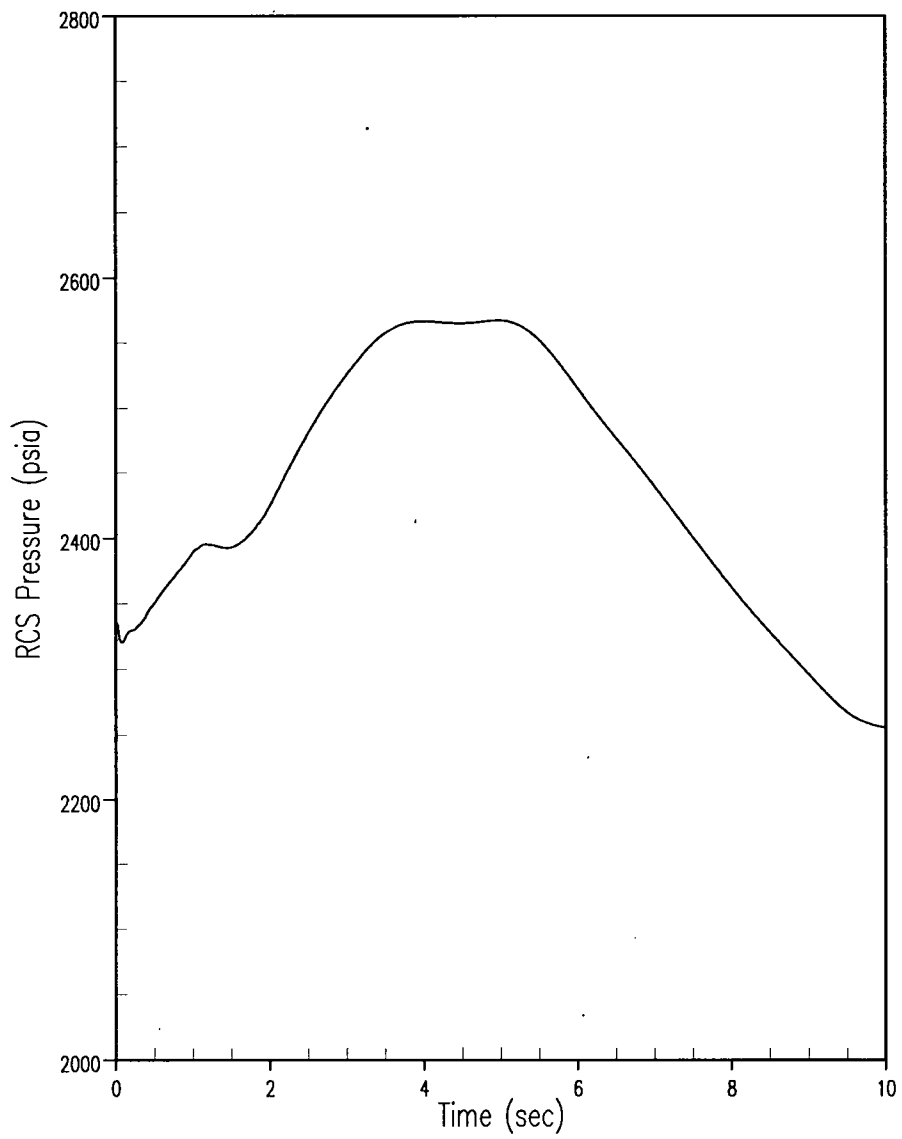


Figure 2.8.5.3.2-3 RCP Locked Rotor/Shaft Break Overpressurization/Peak Clad Temperature – RCS Pressure Versus Time

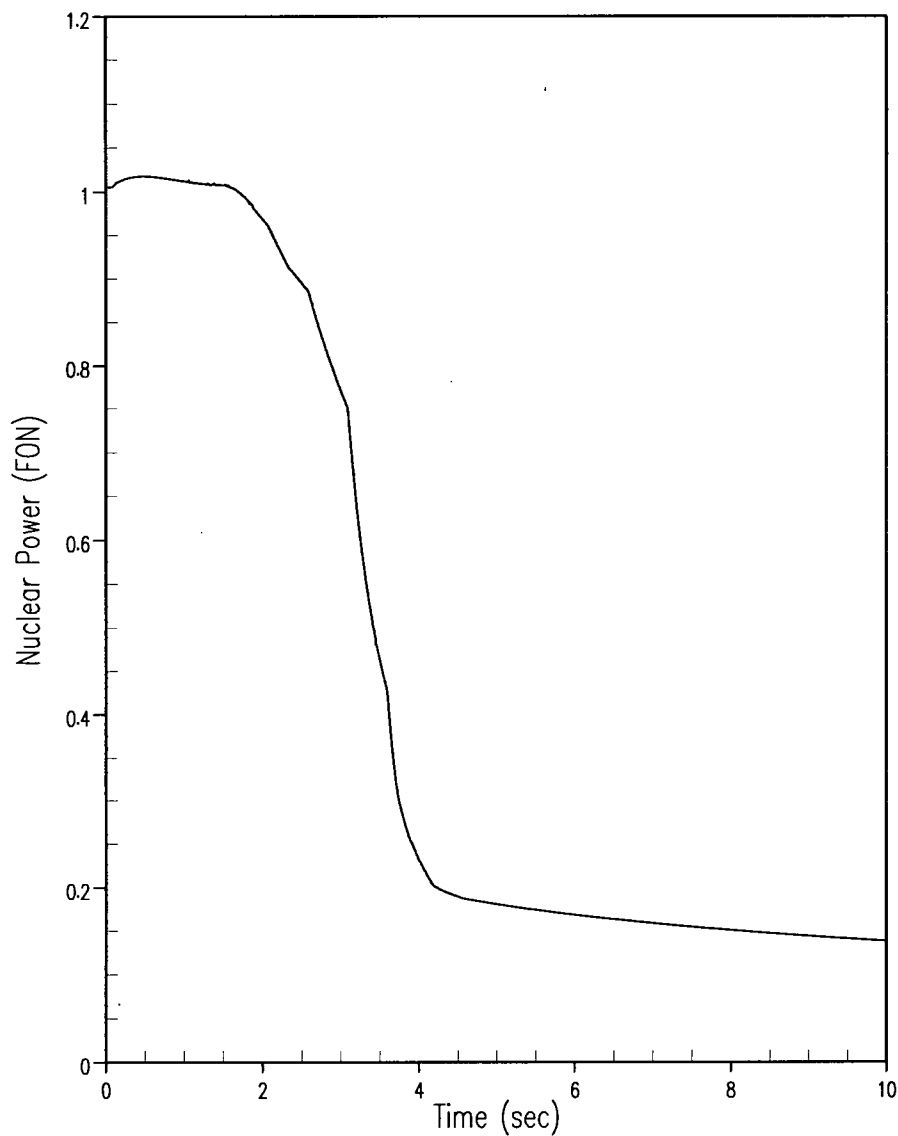


Figure 2.8.5.3.2-4 RCP Locked Rotor/Shaft Break Overpressurization/Peak Clad Temperature – Nuclear Power Versus Time

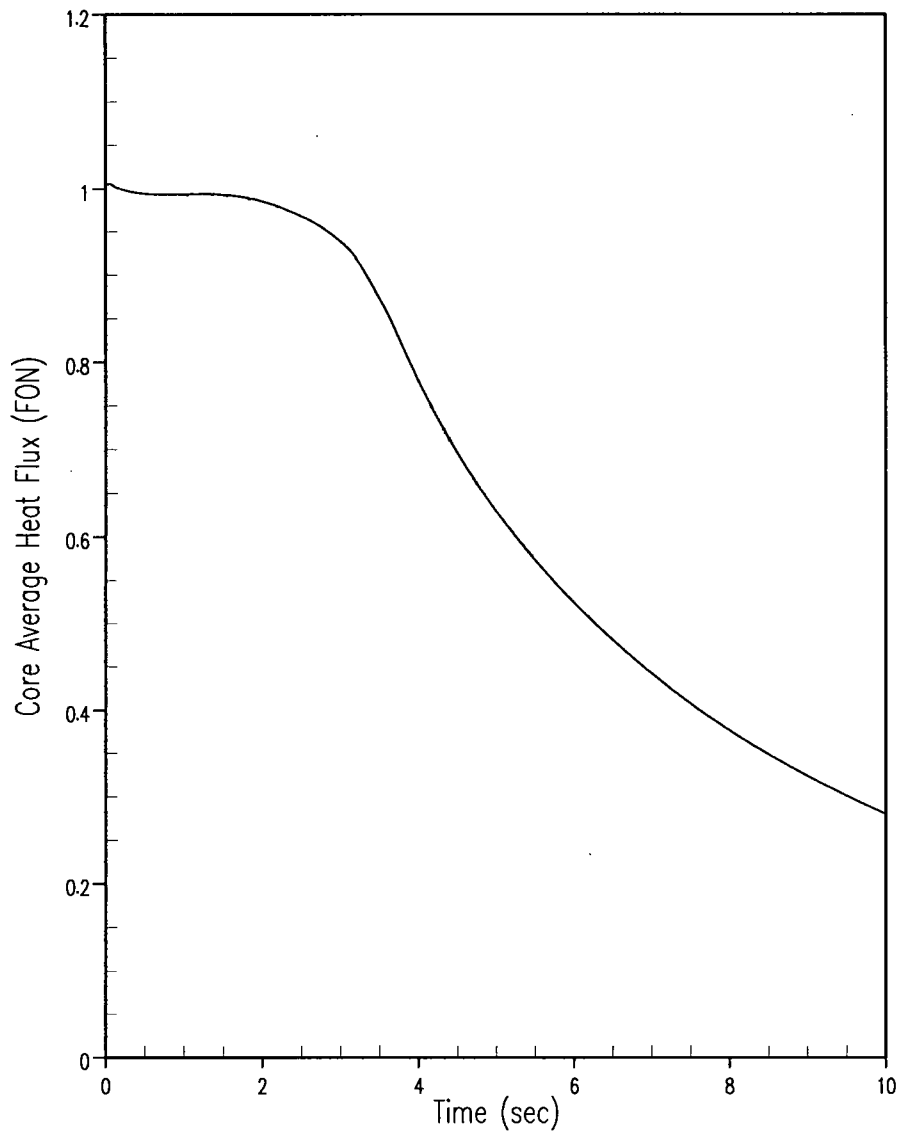


Figure 2.8.5.3.2-5 RCP Locked Rotor/Shaft Break Overpressurization/Peak Clad Temperature – Core Average Heat Flux Versus Time

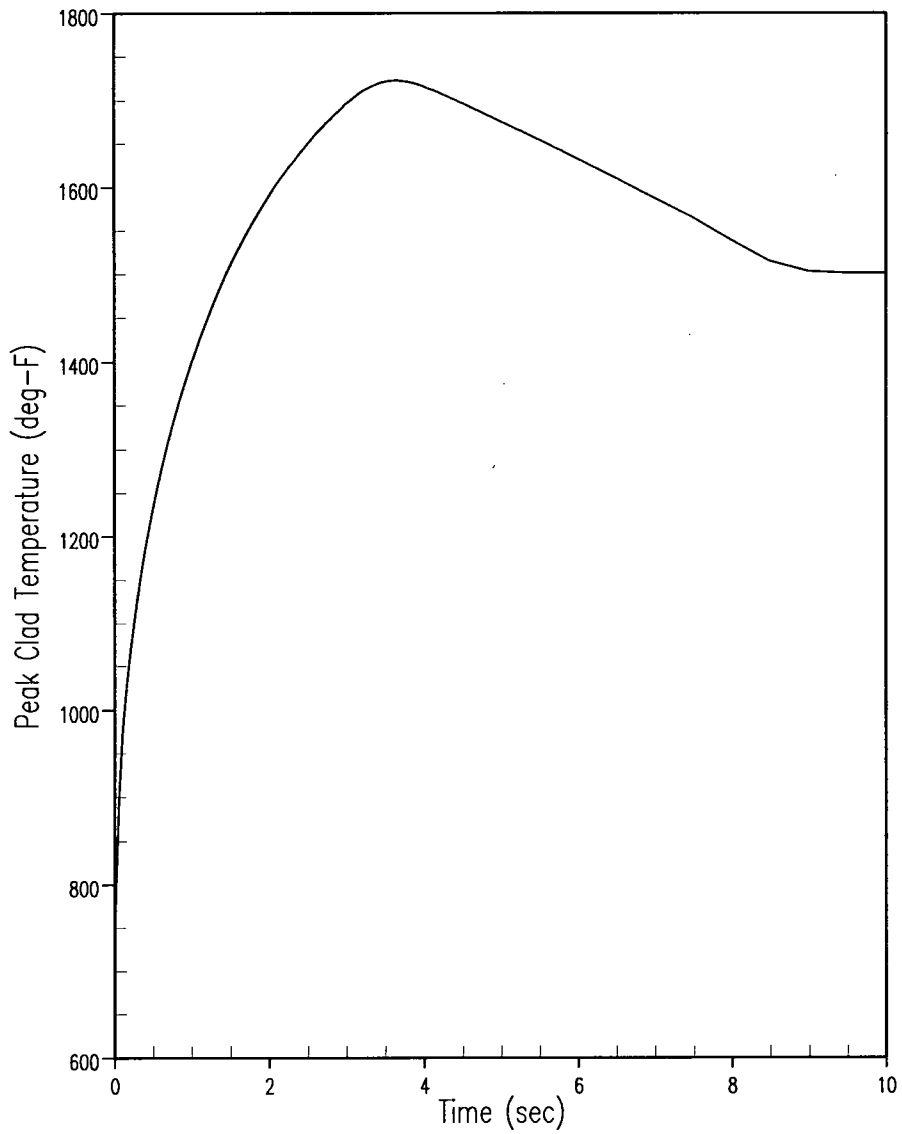


Figure 2.8.5.3.2-6 RCP Locked Rotor/Shaft Break Overpressurization/Peak Clad Temperature – Clad Inner Temperature Versus Time

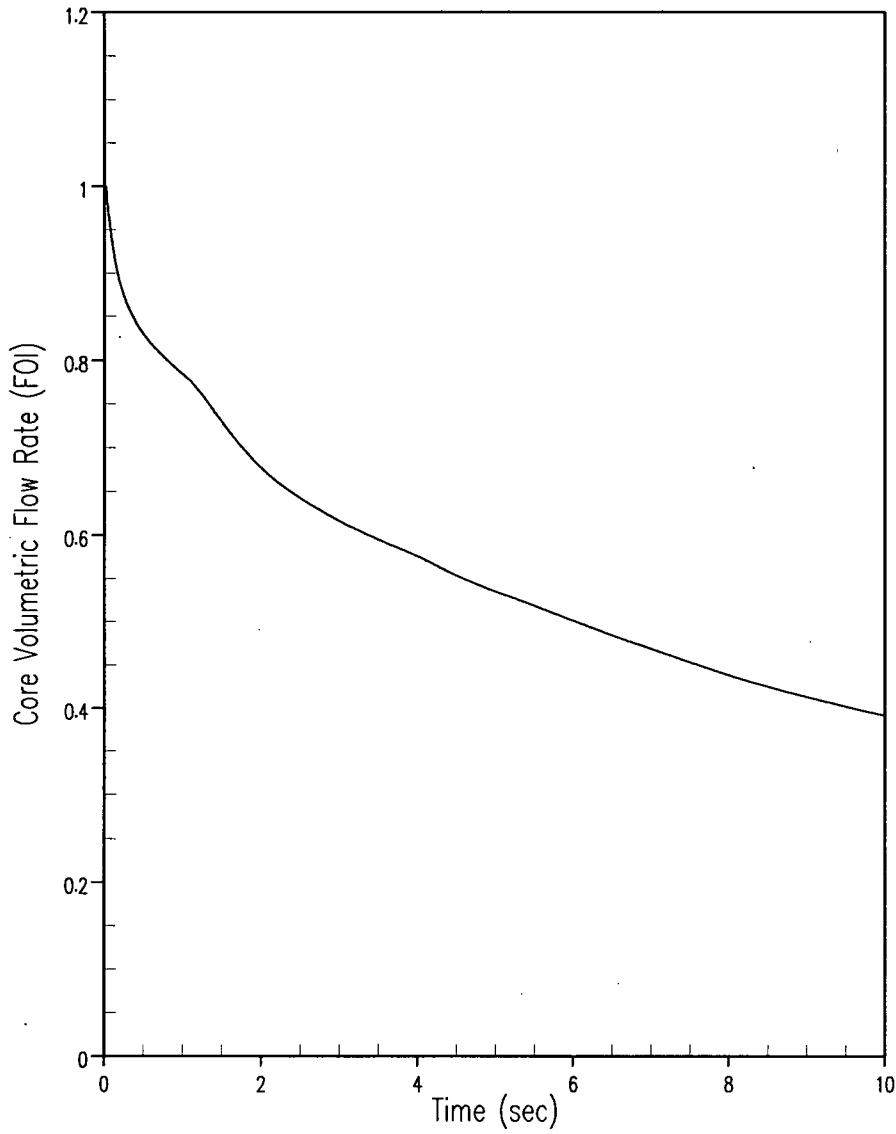


Figure 2.8.5.3.2-7 RCP Locked Rotor/Shaft Break Rods-in-DNB – Core Volumetric Flow Rate Versus Time

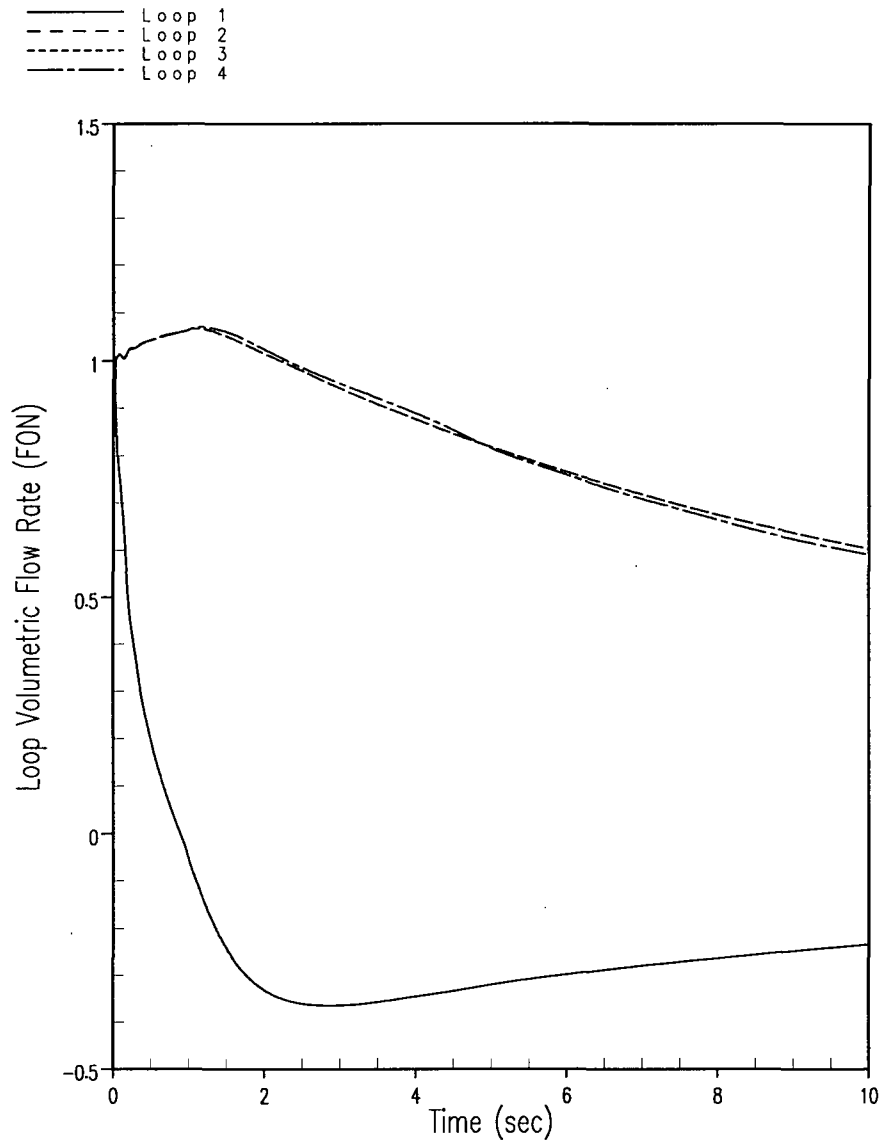


Figure 2.8.5.3.2-8 RCP Locked Rotor/Shaft Break Rods-in-DNB – Loop Volumetric Flow Rate Versus Time

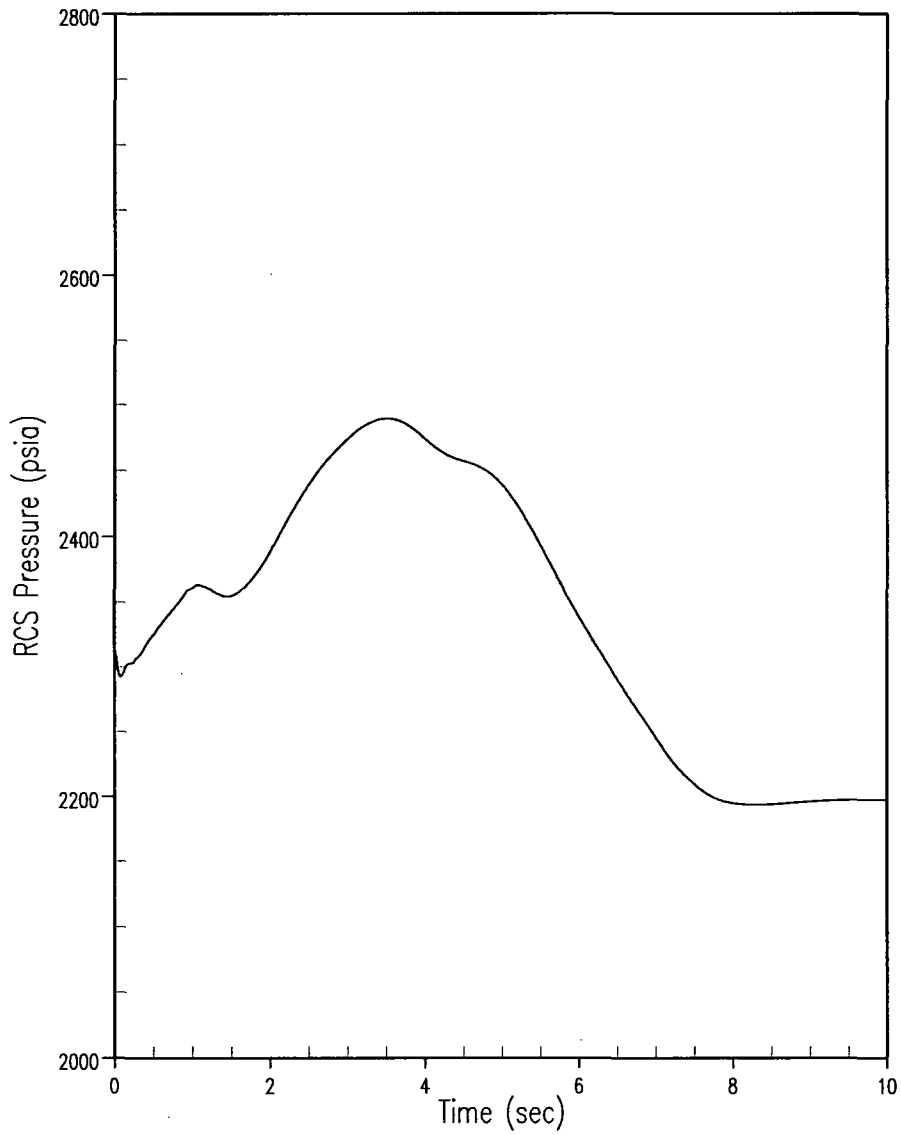


Figure 2.8.5.3.2-9 RCP Locked Rotor/Shaft Break Rods-in-DNB – RCS Pressure Versus Time

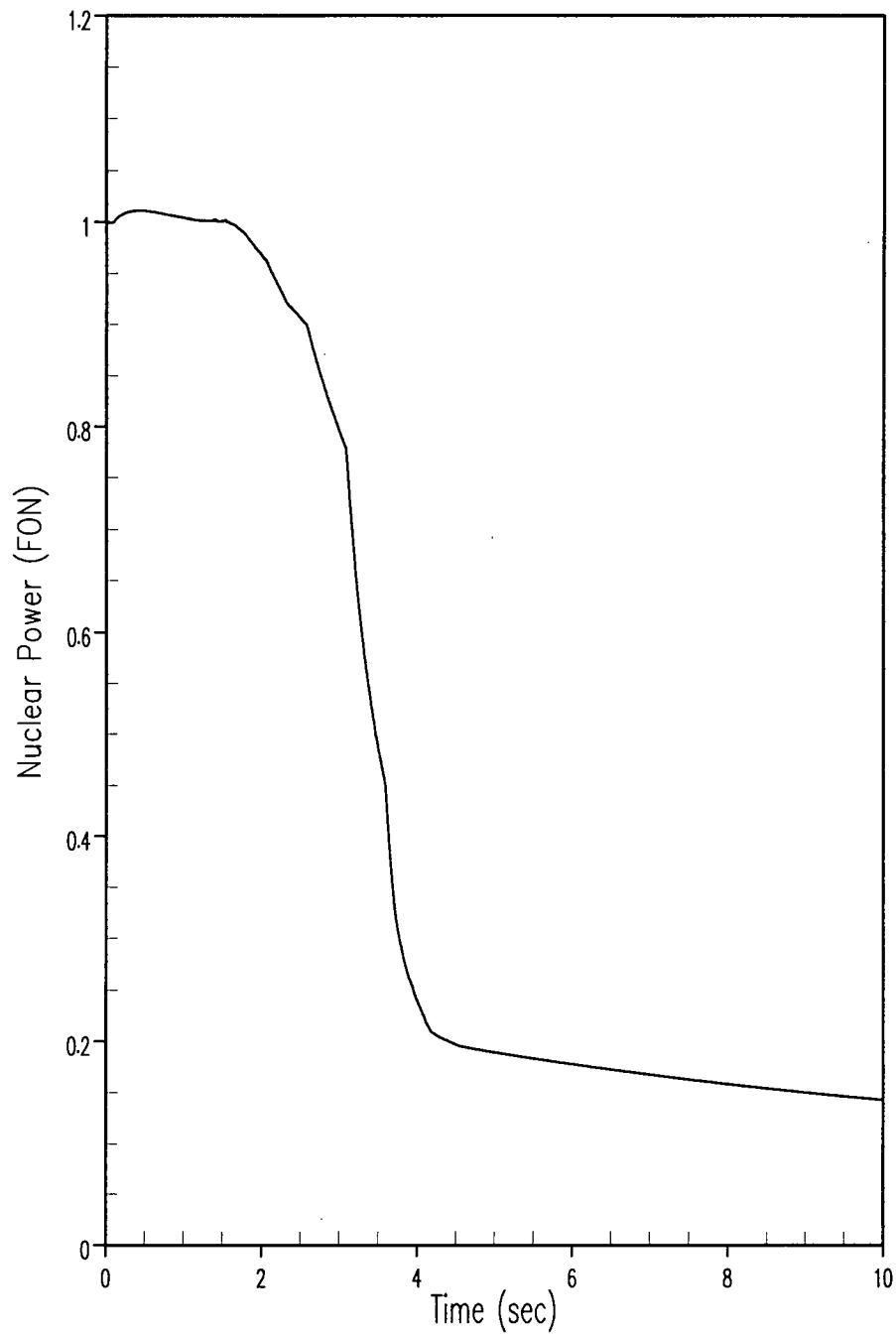


Figure 2.8.5.3.2-10 RCP Locked Rotor/Shaft Break Rods-in-DNB – Nuclear Power Versus Time

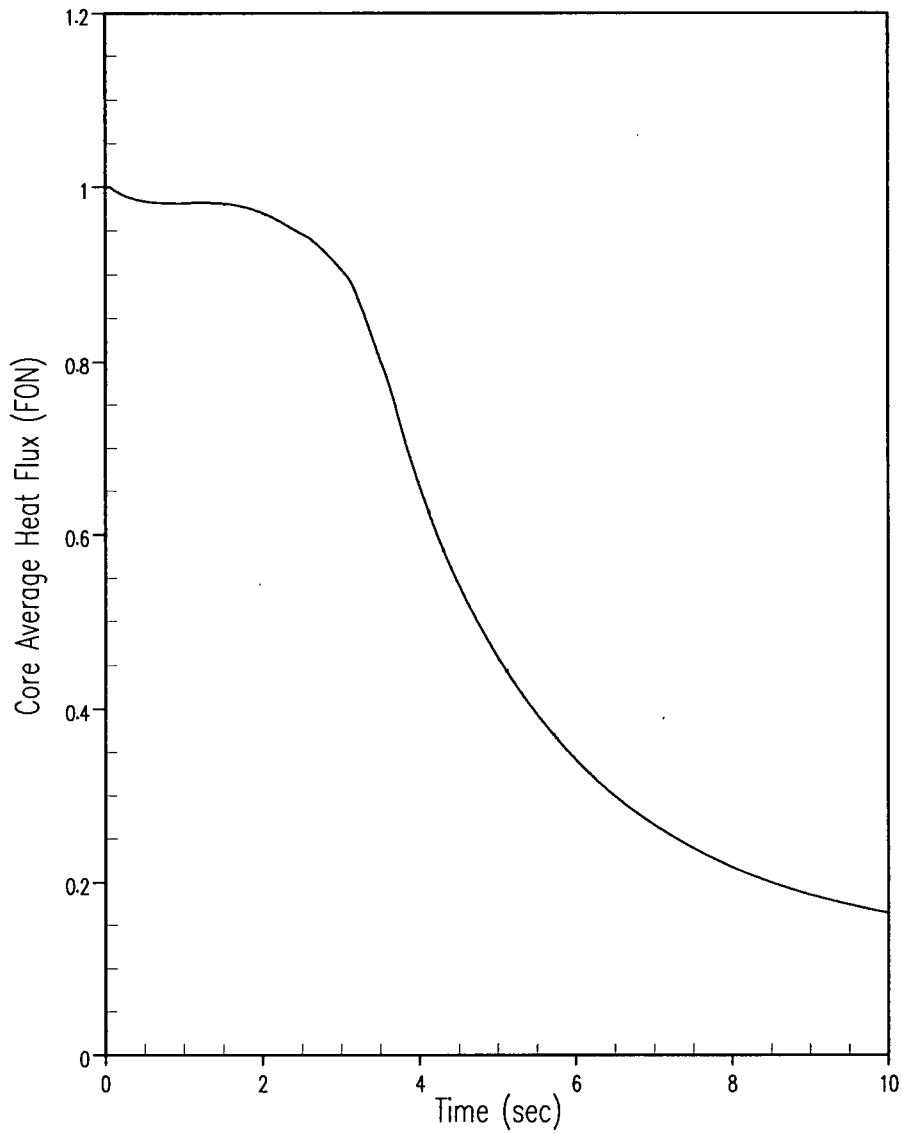


Figure 2.8.5.3.2-11 RCP Locked Rotor/Shaft Break Rods-in-DNB – Core Average Heat Flux Versus Time

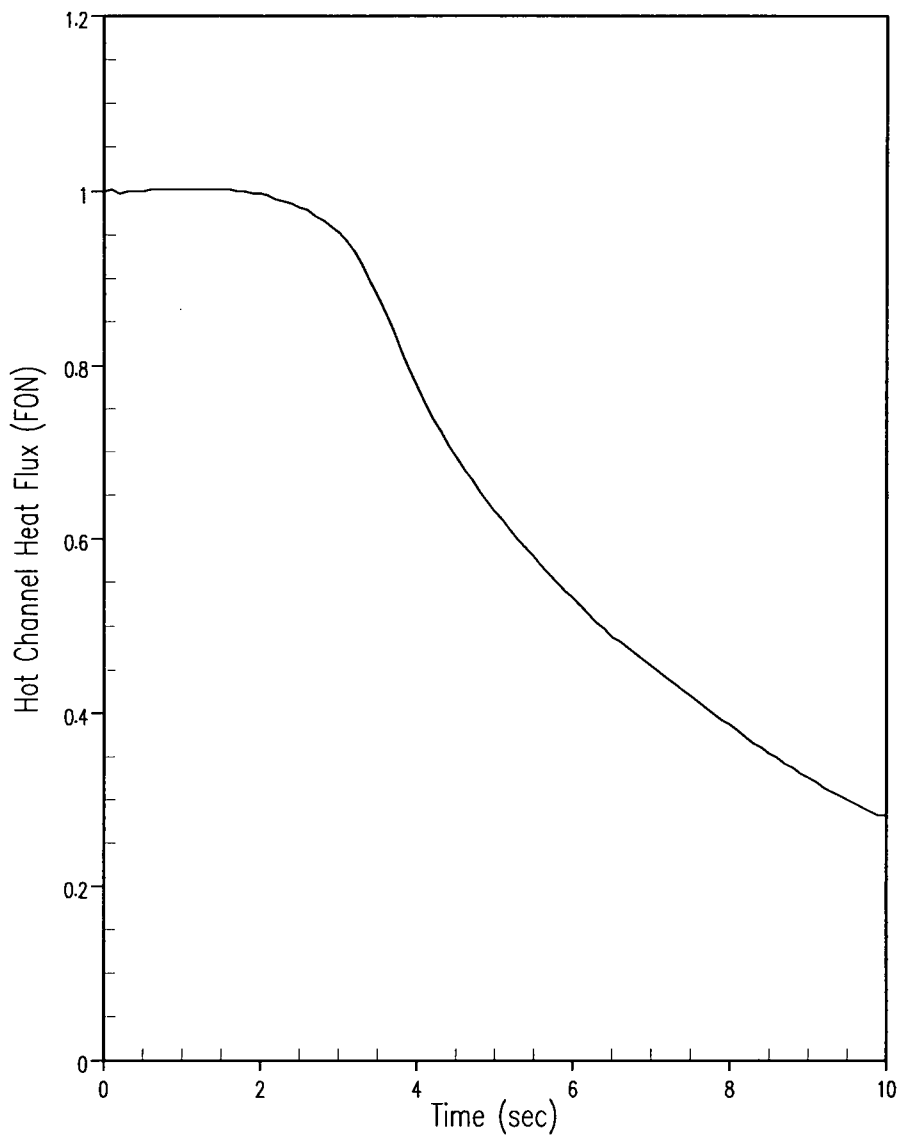


Figure 2.8.5.3.2-12 RCP Locked Rotor/Shaft Break Rods-in-DNB – Hot Channel Heat Flux Versus Time

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition

2.8.5.4.1.1 Regulatory Evaluation

An uncontrolled rod cluster control assembly (RCCA) withdrawal from subcritical or low-power startup conditions can be caused by a malfunction of the reactor control system or rod control system. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. Luminant Power's review covered:

- The description of the causes of the transient and the transient itself
- The initial conditions
- The values of reactor parameters used in the analysis
- The analytical methods and computer codes used
- The results of the transient analyses

The acceptance criteria were based on:

- General Design Criteria (GDC)-10, insofar as it requires that the reactor coolant system is designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences.
- GDC-20, insofar as it requires that the reactor protection system is designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.
- GDC-25, insofar as it requires that the protection system is designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I) or any transient conditions arising from occurrences of moderate frequency (Condition II).
2. To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.
3. To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and design evaluation of reactor components, including the fuel, reactor vessel internals, and reactivity control systems. Details of the control and protection systems instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the FSAR Chapter 15 accident analyses, which show that the acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-20, Protection System Functions, is described in FSAR Section 3.1.3.1

A fully automatic protection system, with appropriate redundant channels, is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 and IEEE Standard 379-1972. The reactor trip system automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the control rod drive mechanisms (CRDMs) of all of the full-length RCCAs. This causes the rods to insert by the force of gravity, which rapidly reduces reactor power output. The response and adequacy of the protection system have been verified by analysis of expected transients.

The engineered safety features (ESF) actuation system automatically initiates emergency core cooling and other safeguards functions by sensing accident conditions using redundant analog channels measuring diverse variables. In addition, manual

actuation of safeguards equipment can be performed where ample time is available for operator action. In either case, the ESF actuation system automatically trips the reactor on manual or automatic safety injection signal (SIS) generation.

- GDC-25, Protections System Requirements for Reactivity Control Malfunctions, is described in FSAR Section 3.1.3.6.

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by full-length rod insertion is completely independent of the normal control function since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. In the postulated accidental withdrawal (assumed to be initiated by a control malfunction), flux, temperature, pressure, level and flow signals are generated independently. Any of these signals (trip demands) can cause the breakers to trip the reactor.

FSAR Chapter 15 discusses analyses of the effects of possible malfunctions. These analyses show that for postulated dilution during refueling, cold shutdown, hot shutdown, hot standby, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of the dilution and to initiate boration before the shutdown margin is lost. The analyses also show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

FSAR Section 15.4 states that the uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition is an American Nuclear Society (ANS) Condition II event. This transient could be caused by a malfunction of the reactor control or rod control systems. FSAR Section 15.4.1.3 concludes that, in the event of an uncontrolled RCCA withdrawal from subcritical conditions, the core and the reactor control system are not adversely affected since the combination of thermal power and the coolant temperature result in a departure from nucleate boiling ratio (DNBR) greater than the limit value. The DNBR design basis is described in FSAR Section 4.4. All applicable acceptance criteria have been met.

2.8.5.4.1.2 Technical Evaluation

2.8.5.4.1.2.1 Introduction

An uncontrolled RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of RCCAs resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the reactor control rod drive system. This could occur with the reactor either subcritical or at power. The “at power” occurrence is discussed in Licensing Report (LR) subsection 2.8.5.4.2. The uncontrolled RCCA withdrawal from a subcritical condition is classified as a Condition II event, a fault of moderate frequency, as defined by the American Nuclear Society’s “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants,” ANSI N18.2-1973.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low-power level during startup by RCCA withdrawal or by reducing the core boron concentration. RCCA motion can cause much faster changes in reactivity than can result from changing boron concentration.

The rods are physically prevented from withdrawing in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The CRDM is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming the simultaneous withdrawal of the combination of the two rod banks with the maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the initial power increase results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup transient since it limits the power to an acceptable level prior to protection system action. After the initial power increase, the nuclear power is momentarily reduced and then, if the incident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Should a continuous RCCA withdrawal be initiated, the transient will be terminated by one of the following automatic protective functions:

- Source range neutron flux reactor trip – actuated when either of two independent source range channels indicates a flux level above a pre-selected, manually adjustable setpoint. This trip function may be manually bypassed only after an intermediate range neutron flux channels indicates a flux level above the source range cutoff power level. It is automatically reinstated when both intermediate channels indicate a flux level below the source range cutoff power level.
- Intermediate range neutron flux reactor trip – actuated when either of two independent intermediate range channels indicates a flux level above a pre-selected, manually adjustable setpoint. This trip function may be manually bypassed when two of the four power range channels are reading above approximately 10 percent of full power and is automatically reinstated when three of the four channels indicate a power level below this value.
- Power range neutron flux reactor trip (low setting) – actuated when two of the four power range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10 percent of full power. This trip function is automatically reinstated when three of the four channels indicate a power level below 10-percent power.

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- Power range neutron flux reactor trip (high setting) – actuated when two out of the four power range channels indicate a power level above approximately 110.8 percent of full power. This trip function is always active.
 - High nuclear flux rate reactor trip – actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicates a rate above the preset setpoint of approximately 6.3 percent in 2 seconds. This trip function is always active.

In addition, control rod stops on high intermediate range flux level (one out of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively. This analysis credits the power range neutron flux trip (low setting) to initiate the reactor trip.

2.8.5.4.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The accident analysis uses the Standard Thermal Design Procedure (STDP) methodology since the conditions resulting from the transient are outside the range of applicability of the Revised Thermal Design Procedure (RTDP) methodology. To obtain conservative results for the analysis of the uncontrolled RCCA bank withdrawal from subcritical event, the following input parameters and initial conditions are modeled:

1. The magnitude of the nuclear power peak reached during the initial part of the transient, for any given reactivity insertion rate, is strongly dependent on the Doppler-only power defect. Therefore, a conservatively low absolute value is used (1,000 pcm) to maximize the nuclear power transient.
2. A most-positive moderator temperature coefficient (+5 pcm/°F) is used since this yields the maximum rate of power increase. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and moderator is much longer than the nuclear flux response time constant. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient.
3. The analysis assumes the reactor to be at hot-zero-power conditions with a nominal no-load temperature of 557°F. This assumption is more conservative than that of a lower initial system temperature (that is, shutdown conditions). The higher initial system temperature yields a larger fuel-to-moderator heat transfer coefficient, a larger specific heat of the moderator and fuel, and a less-negative (smaller absolute magnitude) Doppler defect. The less-negative Doppler defect reduces the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel specific heat and larger heat transfer coefficient yields a larger peak heat flux.
4. The analysis assumes the initial effective multiplication factor (k_{eff}) to be 1.0 since it maximizes the peak neutron flux and results in the most severe nuclear power transient.

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5. Reactor trip is assumed on power range high neutron flux (low setting). A conservative combination of instrumentation error, setpoint error, delay for trip signal actuation, and delay for control rod assembly release is modeled. The analysis assumes a 10-percent uncertainty in the power range flux trip setpoint (low setting), raising it from the nominal value of 25 percent of full power to 35 percent of full power. A delay time of 0.5 seconds is assumed for trip signal actuation and control rod assembly release. No credit is taken for the source range or intermediate range protection. During the transient, the rise in nuclear power is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods release is negligible. In addition, the total reactor trip reactivity is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position.
 6. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at the maximum rod withdrawal speed. The assumed reactivity insertion rate is 75 pcm/sec, which is based on a rod worth of 100 pcm/inch and a maximum rod speed of 72 steps per minute.
 7. The departure from nucleate boiling (DNB) analysis assumes the most limiting axial and radial power shapes possible during the fuel cycle associated with having the two highest combined worth banks in their highest worth position.
 8. The analysis assumes the initial power level to be below the power level expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.
 9. The analysis assumes two of the four reactor coolant pumps to be in operation. This is conservative with respect to the DNB transient.
 10. This accident analysis uses the STDP methodology. The use of the STDP stipulates that the RCS flow rates will be based on a fraction of the thermal design flow for two pumps operating. Since the event is analyzed from hot-zero power, the steady-state non-RTDP uncertainties are not considered in defining the initial conditions.

The uncontrolled RCCA bank withdrawal from subcritical event is considered an ANS Condition II event, a fault of moderate frequency, and is analyzed to show that the core and reactor coolant system are not adversely affected by the event. This is demonstrated by showing that the DNB design basis is not violated and subsequently that there is little likelihood of core damage. It must also be shown that the peak hot spot fuel centerline temperature remains within the acceptable limit (4,800°F), although for this event, the heatup is relatively non-limiting.

2.8.5.4.1.2.3 Description of Analyses and Evaluations

The analysis of the uncontrolled RCCA bank withdrawal from subcritical conditions is performed in three stages. First, a spatial neutron kinetics computer code, TWINKLE (Reference 1), is used to calculate the core average nuclear power transient, including the various core feedback

effects, that is, Doppler and moderator reactivity. Next, the FACTRAN computer code (Reference 2) uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the core average heat flux and hot spot fuel temperature transients. Finally, the core average heat flux calculated by FACTRAN is used in the VIPRE computer code (Reference 3) for transient DNBR calculations.

2.8.5.4.1.2.4 Results

The analysis shows that all applicable acceptance criteria are met for CPNPP Units 1 and 2. The minimum DNBR never goes below the limit value and the peak fuel centerline temperature is 2,304°F. The peak temperatures are well below the minimum temperature where fuel melting would be expected (4,800°F).

Figure 2.8.5.4.1-1 shows the nuclear power transient, Figure 2.8.5.4.1-2 shows the core average heat flux transient, and Figure 2.8.5.4.1-3 shows the clad average, fuel centerline, and fuel average temperature transients at the hot spot.

The time sequence of events for both cases is presented in Table 2.8.5.4.1-1.

In the event of an RCCA withdrawal event from subcritical conditions, the core and the reactor coolant system are not adversely affected since the combination of thermal power and coolant temperature results in a minimum DNBR greater than the safety analysis limit value.

Furthermore, since the maximum fuel temperatures predicted to occur during this event are much less than those required for fuel melting to occur, no fuel damage is predicted as a result of this transient. Cladding damage is also precluded.

2.8.5.4.1.3 Conclusions

Luminant Power has reviewed the analysis of the uncontrolled RCCA withdrawal from a subcritical or low-power startup condition and concludes that the analysis has adequately accounted for the changes in core design necessary for plant operation at the proposed uprated power level. It is also concluded that the analyses were performed using acceptable analytical models. Luminant Power further concludes that the analysis has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits are not exceeded. Based on this, it is concluded that the plant will continue to meet the requirements of GDCs -10, -20, and -25 following implementation of the uprating. Therefore, Luminant Power finds the uprating acceptable with respect to the uncontrolled RCCA withdrawal from a subcritical or low-power startup condition.

2.8.5.4.1.4 References

1. WCAP-7979, "TWINKLE - A Multi-dimensional Neutron Kinetics Computer Code," January 1975
2. WCAP-7908, "FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO₂ Fuel Rod," December 1989.
3. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

Table 2.8.5.4.1-1

Time Sequence of Events – Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

Event	Time (seconds)
Initiation of Uncontrolled Rod Withdrawal	0.0
Power Range High Neutron Flux Low Setpoint is Reached	9.82
Peak Nuclear Power Occurs	9.95
Rod Motion Begins	10.32
Peak Heat Flux Occurs (0.3336 (fraction of nominal))	12.075
Minimum DNBR Occurs (1.616)	12.075
Peak Average Cladding Temperature Occurs (680°F)	12.60
Peak Average Fuel Temperature Occurs (1,884°F)	12.75

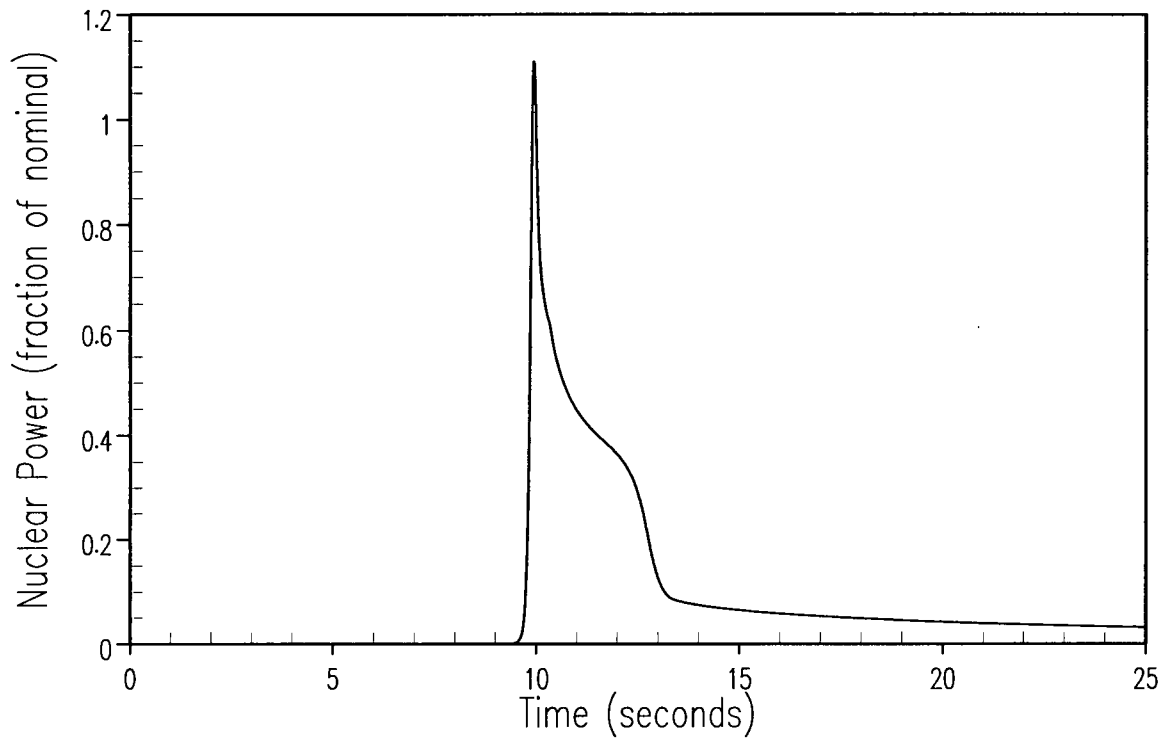


Figure 2.8.5.4.1-1 Rod Withdrawal from Subcritical – Nuclear Power Versus Time

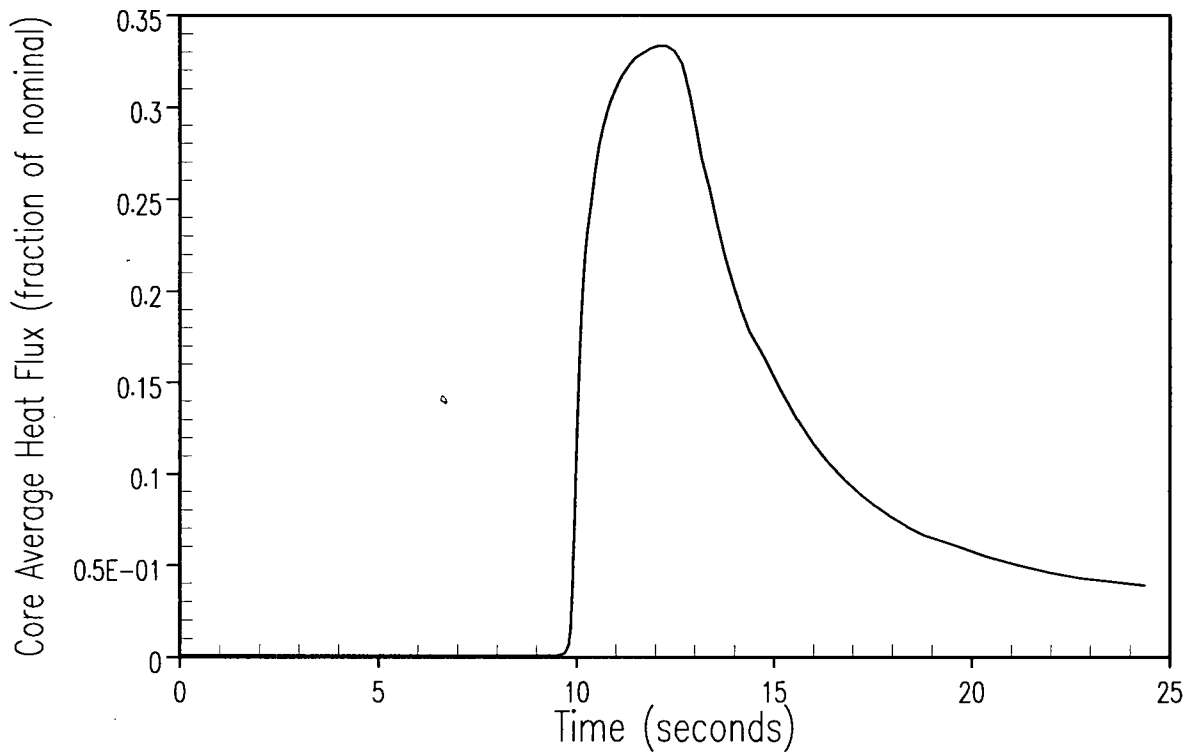


Figure 2.8.5.4.1-2 Rod Withdrawal from Subcritical – Core Average Heat Flux Versus Time

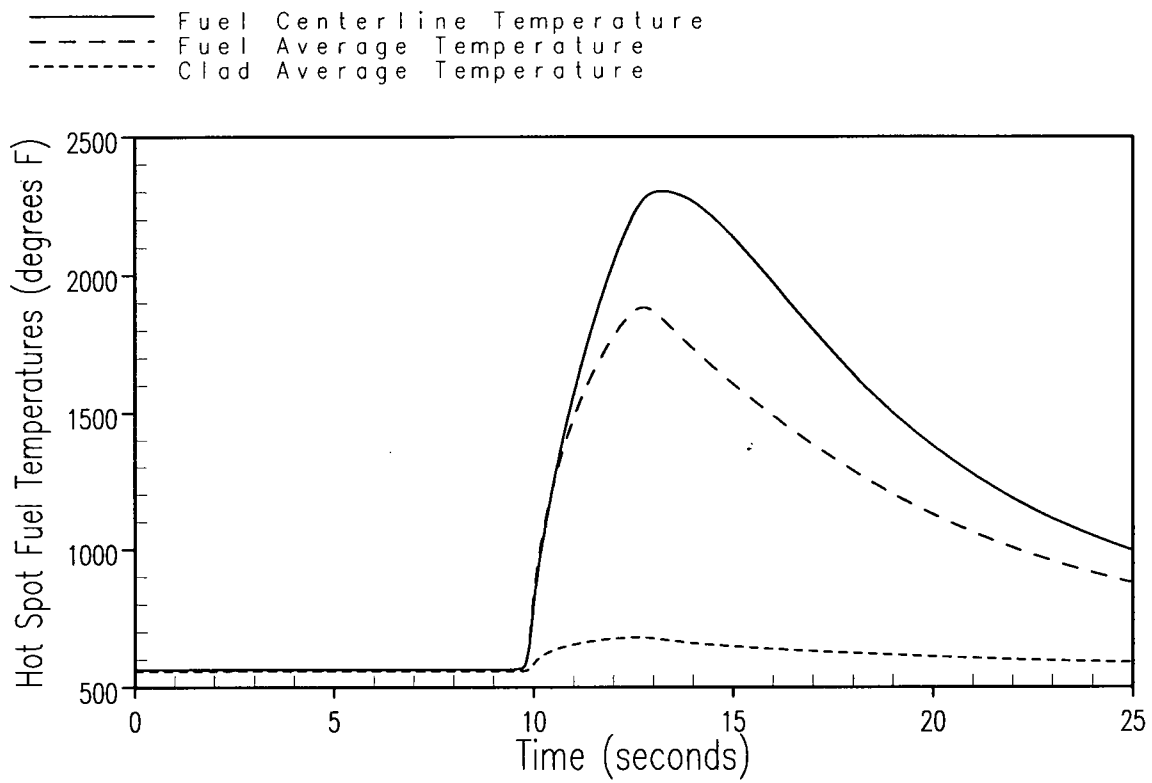


Figure 2.8.5.4.1-3 Rod Withdrawal from Subcritical – Hot Spot Fuel Temperatures Versus Time

2.8.5.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

2.8.5.4.2.1 Regulatory Evaluation

An uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power can be caused by a malfunction of the rod control system. This bank withdrawal will add positive reactivity to the reactor core, resulting in a power excursion.

The review covered:

- The description of the causes of the anticipated operational occurrence (AOO) and the description of the event itself
- The initial conditions
- The values of reactor parameters used in the analyses
- The analytical methods and computer codes used
- The results of the associated analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including AOOs.
- GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs.
- GDC-25, insofar as it requires that the protection system be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Current Licensing Basis

As noted in the Comanche Peak Nuclear Power Plant (CPNPP) Final Safety Analysis Report (FSAR) Section 3.1, the design bases of CPNPP are measured against the Nuclear Regulatory Commission (NRC) GDC for Nuclear Power Plants, Appendix A to 10 CFR 50. The adequacy of the CPNPP design relative to the GDC is discussed FSAR Sections 3.1.1 and 3.1.2.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:

1. Ensure that fuel damage is not expected during normal core operation and operational transients (Condition I) or any transient conditions arising from occurrences of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These failures are within the capability of the plant cleanup system, and are consistent with plant design bases.
2. Ensure return of the reactor to a safe state following infrequent incident (Condition III) events with only a small fraction of fuel rods damaged, although sufficient fuel damage might occur to preclude immediate resumption of operation.
3. Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

Note that the term "fuel damage" as used in Item 1 above is defined as penetration of the fission product barrier (i.e., the fuel rod clad). Also note that American National Standard Institute (ANSI) N18.2-1973 expands the definitions of the four conditions enumerated in Items 1 through 3 above.

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components. FSAR Chapter 7 provides the details of the control and protections systems instrumentation design and logic. This information supports the FSAR Chapter 15 accident analysis, which shows that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-20, Protection System Functions, is described in FSAR Section 3.1.3.1.

A fully automatic protection system, with appropriate redundant channels, is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is Institute of Electrical and Electronic Engineering (IEEE) Standard 279-1971 and IEEE Standard 379-1972. The reactor protection system automatically initiates a reactor trip when any variable exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the rod drive mechanisms of all of the full-length RCCAs. This causes the rods to insert by gravity, which rapidly reduces

reactor power output. The response and adequacy of the protection system have been verified by analysis of expected transients.

The engineered safety features (ESF) actuation system automatically initiates emergency core cooling, and other safeguards functions, by sensing accident conditions using redundant analog channels measuring diverse variables. Manual actuation of safeguards equipment may be performed where ample time is available for operator action. The ESF actuation system automatically trips the reactor on manual or automatic safety injection signal (SIS) generation.

- GDC-25, Protections System Requirements for Reactivity Control Malfunctions, is described in FSAR Section 3.1.3.6.

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by full-length rod insertion is completely independent of the normal control function, since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Therefore, in the postulated accidental withdrawal (assumed to be initiated by a control malfunction), flux, temperature, pressure, level and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

FSAR Chapter 15 discusses analyses of the effects of possible malfunctions. These analyses show that for postulated dilution during refueling, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution, and initiate boration before the shutdown margin is lost. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

FSAR Section 15.4.2.1 states that uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in departure from nucleate boiling (DNB). Therefore, in order to avert damage to the fuel cladding, the reactor protection system is designed to terminate any such transient before the DNB ratio (DNBR) falls below the safety analysis limit value. This event is classified as an American Nuclear Society (ANS) Condition II incident.

FSAR Section 15.4.2.2 states that the transient is analyzed with the RETRAN Code. It computes pertinent plant variables including temperatures, pressures, and power level.

FSAR Section 15.4.2.3 concludes that the high neutron flux and overtemperature N-16 trip channels provide adequate protection over the entire range of possible reactivity insertion rates; that is, the minimum value of DNBR is always larger than the safety analysis limit value. It is assumed that the high pressurizer water level reactor trip would prevent pressurizer filling. In addition, the positive flux rate and high pressurizer pressure reactor trip functions provide a

timely reactor trip to preclude RCS overpressurization in instances where the high neutron flux or overtemperature N-16 trips occur too late to provide the necessary protection.

2.8.5.4.2.2 Technical Evaluation

2.8.5.4.2.2.1 Introduction

An uncontrolled RCCA bank withdrawal at power that causes an increase in core heat flux can result from faulty operator action or a malfunction in the rod control system. Immediately following the initiation of the accident, the steam generator heat removal rate lags behind the core power generation rate until the steam generator pressure reaches the setpoint of the steam generator relief or safety valves. This imbalance between heat removal and heat generation rate causes the reactor coolant temperature to rise. Unless terminated, the power mismatch and resultant coolant temperature rise could eventually result in a violation of the DNBR safety analysis limit and/or fuel centerline melt. Therefore, to avoid core damage, the reactor protection system is designed to automatically terminate any such transient before the DNBR falls below the safety analysis limit value, or the fuel rod linear heat generation rate (kW/ft) limit is exceeded.

The automatic features of the reactor protection system that prevent core damage in an RCCA bank withdrawal incident at power include the following:

- Power range high neutron flux instrumentation actuates a reactor trip on neutron flux if two-out-of-four channels exceed an overpower setpoint.
- Reactor trip actuates if any two-out-of-four channels exceed the high positive neutron flux rate setpoint.
- Reactor trip actuates if any two-out-of-four N-16 channels exceed an overtemperature N-16 setpoint. This setpoint is automatically varied with axial power distribution, coolant average temperature, and coolant average pressure to protect against violating the DNBR safety analysis limit.
- Reactor trip actuates if any two-out-of-four N-16 channels exceed an overpower N-16 setpoint.
- Main steam safety valves (MSSVs) can open for this event and provide an additional heat sink.
- A high pressurizer pressure reactor trip actuated from any two-out-of-four pressure channels which is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.
- A high pressurizer water level reactor trip actuated from any two-out-of-three channels which is set at a fixed point, when the reactor power is above approximately 10 percent (Permissive 7).

2.8.5.4.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

A number of cases were analyzed assuming a range of reactivity insertion rates for both minimum and maximum reactivity feedback conditions at various power levels. The cases presented below are representative for this event.

For an uncontrolled RCCA bank withdrawal at power accident, the analysis assumed the following conservative assumptions:

- This accident was analyzed with the Revised Thermal Design Procedure (RTDP) (Reference 1). Initial reactor power, RCS pressure, and RCS temperature were assumed to be at their nominal values. Minimum measured flow was modeled. Uncertainties in initial conditions were included in the DNBR safety analysis limit as described in the RTDP.
- For reactivity coefficients, two cases were analyzed.
 - Minimum reactivity feedback; A least negative or positive value of the moderator temperature coefficient of reactivity is assumed corresponding to the beginning of core life. A conservatively small (in absolute magnitude) value of the Doppler coefficient is assumed.
 - Maximum reactivity feedback; A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler coefficient are assumed.
- The reactor trip on high neutron flux was assumed to be actuated at a conservative value of 118.0 percent of nominal full power. The N-16 trips included all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation were assumed at their maximum values.
- The RCCA trip insertion characteristic was based on the assumption that the highest-worth RCCA was stuck in its fully withdrawn position.
- A range of reactivity insertion rates was examined. The maximum-positive reactivity insertion rate was greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at a conservative speed (45 inches/minute, which corresponds to 72 steps/minute).
- To be conservative with respect to DNB, the pressurizer sprays and relief valves were assumed operational since they limit the reactor coolant pressure increase.
- Power levels of 10, 60, and 100 percent of the nuclear steam supply system (NSSS) power of 3,628 MWt were considered.

Based on its frequency of occurrence, the uncontrolled RCCA bank withdrawal at-power accident is considered a Condition II event as defined by the ANS. The following items summarize the main acceptance criteria associated with this event:

- The critical heat flux should not be exceeded. This is met by demonstrating that the minimum DNBR does not go below the safety analysis limit value at any time during the transient.
- Pressure in the RCS and main steam system (MSS) should be maintained below 110 percent of the design pressures.

The protection features presented in Licensing Report (LR) subsection 2.8.5.4.2.2.1 provide mitigation of the uncontrolled RCCA bank withdrawal at-power transient such that the above criteria are satisfied.

Also, a conservative generic evaluation that is applicable to CPNPP has shown that the positive flux rate and high pressurizer pressure functions provide a timely reactor trip that precludes RCS overpressurization in instances where the power range high neutron flux or the overtemperature N-16 trip occur too late to provide the necessary protection. This evaluation confirms that the RCS pressure limit is met.

2.8.5.4.2.2.3 Description of Analyses and Evaluations

The purpose of this analysis was to demonstrate the manner in which the protection functions described above actuate for various combinations of reactivity insertion rates and initial conditions. Insertion rate and initial conditions determined which trip function actuated first.

The uncontrolled RCCA bank withdrawal at-power event was analyzed with the RETRAN computer code (Reference 2). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and MSSVs. The code computes pertinent plant variables including temperatures, pressures, power level, and the DNBR (based on a conservative partial derivative approximation of the DNB core limit lines).

2.8.5.4.2.2.4 Results

Figures 2.8.5.4.2-1 through 2.8.5.4.2-3 (Unit 1) and Figures 2.8.5.4.2-10 through 2.8.5.4.2-12 (Unit 2) show the transient response for a rapid uncontrolled RCCA bank withdrawal incident (110 pcm/sec) starting from 100 percent power with minimum reactivity feedback. The neutron flux level in the core rises rapidly while the core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Reactor trip on high neutron flux occurs shortly after the start of the accident prior to a significant increase in the heat flux and water temperature with resultant minimum DNB ratios that remain well above the safety analysis limit value throughout the transient.

The transient response for a slow uncontrolled RCCA bank withdrawal (1 pcm/sec) from 100 percent power with minimum feedback is shown in Figures 2.8.5.4.2-4 through 2.8.5.4.2-6

(Unit 1) and Figures 2.8.4.2-13 through 2.8.4.2-15 (Unit 2). With a lower insertion rate the power increase rate is slower, the rate of rise of the average coolant temperature is slower and the system lags and delays become less significant. A reactor trip on overtemperature N-16 occurs after a longer period of time than for a rapid RCCA bank withdrawal. Again, the minimum DNBR remain greater than the safety analysis limit value.

Figure 2.8.5.4.2-7 (Unit 1) and Figure 2.8.5.4.2-16 (Unit 2) show the minimum DNBR as a function of reactivity insertion rate from 100 percent power for both minimum and maximum reactivity feedback conditions. It can be seen that the high neutron flux and overtemperature N-16 reactor trip functions provided DNB protection over the analyzed range of reactivity insertion rates and the minimum DNBR is never less than the safety analysis limit value.

Figures 2.8.5.4.2-8, 2.8.5.4.2-9 (Unit 1), 2.8.5.4.2-17 and 2.8.5.4.2.18 (Unit 2) show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 60- and 10-percent power, respectively. The results are similar to the 100-percent power case. However, as the initial power level is decreased, the range over which the overtemperature N-16 trip is effective is increased.

A calculated sequence of events for two cases is shown in Table 2.8.5.4.2-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant could subsequently be cooled down further by following normal plant shutdown procedures. The limiting results of the uncontrolled RCCA bank withdrawal at power analysis are shown in Table 2.8.5.4.2-2.

The high neutron flux and overtemperature N-16 reactor trip functions provided adequate protection over the entire range of possible reactivity insertion rates. The results show that the DNB design basis is met and the peak kW/ft is less than the limit. The peak pressures in the RCS and MSS do not exceed 110 percent of their respective design pressures.

Therefore, the results of the analysis show that an uncontrolled RCCA bank withdrawal at-power does not adversely affect the core, the RCS, or the MSS.

2.8.5.4.2.3 Conclusions

This review of the uncontrolled RCCA bank withdrawal at-power event analysis demonstrates that TXU Power has adequately accounted for the changes in core design required for plant operation at the proposed uprated power level. This analysis was performed using acceptable analytical models. This analysis has also demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits are not exceeded. Based on this, it can be concluded that the plant will continue to meet the requirements of GDCs -10, -20, and -25 following implementation of the proposed uprated power level. Therefore, the uprated power level is acceptable with respect to the uncontrolled RCCA bank withdrawal at power event.

2.8.5.4.2.4 References

1. WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-Proprietary), "Revised Thermal Design Procedure," April 1989.
2. WCAP-14882-P-A (Proprietary), WCAP-15234-A (Non-Proprietary), "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April and May 1999 respectively.

Table 2.8.5.4.2-1			
Time Sequence of Events – Uncontrolled RCCA Bank Withdrawal at Power			
Case	Event	Time (sec)	
		Unit 1	Unit 2
100% Power, Minimum Feedback, Rapid RCCA Bank Withdrawal (110 pcm/sec)	Initiation of Uncontrolled RCCA Bank Withdrawal	0.00	0.00
	Power Range High Neutron Flux – High Setpoint Reached	1.67	1.68
	Reactor Trip	3.67	3.68
	Minimum DNBR Occurs	2.58	2.56
100% Power, Minimum Feedback, Slow RCCA Bank Withdrawal (1 pcm/sec)	Initiation of Uncontrolled RCCA Bank Withdrawal	0.00	0.00
	Overtemperature N-16 Setpoint Reached	94.09	90.82
	Reactor Trip	96.09	92.82
	Minimum DNBR Occurs	94.00	91.00

Table 2.8.5.4.2-2				
Uncontrolled RCCA Bank Withdrawal at Power – Limiting Results				
	Limiting value		Safety Analysis Limit	Case
	Unit 1	Unit 2		
Minimum DNBR	1.689	1.726	1.61	10% power, maximum feedback, 80 pcm/sec reactivity insertion rate (Unit 1) 100% power, maximum feedback, 37 pcm/sec reactivity insertion rate (Unit 2)
Peak Core Heat Flux (fon)	1.162	1.163	1.18	100% power, minimum feedback, 5 pcm/sec (Unit 1) 100% power, maximum feedback, 37 pcm/sec (Unit 2)
Peak Secondary System Pressure (psia)	1,275.34	1,272.45	1,318.5	10% power, minimum feedback, 14 pcm/sec (Unit 1) 10% power, minimum feedback, 7 pcm/sec (Unit 2)

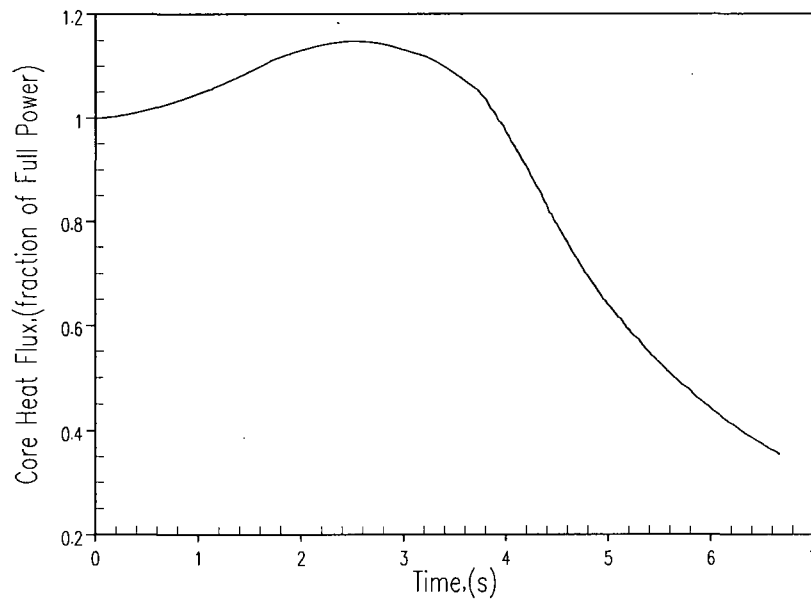
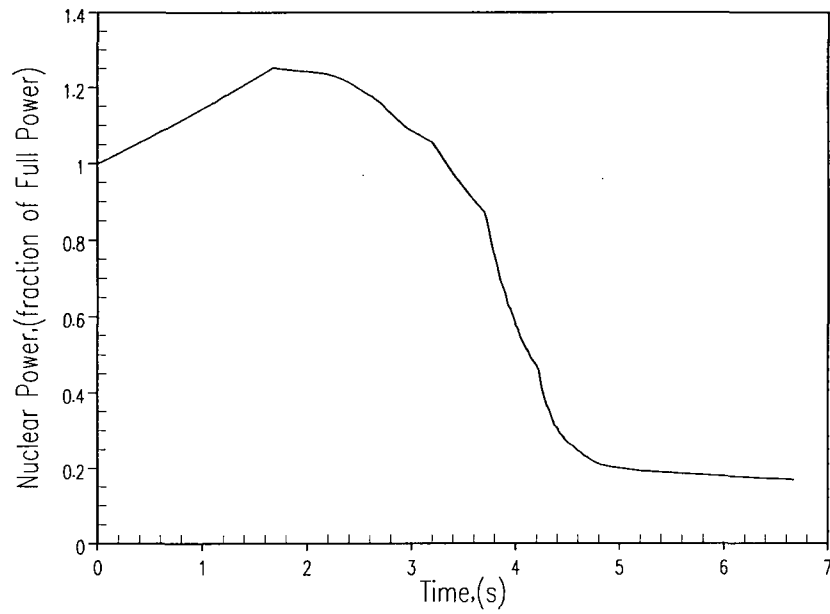


Figure 2.8.5.4.2-1 Bank Withdrawal at Power – Unit 1, Minimum Reactivity Feedback – 100% Power – 110 pcm/sec – Nuclear Power and Core Heat Flux Versus Time

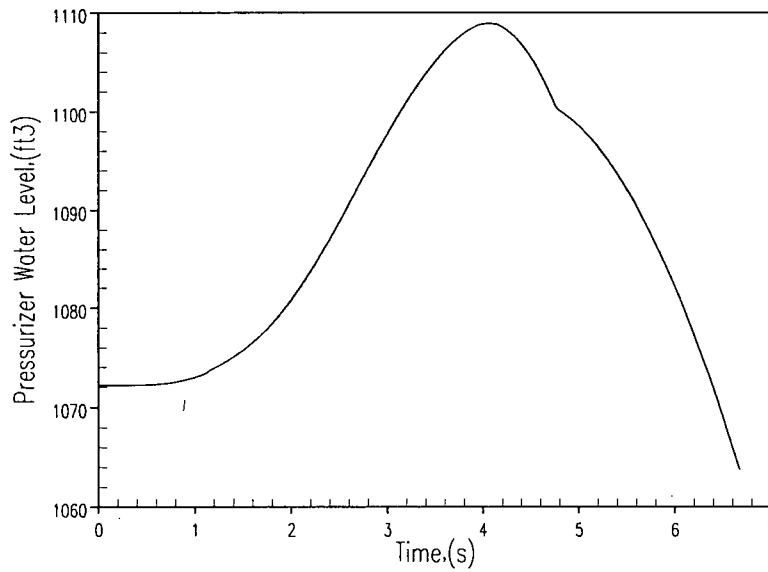
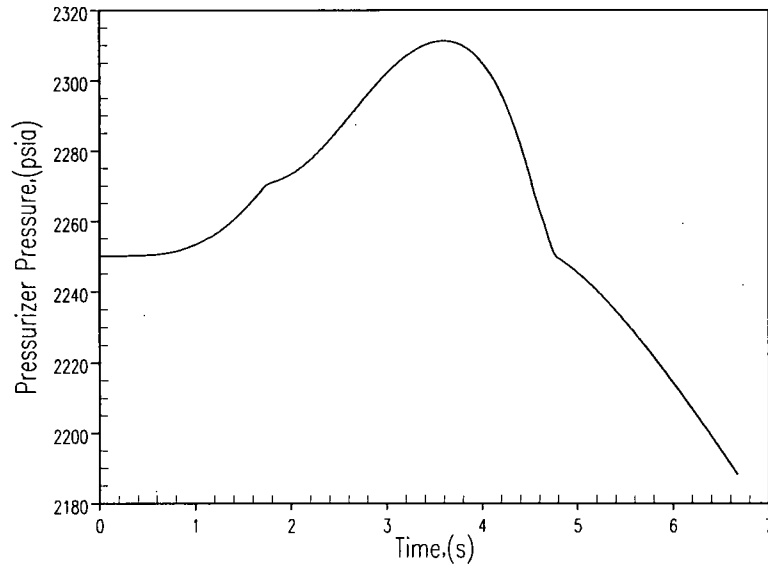


Figure 2.8.5.4.2-2 Bank Withdrawal at Power – Unit 1, Minimum Reactivity Feedback – 100% Power – 110 pcm/sec – Pressurizer Pressure and Water Volume Versus Time

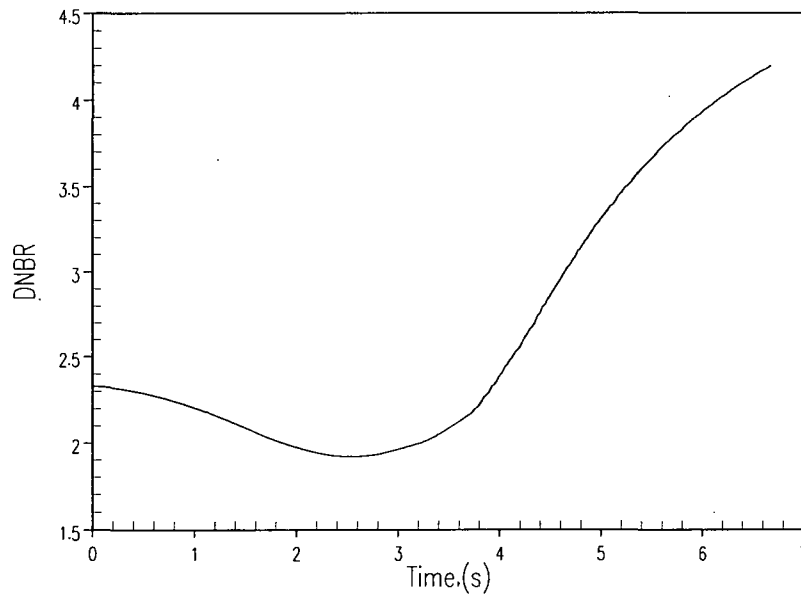
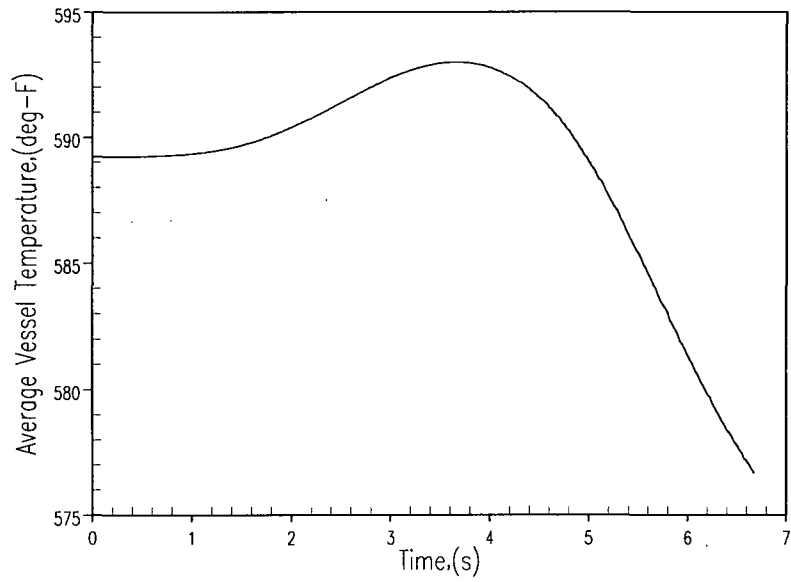


Figure 2.8.5.4.2-3 Bank Withdrawal at Power – Unit 1, Minimum Reactivity Feedback – 100% Power – 110 pcm/sec – Vessel Average Temperature and DNBR Versus Time

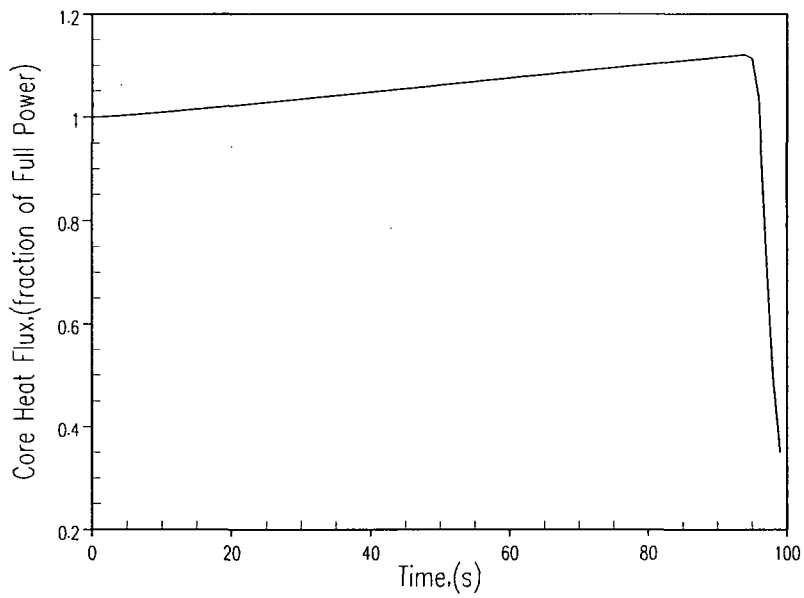
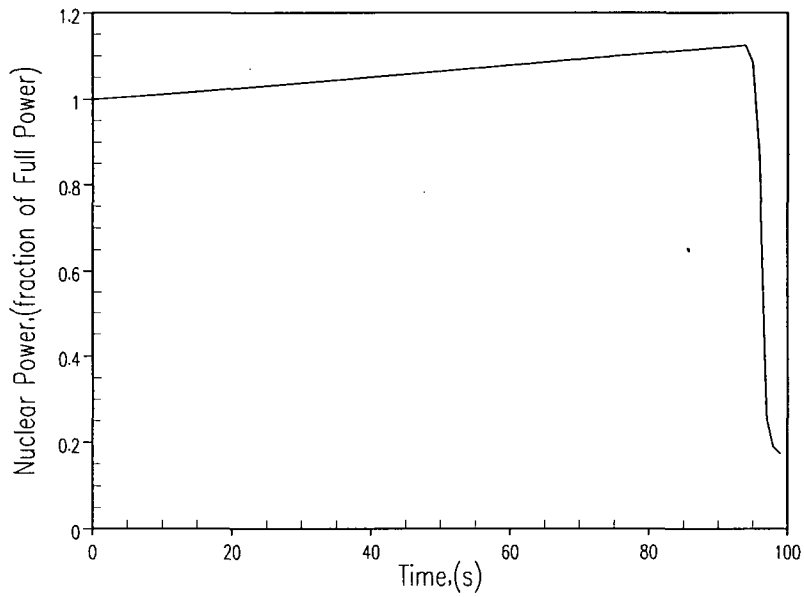


Figure 2.8.5.4.2-4 Bank Withdrawal at Power – Unit 1, Minimum Reactivity Feedback – 100% Power – 1 pcm/sec – Nuclear Power and Core Heat Flux Versus Time

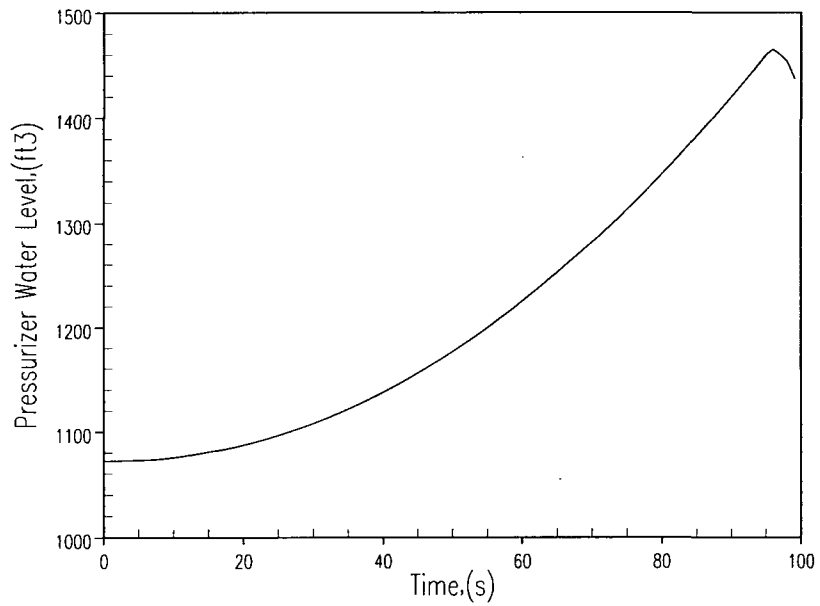
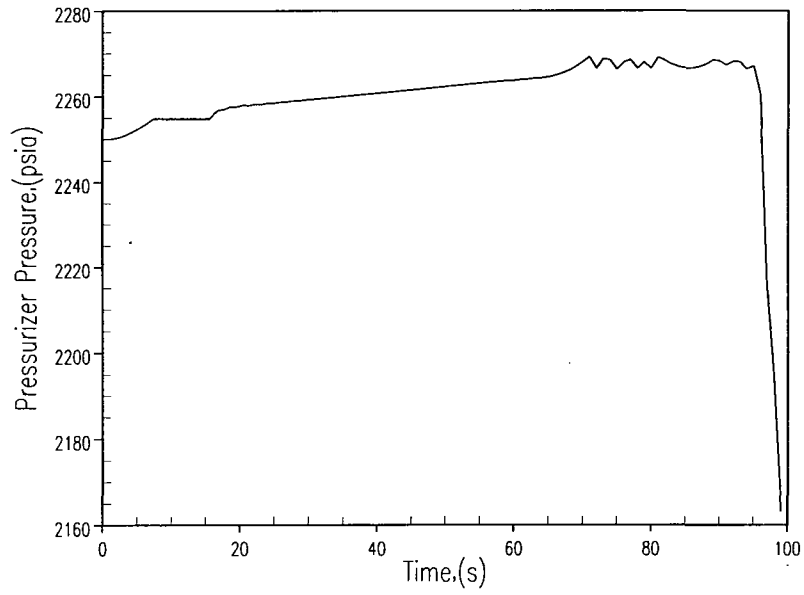


Figure 2.8.5.4.2-5 Bank Withdrawal at Power – Unit 1, Minimum Reactivity Feedback – 100% Power – 1 pcm/sec – Pressurizer Pressure and Water Volume Versus Time

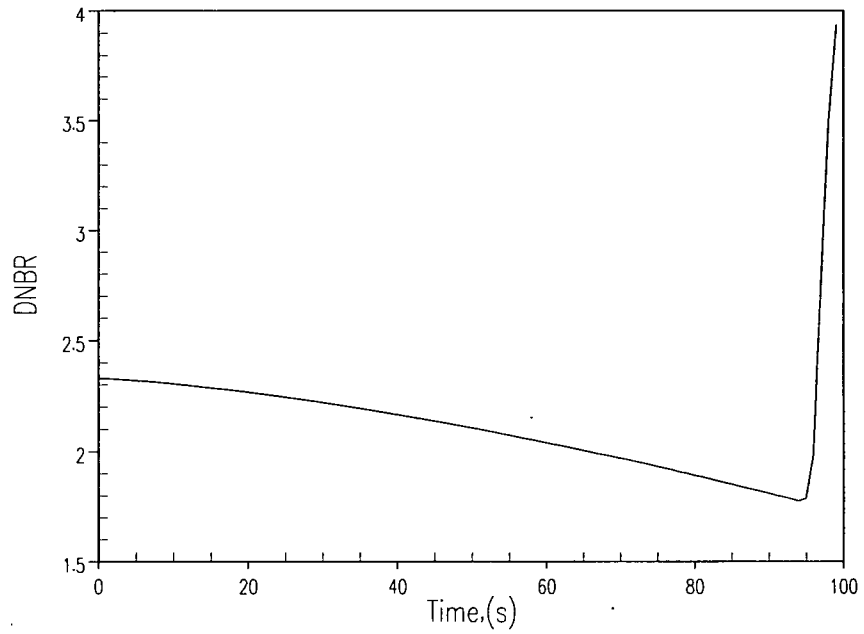
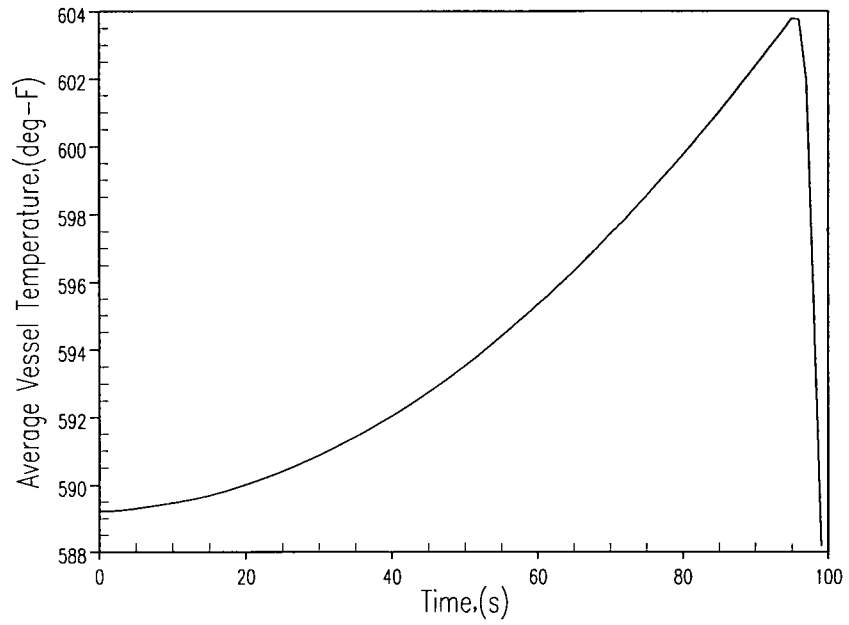


Figure 2.8.5.4.2-6 Bank Withdrawal at Power – Unit 1, Minimum Reactivity Feedback – 100% Power – 1 pcm/sec – Vessel Average Temperature and DNBR Versus Time

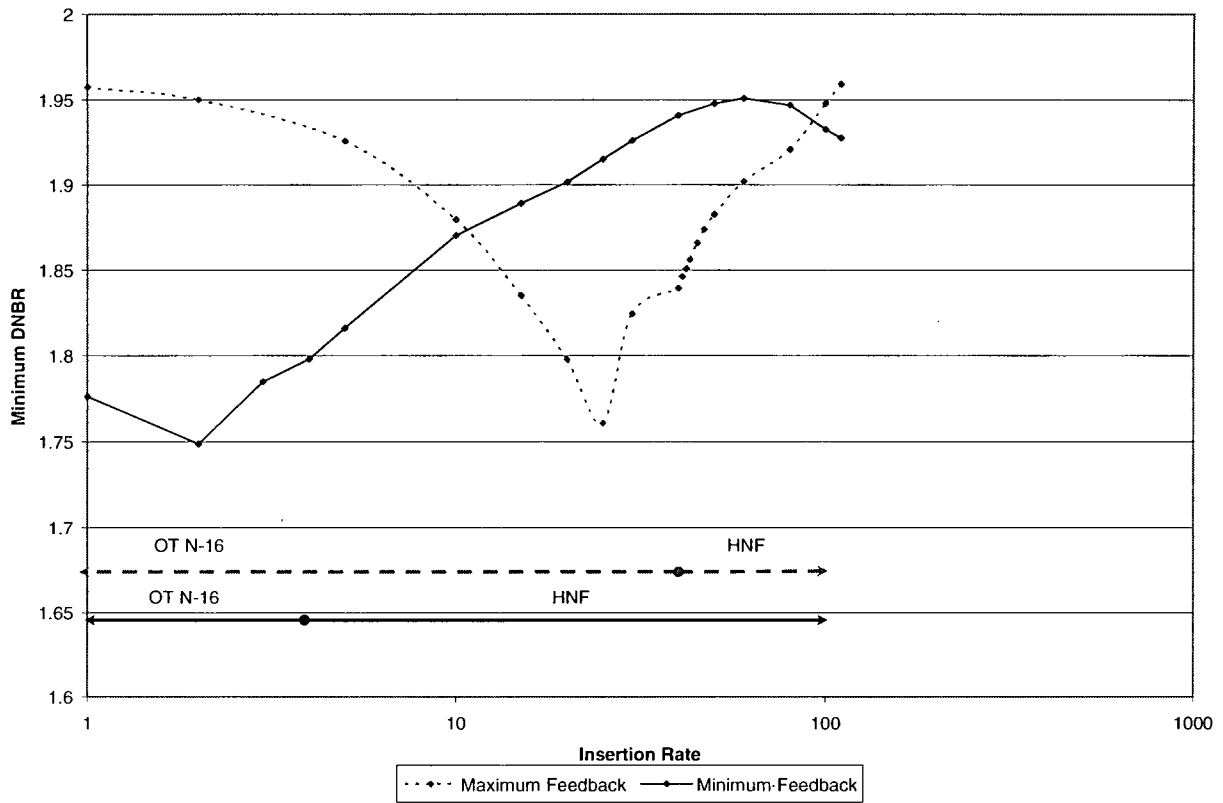


Figure 2.8.5.4.2-7 Bank Withdrawal at Power – Unit 1, 100% Power – Minimum DNBR Versus Reactivity Insertion Rate

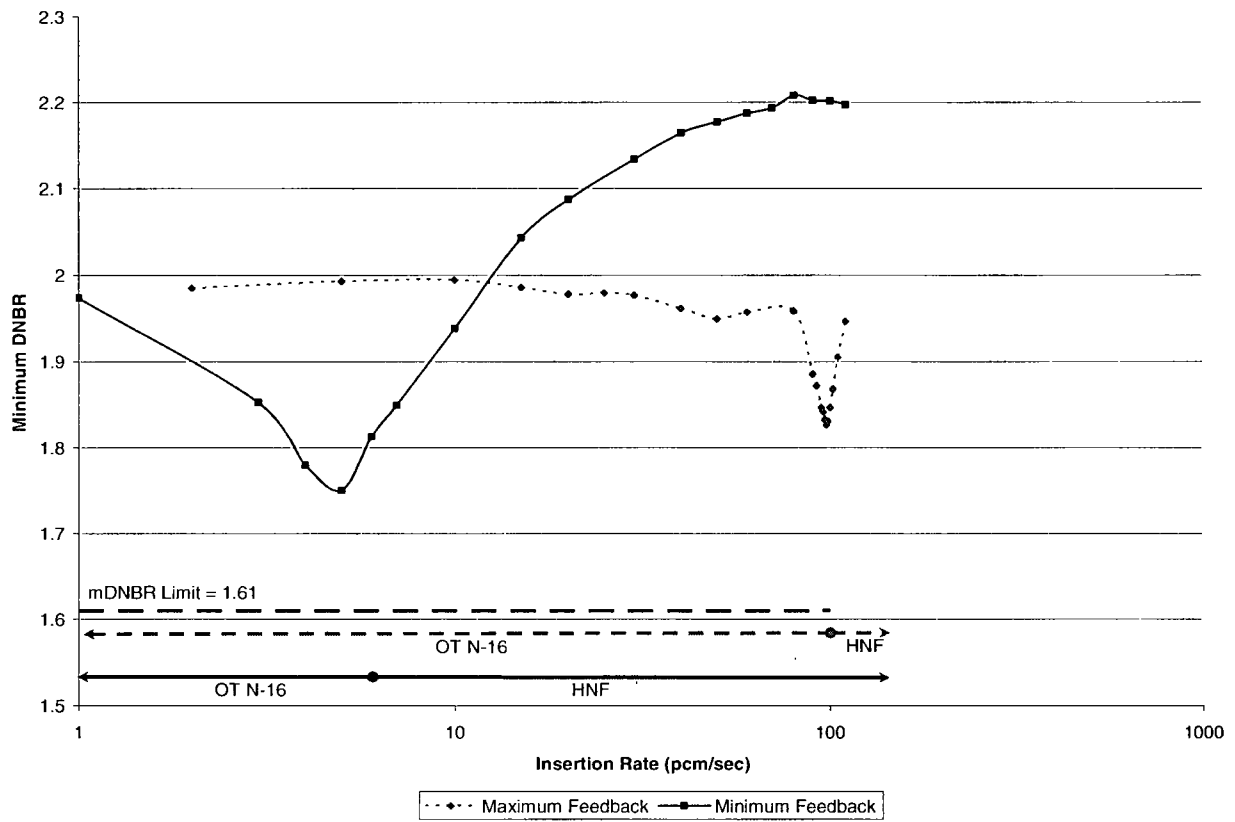


Figure 2.8.5.4.2-8 Bank Withdrawal at Power – Unit 1, 60% Power – Minimum DNBR Versus Reactivity Insertion Rate

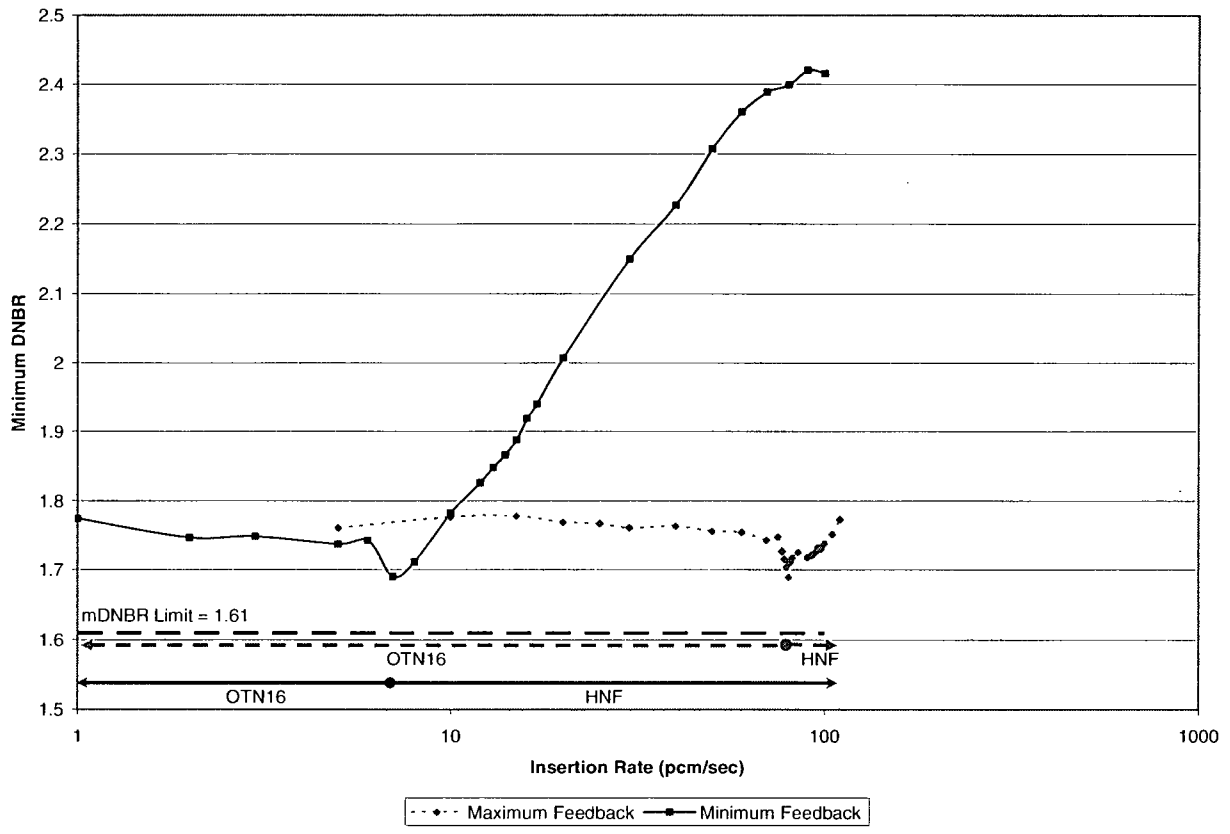


Figure 2.8.5.4.2-9 Bank Withdrawal at Power – Unit 1, 10% Power – Minimum DNBR Versus Reactivity Insertion Rate

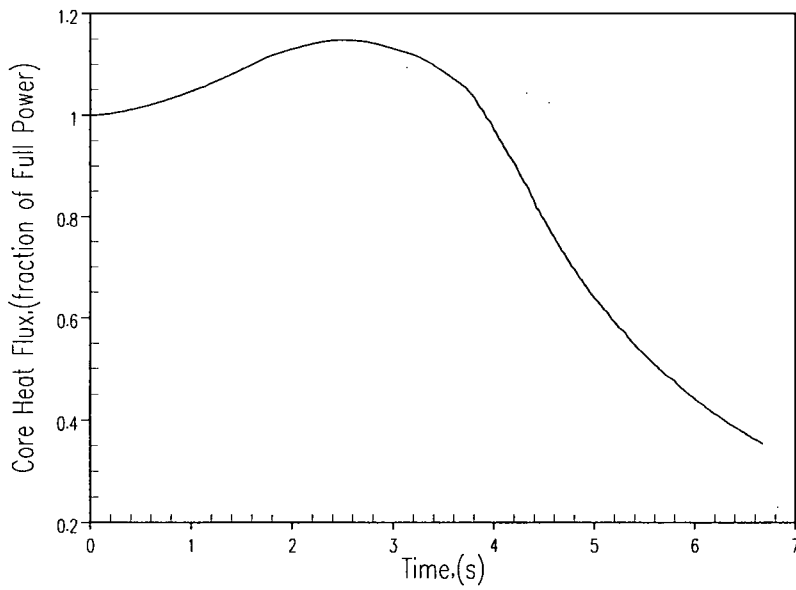
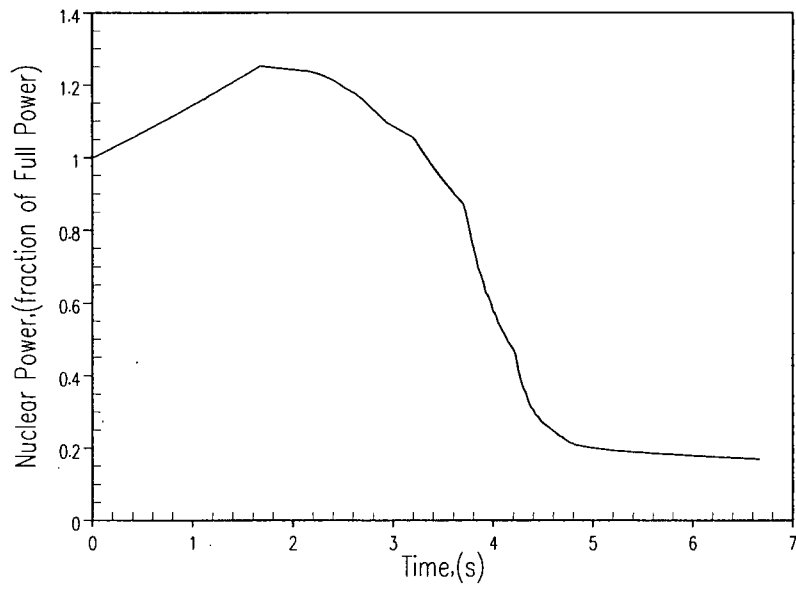


Figure 2.8.5.4.2-10 Bank Withdrawal at Power – Unit 2, Minimum Reactivity Feedback – 100% Power – 110 pcm/sec – Nuclear Power and Core Heat Flux Versus Time

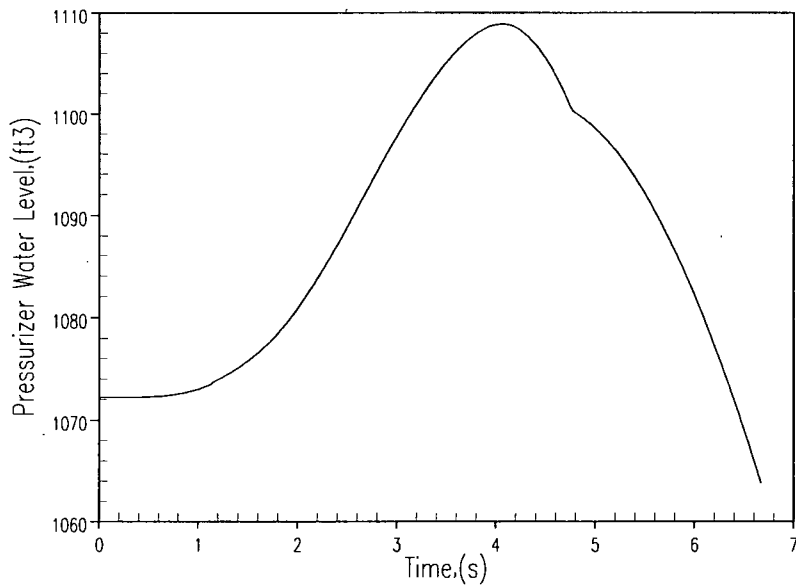
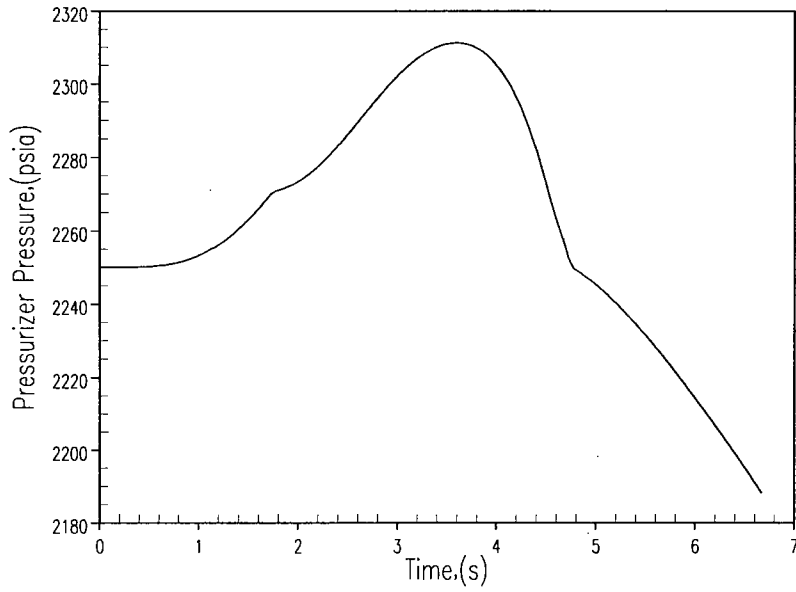


Figure 2.8.5.4.2-11 Bank Withdrawal at Power – Unit 2, Minimum Reactivity Feedback – 100% Power – 110 pcm/sec – Pressurizer Pressure and Water Volume Versus Time

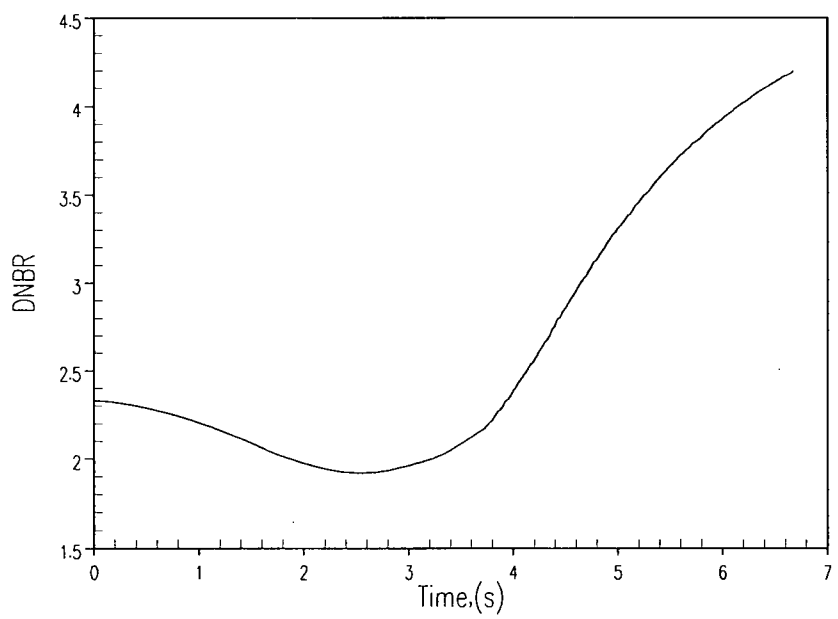
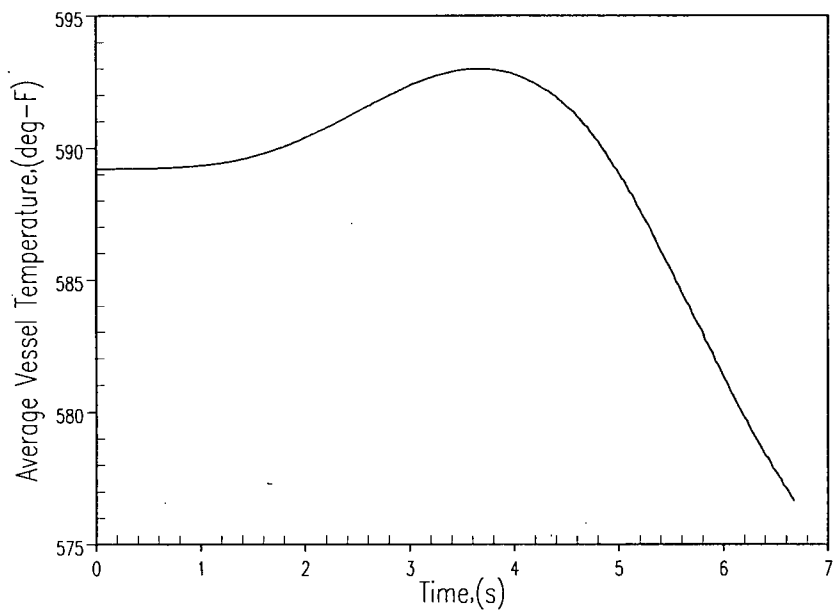


Figure 2.8.5.4.2-12 Bank Withdrawal at Power – Unit 2, Minimum Reactivity Feedback – 100% Power – 110 pcm/sec – Vessel Average Temperature and DNBR Versus Time

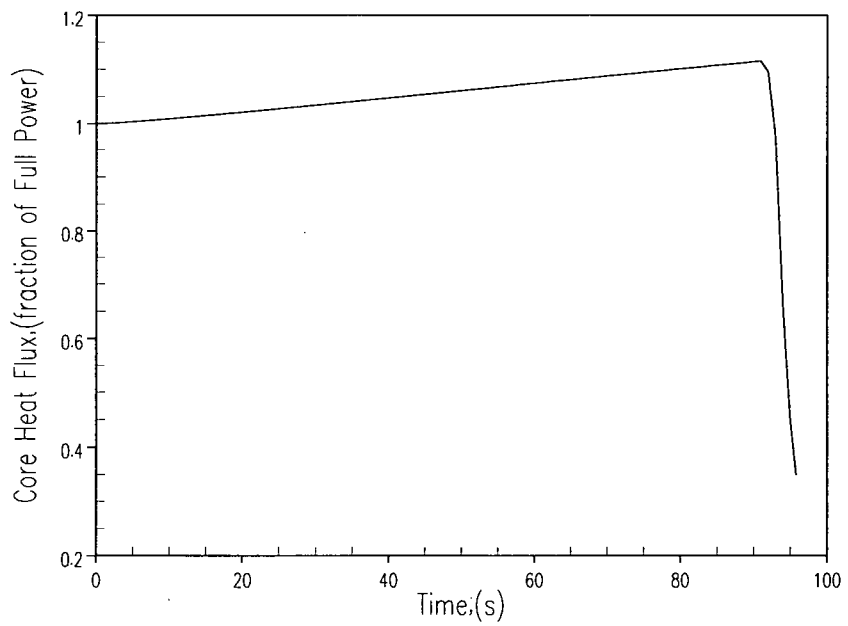
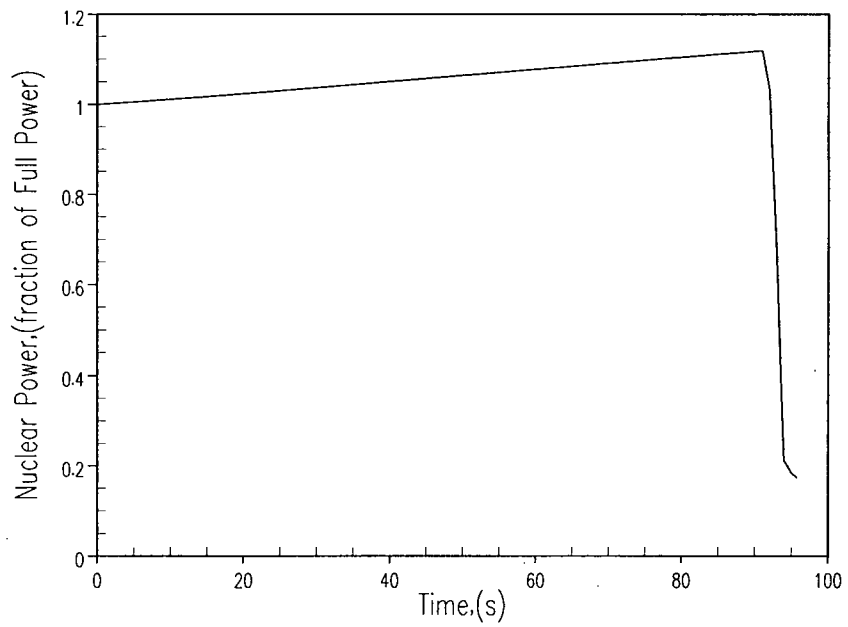


Figure 2.8.5.4.2-13 Bank Withdrawal at Power – Unit 2, Minimum Reactivity Feedback – 100% Power – 1 pcm/sec – Nuclear Power and Core Heat Flux Versus Time

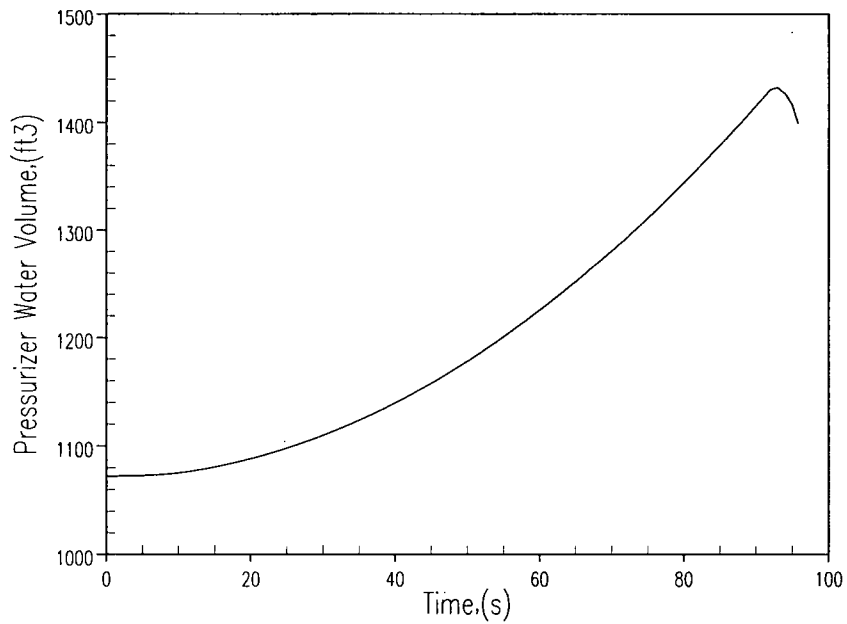
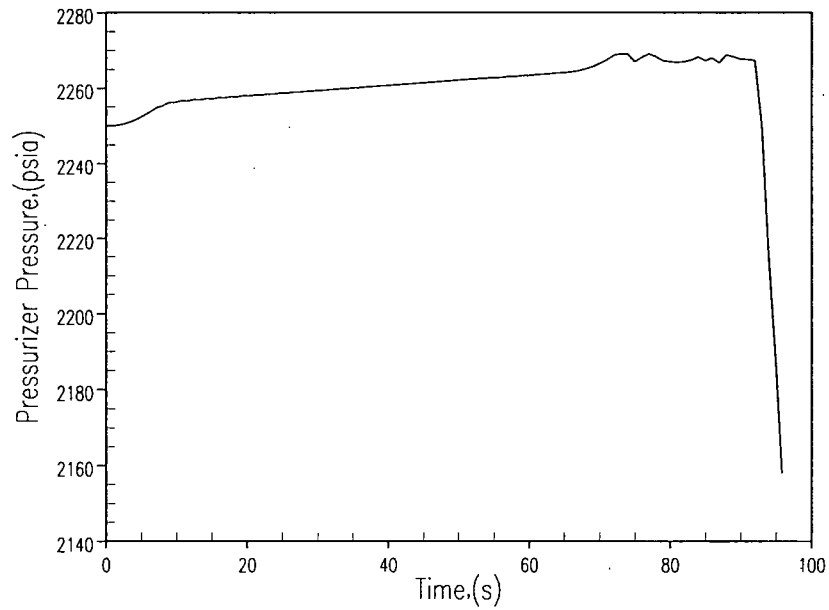


Figure 2.8.5.4.2-14 Bank Withdrawal at Power – Unit 2, Minimum Reactivity Feedback – 100% Power – 1 pcm/sec – Pressurizer Pressure and Water Volume Versus Time

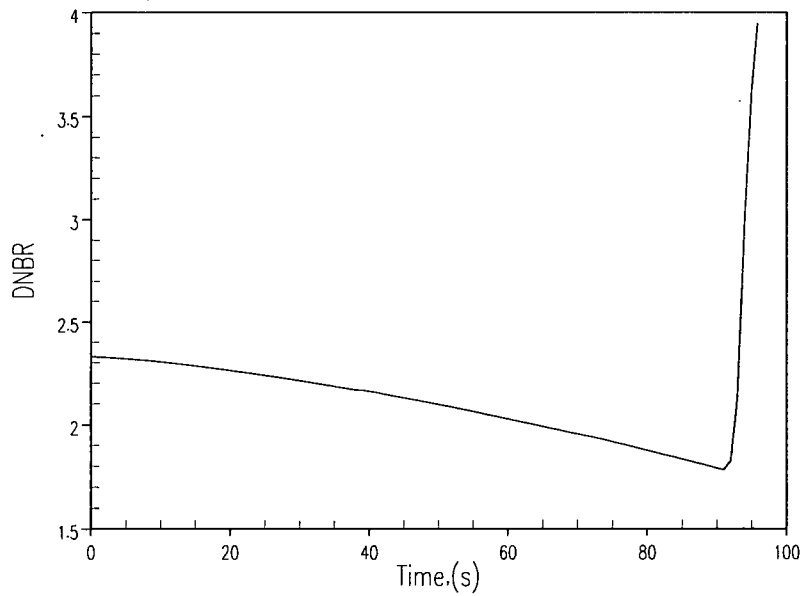
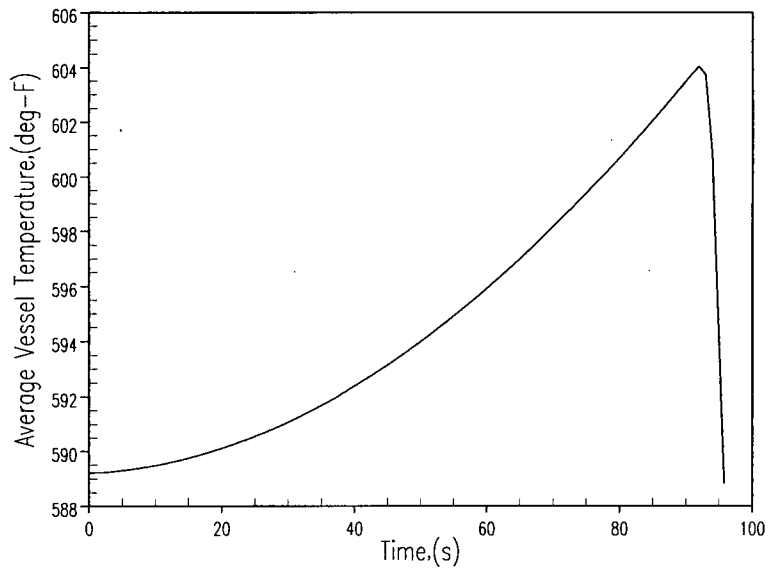


Figure 2.8.5.4.2-15 Bank Withdrawal at Power – Unit 2, Minimum Reactivity Feedback – 100% Power – 1 pcm/sec – Vessel Average Temperature and DNBR Versus Time

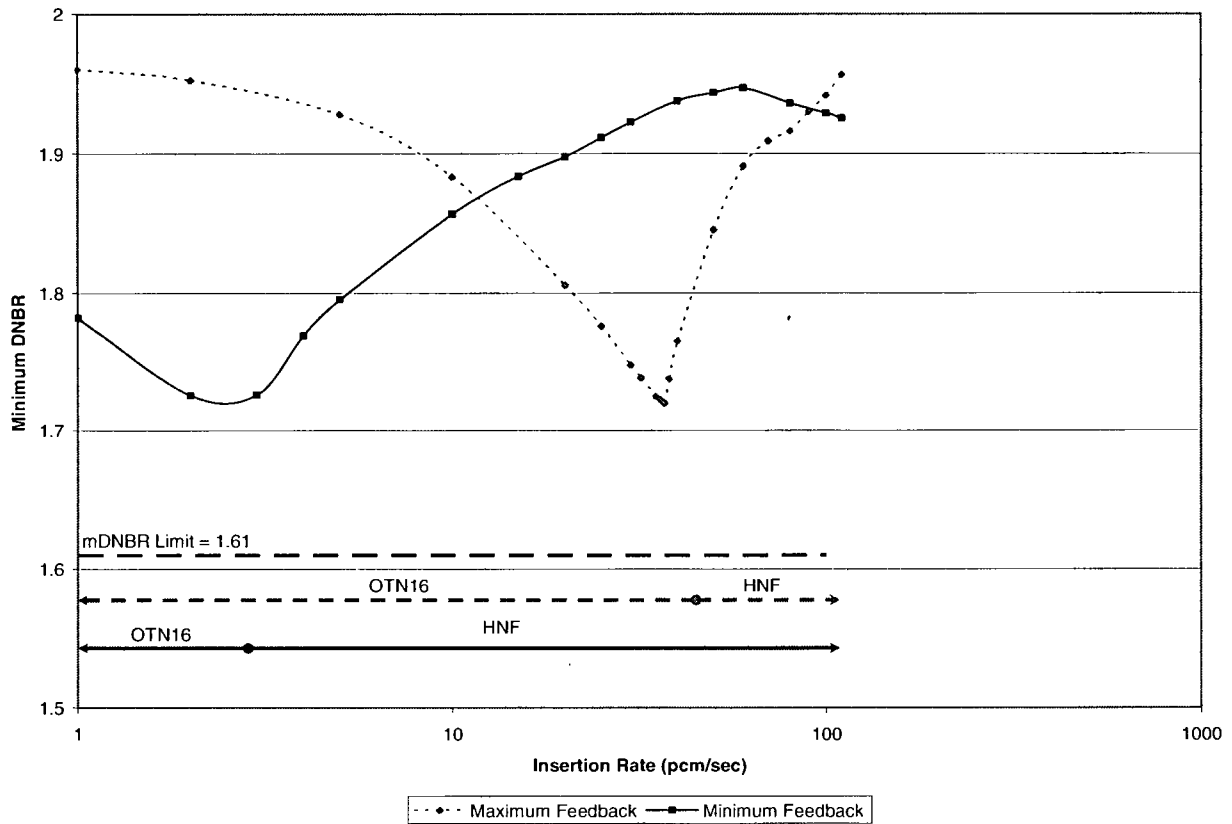


Figure 2.8.5.4.2-16 Bank Withdrawal at Power – Unit 2, 100% Power – Minimum DNBR Versus Reactivity Insertion Rate

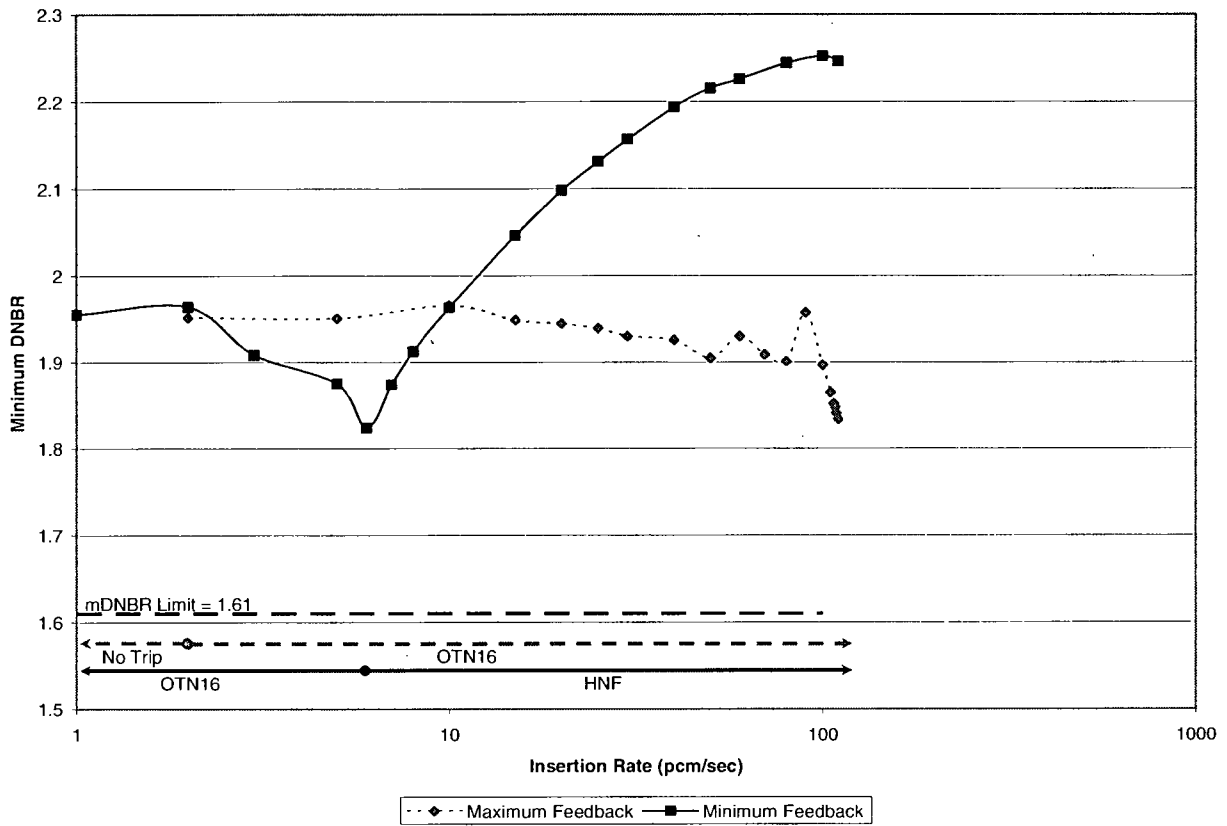


Figure 2.8.5.4.2-17 Bank Withdrawal at Power – Unit 2, 60% Power – Minimum DNBR Versus Reactivity Insertion Rate

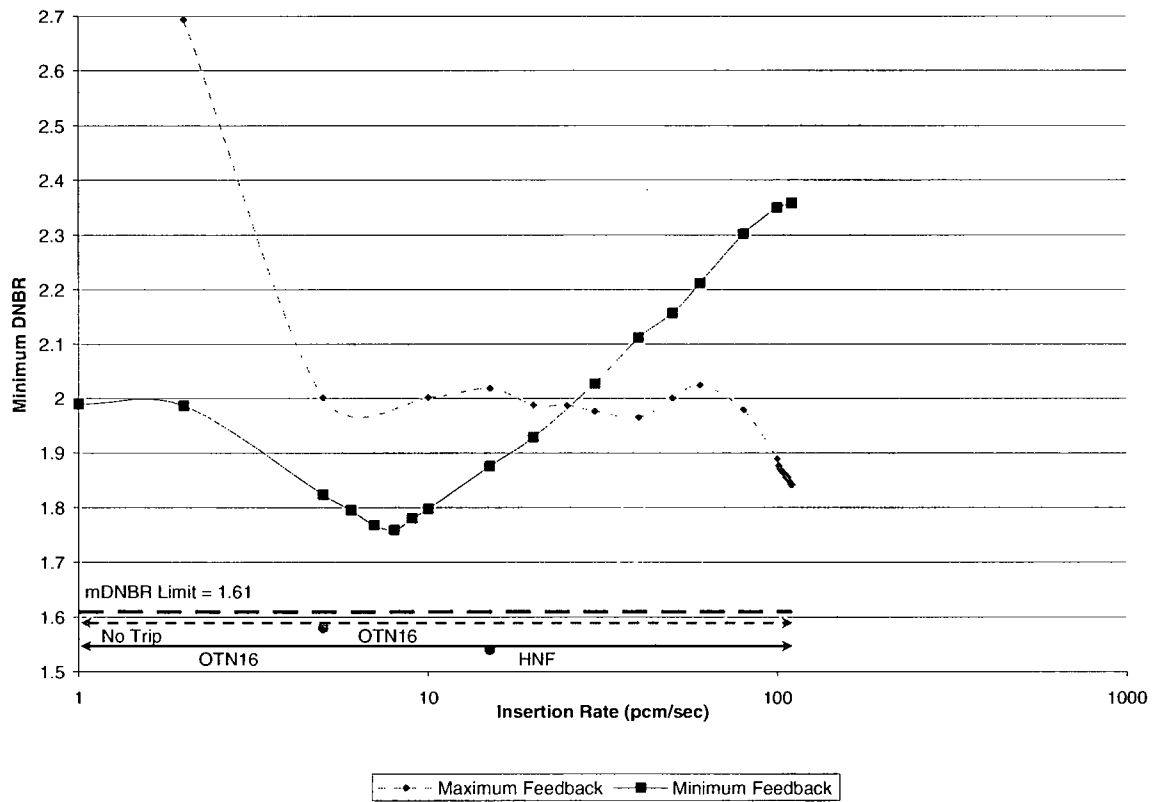


Figure 2.8.5.4.2-18 Bank Withdrawal at Power – Unit 2, 10% Power – Minimum DNBR Versus Reactivity Insertion Rate

2.8.5.4.3 Control Rod Misoperation

2.8.5.4.3.1 Regulatory Evaluation

The Luminant Power review covered the types of control rod misalignments that are assumed to occur, including those caused by a system malfunction or operator error.

The review covered:

- The descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (such as turbine runback, rod withdrawal prohibit, or rod block) that can mitigate the effects or prevent the occurrence of various misalignments
- The sequence of events
- The analytical model used for analyses
- The important inputs to the calculations
- The results of the analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor coolant system be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.
- GDC-20, insofar as it requires that the protection system be designed to automatically initiate the reactivity control systems to ensure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to automatically initiate operation of systems and components important-to-safety under accident conditions.
- GDC-25, insofar as it requires that the protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems.

Current Licensing Basis

As noted in Comanche Peak Nuclear Power Plant (CPNPP) Final Safety Analysis Report (FSAR) Section 3.1 the design bases of CPNPP are measured against the Nuclear Regulatory Commission (NRC) GDC for Nuclear Power Plants, Appendix A to 10 CFR 50. The adequacy of the CPNPP design relative to the general design criteria is discussed in the FSAR Section 3.1.

Specifically, the adequacy of CPNPP safety-related structures, systems, and components with respect to nuclear design relative to conformance to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:

1. Assure that fuel damage is not expected during normal core operation and operational transients (Condition I) or any transient conditions arising from occurrences of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These failures are within the capability of the plant cleanup system to mitigate, and are consistent with plant design bases.
2. Ensure return of the reactor to a safe shutdown state following infrequent incident (Condition III) events with only a small fraction of fuel rods damaged, although sufficient fuel damage might occur to preclude immediate resumption of operation.
3. Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

Note that the term “fuel damage” as used in Item 1 above is defined as penetration of the fission product barrier (that is, the fuel rod cladding). Also note that American National Standards Institute (ANSI) N18.2-1973 expands the definitions of the four conditions enumerated in Items 1 through 3 above.

FSAR Chapter 4 discusses the design bases and the design evaluation of reactor components. FSAR Chapter 7 provides the details of the control and protection systems instrumentation design and logic. This information supports the FSAR Chapter 15 accident analysis, which shows that acceptable fuel design limits are not exceeded for Conditions I and II occurrences.

GDC-10 also requires that the departure from nucleate boiling ratio (DNBR) remain above the 95/95 DNBR limit at all times during the transient. Section 4.4.1.1 of the FSAR indicates that the safety analysis limit (SAL) DNBR remains above the 95/95 value. Therefore, CPNPP meets the current licensing basis requirements regarding DNBR limits.

- GDC-20, Protection System Functions, is described in FSAR Section 3.1.3.1.

A fully automatic protection system, with appropriate redundant channels, is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is the Institute of Electrical and Electronic Engineering (IEEE) Standard 279-1971 and IEEE Standard 379-1972. The reactor

protection system automatically initiates a reactor trip when any variable exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the rod drive mechanisms of all of the full-length rod cluster control assemblies (RCCA). This causes the rods to insert by gravity, which rapidly reduces reactor power output. The response and adequacy of the protection system have been verified by analysis of expected transients.

The engineered safety features (ESF) actuation system automatically initiates emergency core cooling, and other safeguards functions, by sensing accident conditions using redundant analog channels measuring diverse variables. Manual actuation of safeguards equipment may be performed where ample time is available for operator action. The ESF actuation system automatically trips the reactor on manual or automatic safety injection signal (SIS) generation.

- GDC-25, Protection System Requirements for Reactivity Control Malfunctions, is described in FSAR Section 3.1.3.6.

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by full-length rod insertion is completely independent of the normal control function, since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Therefore, in the postulated accidental withdrawal (assumed to be initiated by a control malfunction), flux, temperature, pressure, level, and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

FSAR Section 15.4 addresses reactivity and power distribution anomalies, and contains information on a number of Condition II events. Two reactivity control systems are provided. They are the RCCAs and chemical shim (boric acid). The rod cluster control assemblies (RCCAs) are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The rod control system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specific fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses, and the most reactive rod cluster is assumed to be in the fully withdrawn position.

2.8.5.4.3.2 Technical Evaluation

The specific acceptance criteria applied for this event are as follows:

- The DNBR should remain above the 95/95 DNBR limit at all times during the dropped RCCA(s)/bank and statically misaligned RCCA events, and that the single RCCA withdrawal event be limited to a small fraction of fuel rods that experience DNB. Demonstrating that the DNB limits are met satisfies the requirements of GDC-10.
- Per GDC-20, the protection system should be designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to sense accident conditions and initiate the operation of safety-related systems and components. For this event, protection is provided via the overtemperature N-16 trip, but only for the most limiting cases. The nonlimiting cases considered do not require protection.
- GDC-25 requires that the protection system is designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods. Demonstrating that the fuel design limits (that is, DNBR) are met satisfies the requirements of GDC-25.

The following discussion demonstrates that all applicable acceptance criteria are met for this event at CPNPP Units 1 and 2 at uprate conditions.

2.8.5.4.3.2.1 Introduction

The RCCA misalignment events include the following:

- One or more dropped RCCAs from the same group
- A dropped RCCA bank
- A statically misaligned RCCA
- Withdrawal of a single RCCA

Each RCCA has a position indicator channel that displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates an alarm and a Control Room annunciator. Group demand position is also indicated.

Full-length RCCAs are moved in preselected banks, and the banks are moved in the same preselected sequence. Each control bank of RCCAs is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism.

Since the stationary gripper, moveable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

A dropped RCCA or RCCA bank is detected by one or more of the following:

- Sudden drop in the core power level as seen by the nuclear instrumentation system
- Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples
- Rod at bottom signal
- Rod deviation alarm
- Rod position indication

Dropping of a full-length RCCA is assumed to be initiated by a single electrical or mechanical failure that causes any number and combination of rods from the same group of a given control bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to rapidly decrease. An increase in the hot channel factor can occur due to the skewed power distribution representative of a dropped rod configuration. For this event, it must be shown that the departure from nucleate boiling (DNB) design basis is met for the combination of power, hot channel factor, and other system conditions which exist following a dropped rod.

Misaligned assemblies are detected by:

- Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples
- Rod deviation alarm
- Rod position indicators

For CPNPP Units 1 and 2, rod position is displayed in 6-step increments with an accuracy of ± 4 steps. Deviation of any RCCA from its group by twice this distance (12 steps) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 6 steps. If the rod deviation alarm is not operable, the operator is required to take action per the Technical Requirements Manual.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure the alignment of the non-indicated RCCAs. The operator is also required to take action per the Technical Specifications.

In the extremely unlikely event of simultaneous electrical failures that could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature N-16 reactor trip, although due to the increase in local power density it is not possible in all cases to provide assurance that the core safety limits will not be violated.

2.8.5.4.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The dropped RCCA, dropped RCCA bank, and statically misaligned RCCA events are classified as Condition II events (faults of moderate frequency) as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. The single RCCA withdrawal incident is classified as an ANS Condition III event, as discussed below.

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full-power operation. The operator could deliberately withdraw a single RCCA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event analyzed must result from multiple wiring failures or multiple deliberate operator actions and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is so low that the limiting consequences may include slight fuel damage. Thus, consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged."

Refer to Tables 2.8.5.0-1 and 2.8.5.0-6 in Section 2.8.5.0 for detailed acceptance criteria and initial conditions used in the dropped RCCA/dropped RCCA bank analysis. For the statically misaligned RCCA and single RCCA withdrawal events, see the analysis descriptions and results in Sections 2.8.5.4.3.2.3 and 2.8.5.4.3.2.4 for details of the inputs and acceptance criteria.

2.8.5.4.3.2.3 Description of Analyses and Evaluations

One or More Dropped RCCAs from the Same Group

The LOFTRAN computer code calculates transient system responses for the evaluation of a dropped RCCA event. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and main steam safety valves (MSSVs). The code computes pertinent plant variables including temperatures, pressures, and power levels.

Transient RCS statepoints (temperature, pressure, and power) are calculated by LOFTRAN. Nuclear models are used to obtain a hot-channel factor consistent with the primary-system conditions and reactor power. By incorporating the primary conditions from the transient analysis and the hot-channel factor from the nuclear analysis, it is shown that the DNB design basis is met using dropped rod limit lines developed with the VIPRE code (Reference 1). The transient response analysis, nuclear peaking factor analysis, and performance of the DNB design basis confirmation are performed in accordance with the approved methodology described in Reference 2.

Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core. Assumptions made in the methodology (Reference 2) for the dropped RCCA(s) analysis provide a bounding analysis for the dropped RCCA bank.

Statically Misaligned RCCA

Steady-state power distributions are analyzed using the appropriate nuclear physics computer codes. The peaking factors are then compared to peaking factor limits developed using the VIPRE code, which are based on meeting the DNBR design criterion. The following cases are examined in the analysis assuming the reactor is at full power: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out. It is assumed that the incident occurs at the time in the cycle with maximum predicted peaking factors. This assures a conservative $F_{\Delta H}$ for the misaligned RCCA configuration.

Single RCCA Withdrawal

Power distributions within the core are calculated. The peaking factors are then used by VIPRE to calculate the DNBR for the event. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, was analyzed. This incident is assumed to occur at beginning-of-life since this condition results in a minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

2.8.5.4.3.2.4 Control Rod Misalignment Results

One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period since power is decreasing rapidly. Either reactivity feedback or control bank withdrawal will re-establish power.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank

to restore nominal power. In all cases, the minimum DNBR remains above the limit value, and the peak kW/ft and clad strain limits are met.

Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern, and establishing the automatic rod control mode of operation as the limiting case.

Dropped RCCA Bank

A dropped RCCA bank results in a large negative reactivity insertion. Due to the relatively large worth of the dropped bank, and if the turbine load is constant, a reactor trip may occur on low pressurizer pressure due to the mismatch between the reactor power and the turbine power. The core is not adversely affected during this period, since power is decreasing rapidly. In the event a reactor trip does not occur, the initial power reduction from a dropped RCCA bank is large and the power return due to reactivity feedback and control bank withdrawal is far less than seen from one or more dropped RCCAs from the same group. In either instance, the minimum DNBR remains above the limit value, and the peak kW/ft and clad strain limits are met.

Following plant stabilization, the operator may manually retrieve the RCCA(s) by following approved operating procedures.

Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a control bank insertion position that is as deep as allowed by the DNBR and power peaking factor limits. The full power insertion limits on control bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with bank D inserted to its full-power insertion limit and one RCCA fully withdrawn, the DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, and RCS pressure and temperature are at their nominal values including uncertainties, but with the increased radial peaking factor associated with the misaligned RCCA.

DNB calculations have not been performed specifically for RCCAs missing from other banks. However, power shape calculations have been done as required for the RCCA ejection analysis.

Inspection of the power shapes shows that the DNB and peak kW/ft situation is less severe than the bank D case discussed above assuming insertion limits on the other banks equivalent to a bank D full-in insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, and RCS pressure and temperature are at their nominal values including uncertainties, but with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of an RCCA group misalignment condition, the operator is required to take action per the plant Technical Specifications and operating instructions.

Single RCCA Withdrawal

For the single rod withdrawal event, two cases have been considered as follows:

1. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented for the uncontrolled RCCA bank withdrawal at power event. However, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast enough to prevent the minimum DNBR from falling below the limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature N-16 trip would be expected to trip the plant shows that an upper limit for the number of fuel rods with a DNBR less than the limit value is 5 percent of the total rods in the core.
2. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case (1) described above.

For such cases as above, a reactor trip will ultimately ensue, although not sufficiently fast enough in all cases to prevent a minimum DNBR in the core of less than the limit value. Following reactor trip, normal shutdown procedures are followed. No single failure of the reactor trip system will negate the protection functions required for the single RCCA withdrawal accident, or adversely affect the consequences of the accident.

2.8.5.4.3.2.5 Results

The evaluation of the dropped rod event using the methodology in Reference 2, encompassing all possible dropped RCCA or RCCA bank worths delineated in Reference 2, concluded that the minimum DNBR remains above the safety analysis limit value, and the peak kW/ft and clad strain limits are met for CPNPP Units 1 and 2. For all cases of any single RCCA fully inserted, or bank D inserted to the rod insertion limit and any single RCCA in that bank fully withdrawn (static misalignment), the minimum DNBR remains above the limit value for CPNPP Units 1 and 2. Therefore, the DNB design criterion is met and the RCCA misalignments do not result in core damage given implementation of the SPU Program. For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total number of fuel rods in the core for CPNPP Units 1 and 2.

2.8.5.4.3.3 Conclusion

Luminant Power has reviewed the analyses of control rod misalignment events and concluded that the analyses have adequately accounted for the changes in core design required for plant operation at the uprate power level. It is also concluded that the analyses were performed using acceptable analytical models. Luminant Power further concluded that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits will not be exceeded during normal or anticipated operational transients. Based on this, it is concluded that the plant will continue to meet the requirements of GDCs -10, -20, and -25 following implementation of the uprate. Therefore, Luminant Power finds the uprate acceptable with respect to control rod misalignment events.

2.8.5.4.3.4 References

1. WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.
2. WCAP-11394, "Methodology for the Analysis of the Dropped Rod Event," January 1990.

2.8.5.4.4 Startup of an Inactive Loop at an Incorrect Temperature

2.8.5.4.4.1 Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler or deborated water into the core. This event causes an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. This event is precluded by the Comanche Peak Nuclear Power Plant (CPNPP) Technical Specifications as discussed below. Therefore, no additional review was required.

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).
- GDC-15, insofar as it requires that the RCS and its associated auxiliary system be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during AOOs.
- GDC-20, insofar as it requires that the protection system be designed to automatically initiate the operation of appropriate systems to ensure that the specified acceptable fuel design limits are not exceeded as a result of AOOs.
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.
- GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

Current Licensing Basis

Final Safety Analysis Report (FSAR) Section 15.4.4.1 states that the CPNPP Technical Specifications require all four reactor coolant loops to be in operation while at power or in startup conditions (Modes 1 and 2). Since, by definition, the event can only occur during power operations when a loop is out of service, a detailed analysis of this event was not performed.

2.8.5.4.4.2 Technical Evaluation

If the plant is operating with one pump out of service, there would be reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there would be a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop would be lower than the reactor core inlet temperature.

As indicated in the previous section, the event is precluded by the CPNPP Technical Specifications, which will not change for the uprate. Therefore, a detailed analysis of this event is not required.

2.8.5.4.4.3 Conclusion

The evaluation of the inactive loop startup event was reviewed, and it is concluded that the plant will continue to meet the requirements of GDCs-10, -15, -20, -26, and -28 following implementation of the proposed SPU. Therefore, the proposed SPU is acceptable with respect to the inactive loop startup event.

2.8.5.4.5 Chemical and Volume Control System Malfunction Resulting in a Decrease in Boron Concentration in the Reactor Coolant

2.8.5.4.5.1 Regulatory Evaluation

Unborated water can be added to the reactor coolant system (RCS) via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator should stop this unplanned dilution before the shutdown margin is eliminated. The review consisted of:

- The conditions at the time of the unplanned dilution
- The causes
- The initiating events
- The sequence of events
- The analytical model used for analyses
- The values of parameters used in the analytical model
- The results of the analyses

Acceptance criteria for the boron dilution event are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences
- GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems are designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including anticipated operational occurrences
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under normal operating conditions, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in CPNPP FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I) or any transient conditions arising from occurrences of moderate frequency (Condition II).
2. To ensure return of the reactor to a safe state following infrequent incident (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage might occur to preclude immediate resumption of operation.
3. To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and design evaluation of reactor components, including the fuel, reactor vessel internals, and reactivity control systems. Details of the control and protection systems instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the FSAR Chapter 15 accident analyses, which show that the acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components. FSAR Chapter 5 discusses the RCS design.

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- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control methods are provided. These are the rod cluster control assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the full-length control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses, and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state independent of the position of the control rods. It can compensate for xenon burnout transients.

Details of the construction of the RCCAs are presented in FSAR Chapter 4 and their operation is discussed in FSAR Chapter 7. The means of controlling the boric acid concentration are described in FSAR Chapter 9. Performance analyses under accident conditions are included in FSAR Chapter 15.

FSAR Section 15.4.6.1 states that the CVCS malfunctions that are considered to result in a decrease of the boron concentration in the reactor coolant are the inadvertent opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure. The addition of unborated water to the RCS results in a positive reactivity insertion and an erosion of available shutdown margin. For at power and startup conditions, Modes 1 and 2, the dilution accident erodes the shutdown margin made available through reactor trip. For shutdown mode initial conditions, Modes 3, 4, 5, and 6, the dilution accident erodes the shutdown margin inherent in the borated RCS inventory and that which may be provided by control rods (control and shutdown banks) made available through reactor trip. This event is classified as a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973.

FSAR Section 15.4.6.3 concludes that, for operating Modes 1 through 5, the results show that adequate time is available for the operator to manually terminate the source of dilution flow, assuming the specified shutdown margin requirements are met. Following termination of the dilution flow, the operator can initiate boration to recover the shutdown margin.

In addition, FSAR Section 15.4.6.3 states that no analysis is presented for Mode 6 operation since dilution during refueling is precluded by administrative control of valves in the possible dilution path.

2.8.5.4.5.2 Technical Evaluation

The specific acceptance criterion applied for the boron dilution events is that adequate operator action time is available prior to a complete loss of shutdown margin. For boron dilution events in Modes 1 through 5, there must be at least 15 minutes from operator notification (that is, first alarm) until shutdown margin is lost. For CPNPP Units 1 and 2, a boron dilution event cannot occur during Mode 6 (Refueling) due to administrative controls which isolate the RCS from the potential sources of unborated water. Additionally, for conditions when no reactor coolant pump is in operation, all dilution sources are isolated or under administrative control. Hence, a boron dilution event cannot occur during Mode 5 (Cold Shutdown), or Mode 4 (Hot Shutdown) once operation on the residual heat removal (RHR) system begins. This is consistent with inadvertent boron dilution event analysis methodology approved by the NRC for CPNPP Units 1 and 2 (Reference 1). With shutdown margin maintained, there is no return to critical and no violation of the 95/95 departure from nucleate boiling ratio (DNBR) limit (GDC-10), as well as no violation of the primary and secondary pressures limits (GDC-15). Furthermore, since a return to critical is precluded and fuel design limits are not exceeded, the requirements of GDC-26 are satisfied.

For Modes 1 and 2, the boron dilution analysis is performed to ensure that adequate time is available from alarm to total loss of shutdown margin for the operator to identify and terminate the dilution. For Modes 3 through 5, the boron dilution event is analyzed to generate operating guidelines that, when met, ensure that there is adequate time from alarm to total loss of shutdown margin for the operator to identify and terminate the dilution.

The discussion below demonstrates that all applicable acceptance criteria are met for this event at CPNPP Units 1 and 2 in operating Modes 1 through 5.

2.8.5.4.5.2.1 Introduction

Reactivity can be added to the core by feeding primary-grade water into the RCS via the reactor makeup portion of the CVCS. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of the reactor coolant makeup water during normal charging to the RCS boron concentration. As discussed below, the CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value that, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

2.8.5.4.5.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The opening of the primary water makeup control valves provides makeup to the CVCS and subsequently to the RCS, which can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump.

The limiting dilution flow path is identified as the lowest resistance flow path for an unintentional dilution. The boron dilution analysis excludes deliberate dilution operations from considerations. During intentional boron dilution operations, the plant operators are keenly aware of and continuously monitor the dilution process in progress for any sign of deviation or malfunction, such that the possibility of an undetected malfunction is considered remote. This is a standard assumption in the boron dilution analysis methodology. Thus, the limiting boron dilution flow path does not include either the normal dilute or the alternative dilute flow paths (these paths are used only for deliberate dilution operations). The limiting boron dilution flow path is the makeup flow path of the reactor makeup water system (RMWS) used in normal boration/blend operations.

The principal means of causing an inadvertent boron dilution are the opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure. The CVCS and the RMWS are designed to limit, even under various postulated failure modes, the potential rate of dilution to values that will allow sufficient time for operator response to terminate the dilution. An inadvertent dilution from the RMWS may be terminated by closing the primary water makeup control valve. All expected sources of dilution may be terminated by closing isolation valves in the CVCS. The lost shutdown margin may be regained by the opening of isolation valves to the refueling water storage tank (RWST), thus allowing the addition of borated water to the RCS.

The rate at which unborated water can be added to the RCS is limited by the design of the CVCS and RMWS. The maximum (limiting) boron dilution flow rate is 157.5 gpm for Modes 1 through 5.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the CVCS. Alarms are actuated to warn the operator when boric acid or makeup water flow rates deviate from preset values as a result of system malfunction.

A CVCS malfunction is classified as an ANS Condition II event, a fault of moderate frequency as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. Criteria established for Condition II events are as follows:

- The critical heat flux should not be exceeded. This is met by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressure in the RCS and main steam system (MSS) should be maintained below 110 percent of the design pressures.
- Fuel temperature and fuel cladding strain limits should not be exceeded. The peak linear heat generation rate should not exceed a value that would cause fuel centerline melt.

This event is analyzed to show that there is sufficient time for mitigation of an inadvertent boron dilution prior to the complete loss of shutdown margin. A complete loss of plant shutdown margin results in a return of the core to the critical condition causing an increase in the RCS temperature and heat flux. This could violate the safety analysis limit DNBR value and challenge the fuel and fuel cladding integrity. A complete loss of plant shutdown margin could also result in a return of the core to the critical condition causing an increase in RCS pressure. This could challenge the pressure design limit for the RCS.

If the shutdown margin is shown not to be lost, the condition of the plant at any point in the transient is within the bounds of those calculated for other Condition II transients. By showing that the above criteria are met for those Condition II events, it can be concluded that they are also met for the boron dilution event. Operator action is relied upon to preclude a complete loss of plant shutdown margin.

2.8.5.4.5.2.3 Description of Analyses and Evaluations

Dilution During Mode 6 – An analysis is not performed for an uncontrolled boron dilution accident during refueling. In this mode, the event is prevented by administrative control of valves in the possible dilution paths.

Dilution During Mode 5 – Typically, the plant is maintained in the cold shutdown mode when RCS ambient temperatures are required. Occasionally, reduced RCS inventory may be necessary. Mode 5 can also be a transition mode to either refueling (Mode 6) or hot shutdown (Mode 4). Through the cycle, the plant may enter Mode 5 if reduced temperatures are required in containment or as the result of a Technical Specification action statement. The plant is maintained in Mode 5 at the beginning of each cycle for startup testing of certain systems. During this mode of operation, the control banks are fully inserted. The following conditions are assumed for an uncontrolled boron dilution during cold shutdown.

- The assumed dilution flow (157.5 gpm) is the maximum flow from the RMWS assuming multiple simultaneous failures of control valves.
- The active RCS water volume for CPNPP Unit 1 is 9,903.7 ft³. The active RCS water volume for CPNPP Unit 2 is 8,594.0 ft³. The difference in the active RCS water volume between the two units is due to the different steam generator designs (CPNPP Unit 1 has Δ 76 steam generators installed; CPNPP Unit 2 still uses Model D-5 steam generators). These active volumes assume at least one reactor coolant pump is in operation, with the volume of the pressurizer and surge line excluded to assure that conservative estimates are made. Additionally, since no consideration is given to mixing in the reactor vessel upper head region, the volumes for the upper head and the downcomer from the top of the cold legs to the bottom of the upper head spray nozzles are also removed.
- When no reactor coolant pump is in operation, all dilution sources are isolated or under administrative control. This is consistent with inadvertent boron dilution event analysis methodology approved by the NRC for CPNPP Units 1 and 2.

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- The volume control tank (VCT) high water level alarm alerts the operators that a boron dilution may be in progress. This is consistent with inadvertent boron dilution event analysis methodology approved by the NRC for CPNPP Units 1 and 2.
 - A minimum ratio of initial boron concentration at the most reactive burnup to the maximum critical boron concentration is determined at the upper core average operating temperature limit and the ambient temperature assumed for Mode 5.

Dilution During Mode 5 Drained – The RCS water level can be dropped to the mid-plane of the hot leg for maintenance work that requires the steam generators to be drained. When the water level is drained down to the mid-plane of the hot leg from a filled and vented condition in cold shutdown, an uncontrolled boron dilution accident is prevented by administrative controls which isolate the RCS from the potential source of unborated water. Nevertheless, an analysis has been performed for a Mode 5 drained case.

The conditions assumed for an uncontrolled boron dilution during cold shutdown with the water level drained to the mid-plane of the hot leg are identical to those for the uncontrolled boron dilution during cold shutdown, except that the minimum water volume in the RCS is reduced to 4,513 ft³ for both CPNPP Units 1 and 2. This active volume assumes at least one reactor coolant pump in operation, with the volumes of the pressurizer, the surge line, the steam generators, the upper head and the downcomer from the top of the cold legs to the bottom of the upper head spray nozzles excluded and the hot leg volumes reduced to assure that a conservative estimate is made.

Dilution During Mode 4 – In Mode 4, the plant is being taken from a short-term mode of operation, cold shutdown (Mode 5), to a long-term mode of operation, hot standby (Mode 3). Typically, the plant is maintained in the hot shutdown mode to achieve plant heatup before entering Mode 3. The plant is maintained in Mode 4 at the beginning of each cycle for startup testing of certain systems. Throughout the cycle, the plant will enter Mode 4 if reduced temperatures are required in containment or as a result of a Technical Specification action statement. During this mode of operation, the control banks are fully inserted. In Mode 4, the primary coolant forced flow which provides mixing can be provided by either the RHR system or a reactor coolant pump, depending on system pressure. The following conditions are assumed for an uncontrolled boron dilution during hot shutdown:

- The assumed dilution flow (157.5 gpm) is the maximum flow from the RMWS assuming multiple, simultaneous failures of control valves.
- The active RCS water volume for CPNPP Unit 1 is 9,903.7 ft³. The active RCS water volume for CPNPP Unit 2 is 8,594.0 ft³. The difference in the active RCS water volume between the two units is due to the different steam generator designs (CPNPP Unit 1 has $\Delta 76$ steam generators installed; CPNPP Unit 2 still uses Model D-5 steam generators). These active volumes assume at least one reactor coolant pump is operation, with the volume of the pressurizer and surge line excluded to assure that conservative estimates are made. Additionally, since no consideration is given to mixing in the reactor vessel upper head region, the volumes for the upper head and the

downcomer from the top of the cold legs to the bottom of the upper head spray nozzles are also removed.

- When no reactor coolant pump is in operation, all dilution sources are isolated or under administrative control. This is consistent with inadvertent boron dilution event analysis methodology approved by the NRC for CPNPP Units 1 and 2 (Reference 1).
- The VCT high water level alarm alerts the operators that a boron dilution may be in progress. This is consistent with inadvertent boron dilution event analysis methodology approved by the NRC for CPNPP Units 1 and 2.
- A minimum ratio of initial boron concentration at the most reactive burnup to the maximum critical boron concentration is determined at both the upper and lower core average operating temperature limits for Mode 4.

Dilution During Mode 3 – During this mode, rod control is in manual and the rods can be either withdrawn or inserted. In Mode 3, all reactor coolant pumps may not be in operation. In an effort to balance the heat loss through the RCS and the heat removal of the steam generators, one or more of the pumps may be off to decrease heat input into the system. In the approach to Mode 2, the operator must manually withdraw the control rods and may initiate a limited dilution according to shutdown margin requirements, but not simultaneously. If the shutdown or control banks are withdrawn, the dilution scenario is similar to the Mode 2 analysis where the failure to block the source range trip results in a reactor trip and immediate shutdown of the reactor. The dilution scenario is more limiting if the control rods are not withdrawn and the reactor is shut down by boron to the Technical Specifications' minimum requirement for Mode 3. The following conditions are assumed for an uncontrolled boron dilution during hot standby:

- The assumed dilution flow (157.5 gpm) is the maximum flow from the RMWS assuming multiple, simultaneous failures of control valves.
- The active RCS water volume for CPNPP Unit 1 is 9,903.7 ft³. The active RCS water volume for CPNPP Unit 2 is 8,594.0 ft³. The difference in the active RCS water volume between the two units is due to the different steam generator designs (CPNPP Unit 1 has Δ76 steam generators installed; CPNPP Unit 2 still uses Model D-5 steam generators). These active volumes assume at least one reactor coolant pump is in operation, with the volume of the pressurizer and surge line excluded to assure that conservative estimates are made. Additionally, since no consideration is given to mixing in the reactor vessel upper head region, the volumes for the upper head and the downcomer from the top of the cold legs to the bottom of the upper head spray nozzles are also removed.
- The VCT high water level alarm alerts the operators that a boron dilution may be in progress. This is consistent with inadvertent boron dilution event analysis methodology approved by the NRC for CPNPP Units 1 and 2.

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- A minimum ratio of initial boron concentration at the most reactive burnup to the maximum critical boron concentration is determined at both the upper and lower core average operating temperature limits for Mode 3.

Dilution During Mode 2 – In this mode, the plant is being taken from one long-term mode of operation (Mode 3) to another (Mode 1). The plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle. All normal actions required to change power level, either up or down, require operator initiation. Assumed conditions at startup require the reactor to have available at least 1.30 percent Δk shutdown margin. The following conditions are assumed for an uncontrolled boron dilution during startup:

- The assumed dilution flow (157.5 gpm) is the maximum flow from the RMWS assuming multiple, simultaneous failures of control valves.
- Conservative estimates of the minimum active RCS water volume are made by excluding the pressurizer and surge line. For CPNPP Unit 1, the active RCS water volume is 11,071.5 ft³. For CPNPP Unit 2, the active RCS water volume is 9,761.9 ft³. The difference in Mode 2 active RCS water volume between the two units is due to the different steam generator designs (CPNPP Unit 1 has $\Delta 76$ steam generators installed; CPNPP Unit 2 still uses Model D-5 steam generators).
- The reactor trip on source range high flux level alerts the operators that a boron dilution may be in progress.
- The initial boron concentration is assumed to be 2,100 ppm, which is a conservative maximum value for the critical concentration at the condition of hot zero power, with the rods at the insertion limits, and no xenon.
- The critical boron concentration following reactor trip is assumed to be 1,800 ppm, corresponding to hot zero power, all rods inserted (minus the most reactive RCCA), no xenon conditions. The 300 ppm change from the initial condition noted above is a conservative minimum value.

Mode 2 is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and withdraw the control rods, a process that takes several hours. The Technical Specifications require that the shutdown margin be determined prior to approaching criticality and to be above the minimum requirement by verifying that the predicted position of the rods is within the rod insertion limits. This ensures that the reactor did not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip (nominally at 10^5 cps) after reaching permissive P-6. Too fast of a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in Mode 2, the plant status is such that minimal impact will result. The plant will slowly escalate in power to a reactor trip on the power range neutron flux low setpoint. After reactor trip, more than 15 minutes is available for operator action prior to return to criticality. Mode 2 results are summarized in Table 2.8.5.4.5-1.

Dilution During Mode 1 – In this mode, the plant can be operated in either automatic or manual rod control. With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the power range high neutron flux trip setpoint or the overtemperature N-16 trip setpoint, resulting in a reactor trip. In this case, the boron dilution transient up to the time of trip is essentially equivalent to an uncontrolled RCCA bank withdrawal at power. Following reactor trip, there is at least 15 minutes prior to criticality. This is sufficient time for the operator to determine the cause of dilution and isolate the reactor makeup water source before the available shutdown margin is lost.

With the reactor in automatic rod control, the power and temperature increase from the boron dilution results in insertion of the control rods and a decrease in the available shutdown margin. As the dilution and rod insertion continue, the rod insertion limit alarms (low and low-low settings) and axial flux difference alarm alert the operator at least 15 minutes prior to criticality that a dilution is in progress and that the Technical Specification requirement for shutdown margin may be challenged. This is sufficient time to determine the cause of dilution and isolate the reactor makeup water source before the available shutdown margin is lost.

The effective reactivity addition rate is primarily a function of the dilution rate, boron concentration, and boron worth. The following conditions are assumed for an uncontrolled boron dilution during full power:

- The assumed dilution flow (157.5 gpm) is the maximum flow from the RMWS assuming multiple, simultaneous failures of control valves.
- Conservative estimates of the minimum active RCS water volume are made by excluding the pressurizer and surge line. For CPNPP Unit 1, the active RCS water volume is 11,071.5 ft³. For CPNPP Unit 2, the active RCS water volume is 9,761.9 ft³. The difference in Mode 1 active RCS water volume between the two units is due to the different steam generator designs (CPNPP Unit 1 has Δ 76 steam generators installed; CPNPP Unit 2 still uses Model D-5 steam generators).
- The reactor trip on high flux level or the overtemperature N-16 alerts the operators that a boron dilution may be in progress.
- The initial boron concentration is assumed to be 2,100 ppm, which is a conservative maximum value for the initial concentration at the condition of hot full power, with the rods at the insertion limits, and no xenon.

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- The critical boron concentration following reactor trip is assumed to be 1,800 ppm, corresponding to the hot zero power, all rods inserted (minus the most reactive RCCA), no xenon condition. The 300 ppm change from the initial condition noted above is a conservative minimum value.
 - A 1.3-percent minimum shutdown margin is assumed in the analysis.
 - Bounding boron worths of -15 and -5 pcm/ppm are conservatively considered. The larger absolute value maximizes the reactivity insertion rate, while the smaller absolute value minimizes the reactivity insertion rate thereby delaying the time to reach the reactor trip setpoint.

2.8.5.4.5.2.4 Results

The boron dilution analysis demonstrated that all applicable acceptance criteria are met for CPNPP Units 1 and 2. This means that operator action to terminate the dilution flow within 15 minutes from operator notification from Modes 1, 2, 3, 4 and 5 precludes a complete loss of shutdown margin. The results of the boron dilution analysis are provided in Table 2.8.5.4.5-1.

No analysis is presented for Mode 6 operation since dilution during refueling is precluded by administrative controls.

If an unintentional dilution of boron in the RCS does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator sufficient time to determine the cause of the addition and take corrective action before shutdown margin is lost. The acceptance criteria as specified in Licensing Report (LR) subsection 2.8.5.4.5.2.2 are met.

2.8.5.4.5.3 Conclusion

A review of the analyses of the decrease in boron concentration in the reactor coolant due to a CVCS malfunction has been conducted. It is concluded that the analyses have adequately accounted for plant operation at the proposed uprated power level and were performed using acceptable analytical models. It is further concluded that the analyses demonstrate that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of the decrease in boron concentration events. Based on this, Luminant Power has concluded that CPNPP Units 1 and 2 will continue to meet the requirements of GDCs -10, -15, and -26 following implementation of the SPU. Therefore, it is concluded that the SPU is acceptable with respect to the decrease in boron concentration in the reactor coolant due to a CVCS malfunction.

2.8.5.4.5.4 References

1. RXE-94-001-A, "Safety Analysis of the Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5," February 1994.

Table 2.8.5.4.5-1 CVCS Malfunction Boron Dilution Event Results		
Operating Mode	Available Operator Action Time (minutes)	Limit (minutes)
Mode 1 – Manual Rod Control	Unit 1: 54.6 Unit 2: 48.0	15
Mode 1 – Automatic Rod Control	Unit 1: 56.5 Unit 2: 49.8	15
Mode 2	Unit 1: 59.5 Unit 2: 52.5	15
Mode 3	The maximum critical boron concentration is controlled as a function of the plant initial boron concentration to meet a minimum operator action time of 15 minutes	15
Mode 4		15
Mode 5 – Drained		15
Mode 5 – Filled		15
Mode 6	N/A ⁽¹⁾	
Note:		
1. No analysis is presented for Mode 6 operation since boron dilution during refueling is precluded by the Technical Specifications requirements.		

2.8.5.4.6 Spectrum of Rod Ejection Accidents

2.8.5.4.6.1 Regulatory Evaluation

Control rod ejection accidents cause a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. The analysis evaluated the consequences of a control rod ejection accident to determine the potential damage caused to the reactor coolant pressure boundary (RCPB) and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The review covered:

- The description of the causes of the transient and the transient itself
- The initial conditions
- The rod patterns and worths, scram worth as a function of time and reactivity coefficients
- The analytical model
- The core parameters that affect the peak reactor pressure or the probability of fuel rod failure
- The results of the transient analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-28, insofar as it requires that the reactivity control systems are designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

- GDC-28 is described in FSAR Section 3.1.3.9, General Design Criterion 28 – Reactivity Limits.

The maximum positive reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values that prevent rupture of the reactor coolant system (RCS) boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

As described in this FSAR section, the maximum positive reactivity insertion rates for the withdrawal of rod cluster control assemblies (RCCAs) and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCAs and of the chemical and volume control system (CVCS). Technical Specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of FSAR Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in FSAR Section 4.3. The capability of the CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in FSAR Chapter 15.

Assurance of core cooling capability following Condition IV accidents, such as rod ejections or steam line breaks, is given by keeping the RCPB stresses within faulted condition limits as defined by applicable American Society of Mechanical Engineering (ASME) Codes. Structural deformations are also checked and limited to values that do not jeopardize the operation of necessary safety features. FSAR Section 15.4 provides a description of rod ejection events and an analysis of their possible consequences.

2.8.5.4.6.2 Technical Evaluation

The criterion applied to ensure the core remains in a coolable geometry following a rod ejection incident is that the average fuel pellet enthalpy at the hot spot must remain less than 200 cal/gm (360 Btu/lbm). The use of the initial conditions presented in Table 2.8.5.4.6-1 resulted in conservative calculations of the fuel pellet enthalpy. The results of the licensing basis analyses demonstrated that the fuel pellet enthalpy does not exceed 360 Btu/lbm for any of the rod ejection cases analyzed.

Overpressurization of the RCS during a rod ejection event is generically addressed in WCAP-7588, Revision 1-A (Reference 1)

Another applicable acceptance criterion is that fuel melting must be limited to less than the innermost 10 percent of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy at the hot spot is less than the limit of 360 Btu/lbm. Conservative fuel melt temperatures of 4,900° and 4,800°F were assumed for the hot spot for the beginning-of-life (BOL) and end-of-life (EOL) cases, respectively. These fuel melting temperatures correspond to a specific burnup limit at the hot spot. The peak UO_2 burnup at the hot spot is based on the assembly with the maximum post-ejection F_0 , which is typically a fresh fuel assembly. Therefore, the fuel melting temperatures represent bounding values for the assumed UO_2 burnup at the hot spot. The maximum burnup at the hot spot at BOL and EOL is confirmed to be below these values as part of the reload process. This assumption does not affect the maximum licensed fuel burnup limit. The results of the licensing basis rod ejection analyses demonstrated that the amount of fuel melting was limited to less than 10 percent of the fuel pellet at the hot spot for each of the rod ejection cases.

2.8.5.4.6.2.1 Introduction

This accident is defined as a mechanical failure of a control rod drive mechanism (CRDM) pressure housing resulting in the ejection of the RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals.

A failure of a CRDM housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

- Each full-length CRDM housing is completely assembled and shop tested at 4,100 psig.
- The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head and checked during the hydrotest of the completed RCS.
- Stress levels in the mechanism are not affected by anticipated system transients at power or by the thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress ranges specified in the ASME Boiler and Pressure Vessel Code, Section III, for Class I components.
- The latch mechanism housing and rod travel housing are each a single length of forged type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

A significant amount of margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that the gross failure of the housing will not occur. The joints between the latch mechanism housing and rod housing are threaded joints reinforced by canopy-type rod welds.

In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by the core depletion are compensated by boron changes. Furthermore, the location and grouping of control rod banks are selected during the nuclear design to lessen the severity of an RCCA ejection accident. Therefore, if an RCCA is ejected from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur. The position of all of the RCCAs is continuously indicated in the Control Room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one control rod assembly deviates from its bank. There are low and low-low level insertion alarm circuits for each bank. The control rod position monitoring and alarm systems are described in WCAP-7588 (Reference 1).

2.8.5.4.6.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Input parameters for the analysis were conservatively selected on the basis of values calculated for this type of core. The most important parameters are discussed below. Table 2.8.5.4.6-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for the ejected rod worths and hot channel factors were calculated using either three-dimensional (3-D) static methods or a synthesis of one-dimensional (1-D) and two-dimensional (2-D) calculations. Standard nuclear design codes were used in the analysis. No credit was taken for the flux-flattening effects of reactivity feedback. The calculation was performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. The analysis assumed adverse xenon distributions to provide worst-case results.

Appropriate margins were added to the ejected rod worth and hot channel factors to account for any calculational uncertainties.

Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values of approximately 0.75 percent at BOL and 0.40 percent at EOL. The ejected rod accident is sensitive to β_{eff} if the ejected rod worth is equal to or greater than β_{eff} , as in the zero-power transients. In order to allow for future fuel cycle flexibility, conservative estimates of β_{eff} of 0.55 percent at beginning of cycle and 0.44 percent at end of cycle were used in the analysis.

Reactivity Weighting Factor

The largest temperature rises, and therefore the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions.

Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single-channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape. In this analysis, a 1-D (axial) spatial kinetics method was employed. Therefore, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting was applied to the moderator feedback. A conservative radial weighting factor was applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to 3-D analysis.

Moderator and Doppler Coefficient

The critical boron concentrations at BOL and EOL were adjusted in the nuclear code in order to obtain moderator density coefficient curves that were conservative when compared to the actual design conditions for the plant. As discussed above, no weighting factor was applied to these results. The moderator temperature coefficients (MTCs) that were modeled are +5 pcm/°F at zero-power nominal T_{avg} and 0 pcm/°F at full-power T_{avg} for the BOL cases. For the EOL cases the applicable zero-power MTC was -16.817 pcm/°F and the full-power MTC was -22.920 pcm/°F.

The Doppler reactivity defect as a function of power level was adjusted in the nuclear code to a conservative design value using a Doppler weighting factor of 1.0. The Doppler weighting factor was increased under accident conditions, as discussed above.

Heat Transfer Data

The FACTRAN code (Reference 2), used to determine the hot spot transient, contains standard curves of thermal conductivity versus fuel temperature. During the transient, the peak centerline fuel temperature is nearly independent of the gap conductance. The cladding temperature is, however, strongly dependent on the gap conductance and is highest for high gap conductance. For conservatism, a low initial gap heat transfer coefficient was used at the beginning of the transient to maximize the initial fuel temperature, and a high gap heat transfer coefficient value of 10,000 Btu/hr-ft² was used for the remainder of the transient to maximize the cladding temperature. This high gap heat transfer coefficient corresponds to a negligible gap resistance, and a further increase would have essentially no effect on the rate of heat transfer.

Coolant Mass Flow Rates

When the core is operating at full power, all four coolant pumps are always operational. For zero power conditions, the system was conservatively assumed to be operating with two pumps. The principal effect of operating at reduced flow is to reduce the film boiling heat transfer coefficient. This resulted in higher peak cladding temperatures, but did not affect the peak centerline fuel temperature. Reduced flow also lowers the critical heat flux. However, since departure from nucleate boiling (DNB) was always assumed at the hot spot, and since the heat flux rose very rapidly during the transient, this produced only second-order changes in the cladding and centerline fuel temperatures.

Trip Reactivity Insertion

The trip reactivity insertion was assumed to be 4.0 percent Δk from hot full power (HFP) and 2.0 percent Δk from hot zero power (HZP), including the effect of one stuck RCCA. These values were also reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after reaching the power range high neutron flux trip setpoint. It was assumed that insertion to dashpot did not occur until 2.7 seconds after the rods began to fall. The time delay

to full insertion, combined with the 0.5 second trip delay, conservatively delayed insertion of shutdown reactivity into the core.

Due to the extremely low probability of an RCCA ejection accident, this event is classified as a Condition IV event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. As such, some fuel damage is considered an acceptable consequence.

The real physical limits of this accident are that the rod ejection event and any consequential damage to either the core or the RCS must not prevent long-term core cooling. More specific and restrictive criteria are applied to ensure that there is no fuel dispersal in the coolant and that gross lattice distortion or severe shock waves do not occur. In view of the above experimental results and the conclusions of WCAP-7588, Revision 1-A (Reference 1), the applied criteria are:

- Average fuel pellet enthalpy at the hot spot must remain below 200 cal/gm for irradiated fuel. This bounds non-irradiated fuel which has a slightly higher enthalpy limit.
- Peak reactor coolant pressure must be less than that which could cause RCS stresses to exceed the faulted-condition stress limits (Note: the peak pressure aspects of the rod ejection transient are addressed generically in Reference 1).
- Fuel melting is limited to less than the innermost 10 percent of the pellet volume at the hot spot even if the average fuel pellet enthalpy at the hot spot is below the 200 cal/gm fuel enthalpy limit.

2.8.5.4.6.2.3 Description of Analyses and Evaluations

This section describes the models used in the analysis of the rod ejection accident. Only the initial few seconds of the power transient are discussed, since the long-term considerations are the same as those for a small loss-of-coolant accident (LOCA).

The calculation of the RCCA ejection transient was performed in two stages: first an average core channel calculation, and then a hot spot calculation. The average core calculation used spatial neutron-kinetics methods to determine the average power generation with time including the various total core feedback effects, that is, Doppler reactivity and moderator reactivity. Enthalpy and temperature transients at the hot spot were then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback was conservatively assumed to continue throughout the transient. A detailed discussion of the method of analysis can be found in Reference 1.

Average Core

The spatial-kinetics computer code TWINKLE (Reference 3) was used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups

and up to 2,000 spatial points. The computer code includes a detailed, multi-region, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. This analysis used the code as a 1-D axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension was missing, it was still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal heat flux times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The average core energy addition, calculated as described above, was multiplied by the appropriate hot channel factors. The hot spot analysis used the detailed fuel and cladding transient heat transfer computer code FACTRAN (Reference 2). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO₂ fuel rod, and the heat flux at the surface of the rod, using the nuclear power versus time and local coolant conditions as input. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power distribution was assumed within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (Reference 4) to determine the film boiling coefficient after DNB. The Bishop-Sandberg-Tong correlation was conservatively used assuming zero bulk fluid quality. The DNB heat flux was not calculated. Instead, the code was forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient could be calculated by the code. However, it was adjusted to force the full-power, steady-state temperature distribution to agree with fuel heat transfer design codes.

Reactor Protection

The protection for this accident, as explicitly modeled in the analysis, is provided by the power range high neutron flux trip (high and low settings). The power range high neutron flux positive rate trip complements the high and low flux trip functions to ensure that the criteria are met for rod ejection from partial power.

2.8.5.4.6.2.4 Spectrum of Rod Ejection Accidents Results

The results of the analyses performed for the rod ejection event, which cover BOL and EOL conditions at hot full power and HZP for both CPNPP Units 1 and 2, are discussed below.

Beginning of Cycle, Zero Power

The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.75-percent Δk and 11.0-percent Δk , respectively. The peak hot spot average fuel pellet enthalpy reached 205.7 Btu/lbm (114.3 cal/gm). The peak fuel centerline temperature never reached the BOL melt temperature of 4,900°F. Therefore, no fuel melting is predicted.

Beginning of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.24-percent Δk and 5.5-percent Δk , respectively. The peak hot spot average fuel pellet enthalpy reached 290.9 Btu/lbm (161.6 cal/gm). The peak fuel centerline temperature reached the BOL melt temperature of 4,900°F. However, fuel melting remained well below the limiting criterion of 10 percent of total pellet volume at the hot spot.

End of Cycle, Zero Power

The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.84-percent Δk and 26.0-percent Δk , respectively. The peak hot spot average fuel pellet enthalpy reached 250.0 Btu/lbm (138.9 cal/gm). The peak fuel centerline temperature never reached the EOL melt temperature of 4,800°F. Therefore, no fuel melting is predicted.

End of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.25-percent Δk and 6.0-percent Δk , respectively. The peak hot spot average fuel pellet enthalpy reached 283.5 Btu/lbm (157.5 cal/gm). The peak fuel centerline temperature reached the EOL melting temperature of 4,800°F. However, fuel melting remained well below the limiting criterion of 10 percent of total pellet volume at the hot spot.

A summary of the parameters used in the rod ejection analyses, and the analyses results, are presented in Table 2.8.5.4.6-1. The sequence of events for all four cases is presented in Table 2.8.5.4.6-2. Figure 2.8.5.4.6-1 shows the plot results for the BOL/HZP case and Figure 2.8.5.4.6-2 shows the BOL/HFP plot results. The EOL/HZP and EOL/HFP plot results are presented in Figures 2.8.5.4.6-3 and 2.8.5.4.6-4, respectively.

A detailed calculation of the pressure surge for an ejected rod worth of 1 dollar at BOL HFP indicates that the peak pressure did not exceed that which would cause the reactor pressure vessel stress to exceed the faulted condition stress limits (Reference 1). Since the severity of the present analysis did not exceed the worst-case analysis, the accident for this plant will not result in an excessive pressure rise or further adverse effects on the RCS.

2.8.5.4.6.2.5 Results

Despite the conservative assumptions, the analyses indicate that the described fuel and cladding limits were not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure did not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. Generic analyses demonstrated that the fission product release as a result of fuel rods entering DNB was limited to less than 10 percent of the fuel rods in the core.

The results and conclusions of the analyses performed for the rupture of a CRDM housing RCCA ejection support operation up to the uprated reactor core power of 3,612 MWt.

2.8.5.4.6.3 Conclusion

The Luminant Power review of the analyses of the rod ejection accident concludes that the analyses have adequately accounted for plant operation at the uprated power level and were performed using acceptable analytical models. It is further concluded that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could result in damage to the RCPB greater than limited local yielding, or cause sufficient damage that would significantly impair the capability to cool the core. Based on this, it has been concluded that the plant will continue to meet the requirements of GDC-28 following implementation of the SPU. Therefore, it has been found that the SPU is acceptable with respect to the rod ejection accident.

2.8.5.4.6.4 References

1. WCAP-7588, Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Special Kinetics Methods," January 1975.
2. WCAP-7908, "FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO₂ Fuel Rod," December 1989.
3. WCAP-7979, "TWINKLE, A Multi-Dimensional Neutron Kinetics Computer Code," January 1975.
4. ASME 65-HT-31, "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," August 1965.

Table 2.8.5.4.6-1

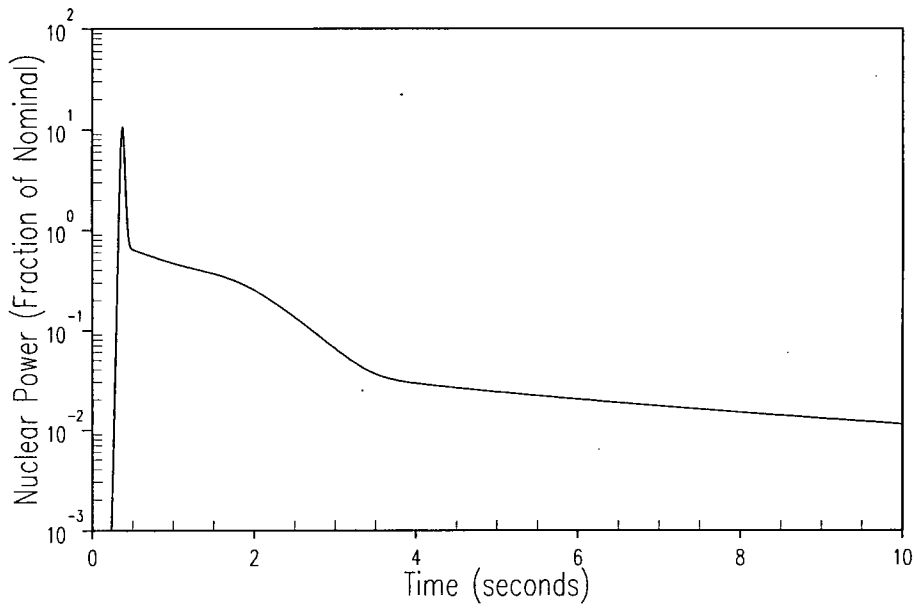
Parameters and Results of the Limiting RCCA Ejection Analyses

	Beginning of Cycle HEP	Beginning of Cycle HEP	End of Cycle HEP	End of Cycle HEP
Initial Reactor Core Power Level (MWt)	3,634	0	3,634	0
Ejected Rod Worth (% Δk)	0.24	0.75	0.25	0.84
Delayed Neutron Fraction (%)	0.55	0.55	0.44	0.44
Feedback Reactivity Weighting	1.2927	2.0079	1.3549	3.7649
Trip Reactivity (%Δk)	4.0	2.0	4.0	2.0
F ₀ Before Rod Ejection	2.50	--	2.50	--
F ₀ After Rod Ejection	5.5	11.0	6.0	26.0
Number of Operational Pumps	4	2	4	2
Maximum Fuel Pellet Average Temperature (°F)	3,754	2,797	3,674	3,304
Maximum Fuel Centerline Temperature (°F)	4,913	3,270	4,825	3,756
Maximum Cladding Average Temperature (°F)	2,121	2,085	2,073	2,503
Maximum Fuel Stored Energy (cal/gm)	161.6	114.3	157.5	138.9
Maximum Fuel Melt at the Hot Spot (%)	0.04	0.00	0.23	0.00

Table 2.8.5.4.6-2

Time Sequence of Events – RCCA Ejection

Event	Time (sec)	
	BOL HFP	EOL HFP
Initiation of Rod Ejection	0.0	0.0
Power Range High Neutron Flux Setpoint Reached	0.05	0.04
Peak Nuclear Power Occurs	0.13	0.13
Rods Begin to Fall	0.55	0.54
Peak Fuel Average Temperature Occurs	2.25	2.25
Peak Cladding Temperature Occurs	2.30	2.31
	BOL HZP	EOL HZP
Initiation of Rod Ejection	0.0	0.0
Power Range High Neutron Flux Setpoint Reached	0.31	0.19
Peak Nuclear Power Occurs	0.38	0.23
Rods Begin to Fall	0.81	0.69
Peak Fuel Average Temperature Occurs	2.52	1.81
Peak Cladding Temperature Occurs	2.38	1.58



— Fuel Centerline
 - - - Fuel Average
 ····· Clad Outer

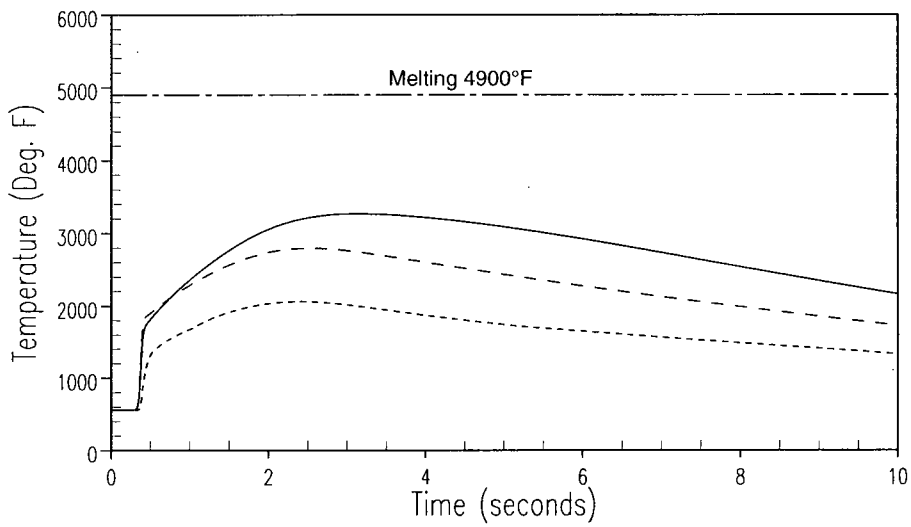
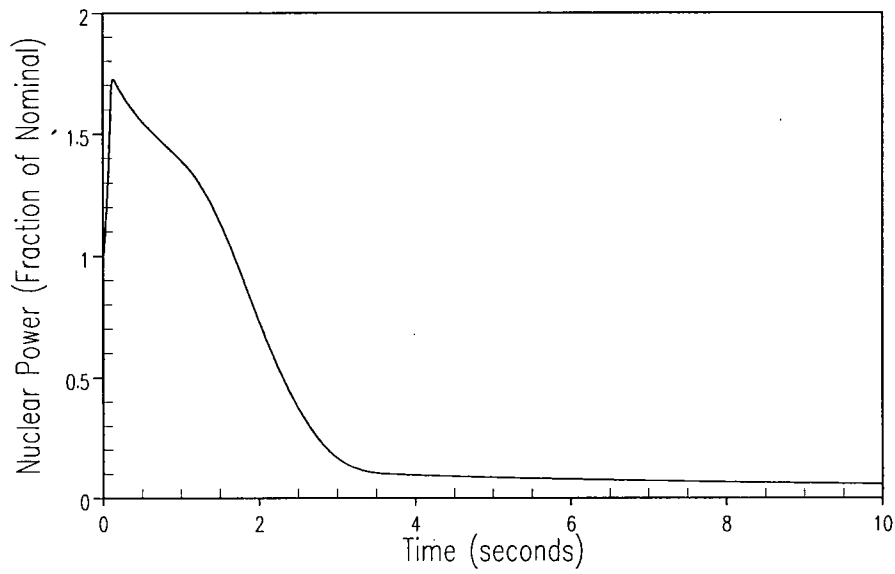


Figure 2.8.5.4.6-1 Rod Ejection – BOL/HZP Case



— Fuel Centerline
 - - - Fuel Average
 ····· Clad Outer

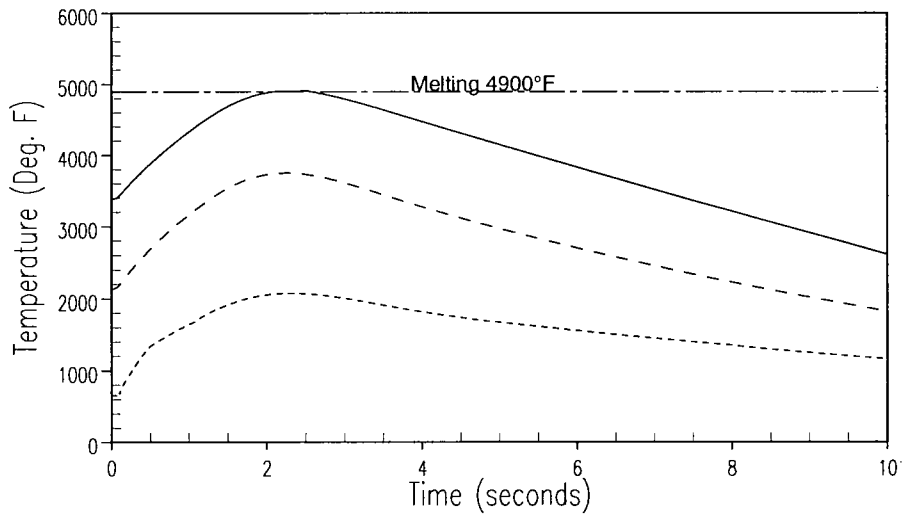
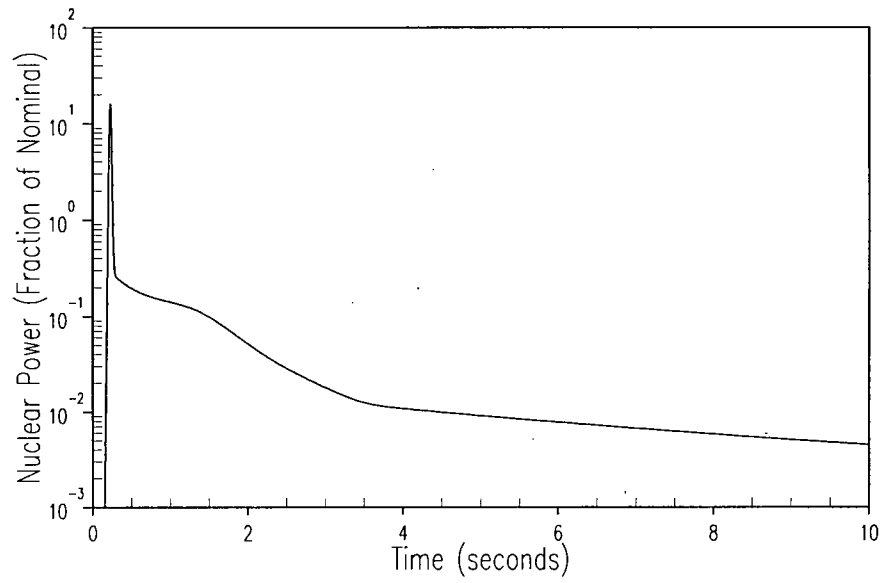


Figure 2.8.5.4.6-2 Rod Ejection – BOL/HFP Case



— Fuel Centerline
 - - - Fuel Average
 - · - · - Clad Outer

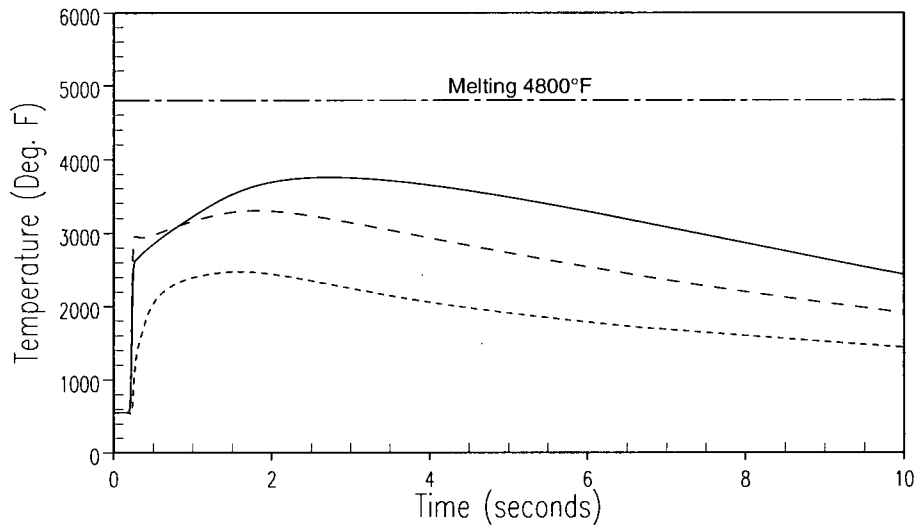
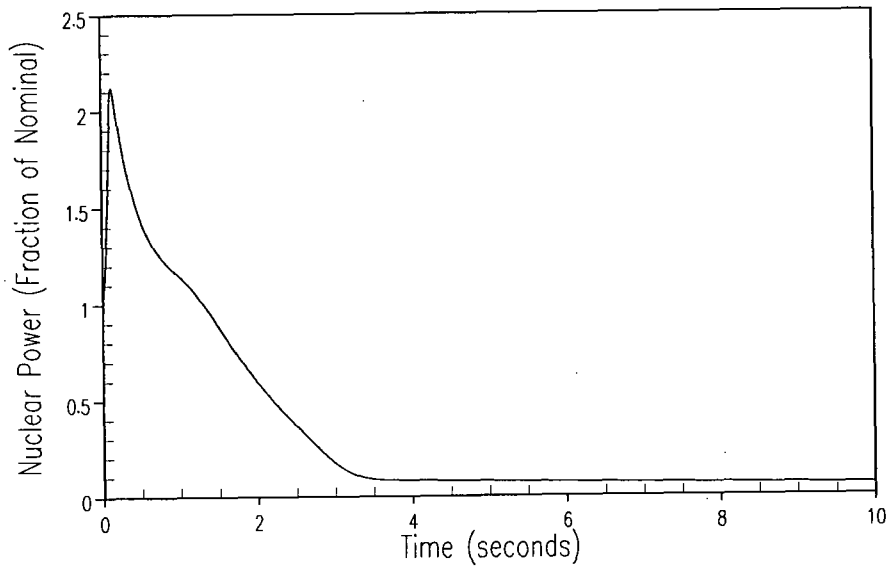


Figure 2.8.5.4.6-3 Rod Ejection – EOL/HZP Case



— Fuel Centerline
 - - - Fuel Average
 ····· Clad Outer

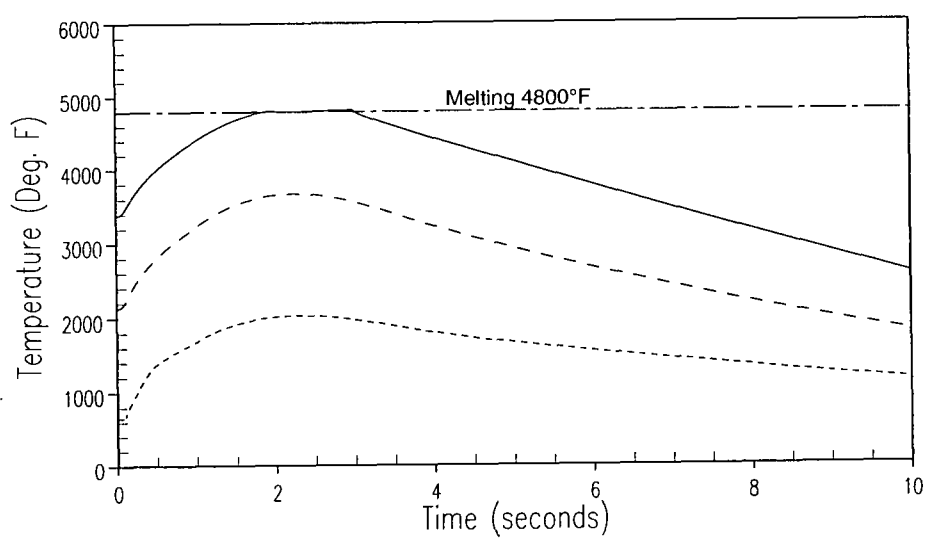


Figure 2.8.5.4.6-4 Rod Ejection – EOL/HFP Case

2.8.5.5 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases in Reactor Coolant Inventory

2.8.5.5.1 Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause an unplanned increase in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or over pressurization of the reactor coolant system (RCS). Alternatively, a power level decrease and depressurization may result. An increase in the RCS inventory as a result of safety injection (SI) flow may lead to the potential filling of the pressurizer. The reactor protection and safety systems are designed to mitigate these events. The review covered:

- The description of the causes of the transient and the transient itself
- The postulated initial core and reactor conditions
- The operator actions for event mitigation
- The analytical model used for analyses
- The sequence of events
- The results of the transient analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including anticipated operational occurrences (AOOs).
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation.
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP's design relative to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I) and during transient conditions arising from occurrences of moderate frequency (Condition II).
2. To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.
3. To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

FSAR Chapter 4 discusses the design bases and the design evaluation of the reactor components, including the fuel, reactor vessel internals, and the reactivity control systems. Details of the control and protection system instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the accident analysis of FSAR Chapter 15, which show that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and of integrated hydrostatic testing of assembled components.

FSAR Chapter 5 discusses the RCS design.

- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control methods are provided. These are rod control cluster assemblies (RCCA) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the full-length control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and AOOs, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCA are presented in FSAR Chapter 4. The operation is discussed in FSAR Chapter 7. The means of controlling the boric acid concentration are described in FSAR Chapter 9. Performance analyses under accident conditions are included in FSAR Chapter 15.

FSAR Section 15.5 addresses the impact of an increase in reactor coolant inventory, while FSAR Appendices 4A and 4B discuss the results of the latest evaluations for the inadvertent operation of the ECCS system during power operation for Units 1 and 2, respectively. It is concluded that all acceptance criteria are satisfied.

2.8.5.5.2 Technical Evaluation

The following events, which could result in an increase in reactor coolant inventory for a pressurized water reactor (PWR) are presented in this section:

1. Inadvertent operation of the ECCS during power operation
2. CVCS malfunction that increases reactor coolant inventory

2.8.5.5.2.1 Inadvertent Operation of the Emergency Core Cooling System During Power Operation

2.8.5.5.2.1.1 Introduction

An inadvertent actuation of the ECCS at power event results in an increase in RCS inventory leading to the potential filling of the pressurizer. Operator error or a spurious electrical actuating signal could cause the event.

Following the actuation signal, the SI system is actuated, which results in borated water being pumped into the cold leg of each RCS loop. Normally, an SI actuation signal results in an immediate automatic reactor trip, which in turn generates a turbine trip. However, even without an immediate reactor trip, the reactor will experience a negative reactivity excursion as a result of the injected borated water. This negative reactivity results in a decrease in reactor power.

In manual rod control, the primary-to-secondary system power mismatch causes a drop in coolant temperature and a contraction of the reactor coolant. Assuming an immediate reactor trip signal is not received, the RCS responds with a decrease in pressurizer pressure and water level and the turbine load will decrease due to the effect of reduced steam pressure once the turbine throttle valves are fully open. The decrease in RCS pressure results in an increase in SI flow associated with the SI pump performance characteristics.

In automatic rod control, RCCA withdrawal may compensate for the above effects as the control system responds to maintain programmed T_{avg} . Once the rods have been fully withdrawn, the event continues as described for operation in manual rod control.

The inadvertent ECCS actuation at power event is performed to demonstrate that sufficient time is available for the appropriate operator actions to be taken to preclude a pressurizer water-solid condition (and avoid actuation of the pressurizer relief and safety valves). In the design-basis analysis of the inadvertent ECCS transient, operator actions that are credited for the mitigation of the transient are the control of T_{avg} by manually dumping steam through steam generator atmospheric relief valves (ARVs), the manual control of the Pressurizer Pressure and Level Control System, and ultimately, securing ECCS flow.

2.8.5.5.2.1.2 Input Parameters, Assumptions, and Acceptance Criteria

The following assumptions were made in the inadvertent ECCS analyses:

- An initial nuclear steam supply system (NSSS) power of 3,628 MWt plus 0.6-percent power uncertainty was assumed.
- A range of the full-power reactor vessel average coolant temperature (T_{avg}) of 585.4° to 589.2°F was considered in the analysis.
- The initial pressurizer water level is assumed to be 65-percent level span (the programmed full-power value of 60 percent plus 5-percent span uncertainty). A high initial pressurizer water level minimizes the initial margin to a water-solid pressurizer condition.
- The initial pressurizer pressure is assumed to be 2,220 psia (the nominal value of 2,250 psia minus 30 psi uncertainty). A lower initial RCS pressure allows a higher ECCS flow.
- The pressurizer heaters and sprays are assumed to function because their operation generates a more limiting condition with respect to filling the pressurizer.
- An immediate reactor trip on the SI actuation signal and a turbine trip derived from the reactor trip are assumed because this limits the primary-to-secondary heat transfer rate, thus minimizing the magnitude of the initial reactor coolant shrinkage.

-
- The first operator action assumed is the initiation of the opening of at least three of the four steam generator atmospheric relief valves within 7 minutes and 30 seconds after the event initiation to control T_{avg} to the no-load temperature of 557°F. (The valves' stroke times are included in the analysis.)
 - The operators are then assumed to secure ECCS within 13 minutes after the event initiation. Operator action is also assumed to take positive control of the Pressurizer Pressure and Level Control System. The analysis assumes positive control of the pressurizer heaters coincident with securing ECCS within 13 minutes.

Based on its frequency of occurrence, the inadvertent ECCS accident is considered a Condition II event as defined by the American Nuclear Society (ANS). The following items summarize the acceptance criteria associated with this event:

- Fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR safety analysis limit.
- Pressures in the RCS and MSS are maintained below 110 percent of the design pressures.
- An incident of moderate frequency does not generate a more serious plant condition without other faults occurring independently.

With respect to the overpressure evaluation, the inadvertent ECCS actuation at power event is bounded by the loss of load/turbine trip (LOL/TT) event, discussed in LR Section 2.8.5.2.1, in which assumptions are made to conservatively maximize the RCS and main steam system (MSS) pressure transients. For the inadvertent ECCS actuation at power event, turbine trip occurs following reactor trip, whereas for the LOL/TT event, the turbine trip is the initiating fault. Therefore, the primary-to-secondary power mismatch and resultant RCS and MSS heatup and pressurization transients are always more severe for the LOL/TT event. For this reason, it is not necessary to calculate the maximum RCS or MSS pressures for the inadvertent ECCS actuation at power event.

This event is non-limiting with respect to departure from nucleate boiling (DNB) since the conditions resulting from injecting borated water into the RCS are beneficial with respect to DNB. Depending on the control systems in operation, core power and RCS temperatures either remain near the initial nominal conditions or decrease during this event. The RCS flow remains constant throughout the event. A decrease in RCS pressure is the only condition that may occur which would adversely affect DNB. However, for the decrease in RCS pressure that may occur, the effects are more than offset by significant reduction in the power and temperature. The net effect is a DNBR that remains near the initial DNBR or increases throughout the event.

The major concern from an inadvertent ECCS actuation at power event is the potential to violate the ANS Condition II acceptance criterion where an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. The pressurizer water volume increases for this event as a result of the continuous safety injection

flow. Operator actions are required to initiate the opening of the ARVs, and to control filling of the pressurizer, thus demonstrating that this ANS Condition II event does not propagate to a more serious plant condition. This event is analyzed to demonstrate that sufficient time is available for the appropriate operator actions to be taken to preclude a pressurizer water-solid condition.

2.8.5.5.2.1.3 Description of Analyses and Evaluations

The inadvertent ECCS at power event was analyzed using the RETRAN computer code (Reference 1). The RETRAN computer code is a digital computer code, developed to simulate transient behavior in light water reactor systems. The main features of the code include a point kinetics and one-dimensional kinetics model, one-dimensional homogeneous equilibrium mixture thermal-hydraulic model, control system models, two-phase natural convection heat transfer correlations and non-equilibrium pressurizer model. The code computes pertinent plant variables including temperatures, pressures and power level.

2.8.5.5.2.1.4 Results

Table 2.8.5.5-1 presents the sequence of events for the Unit 1 analysis; Table 2.8.5.5-2 presents the sequence of events for the Unit 2 analysis. Figures 2.8.5.5-1 through 2.8.5.5-3 show the transient behavior of the most pertinent plant parameters for Unit 1; Figures 2.8.5.5-4 through 2.8.5.5-6 show the transient behavior of the most pertinent plant parameters for Unit 2. Reactor trip occurs at the event initiation followed by a rapid cooldown of the RCS. The initial coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the RCS heatup, due to residual RCS heat generation, and ECCS injected flow causes an increase in the rate of pressurizer filling. At 7 minutes and 30 seconds (after event initiation), the first operator action is assumed to initiate the opening of three of the four ARVs to control T_{avg} to the no-load temperature of 557°F. This results in a cooldown of the secondary side, and a subsequent cooldown of the primary side resulting in a shrink of the RCS fluid. At 13 minutes (after event initiation), it is assumed operators effectively control the pressurizer from filling. This is modeled by terminating all sources of safety injection flow coincident with turning off the pressurizer heaters.

2.8.5.5.2.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

An increase in reactor coolant inventory that results from the addition of cold, unborated water to the RCS is analyzed in LR subsection 2.8.5.4.5, CVCS malfunction that results in a decrease in boron concentration in the reactor coolant. The analysis of the inadvertent operation of the ECCS during power operation (LR subsection 2.8.5.5) bounds the CVCS malfunction that results in an increase in reactor coolant inventory.

2.8.5.5.3 Conclusion

The analysis of the inadvertent ECCS at power event has been reviewed and Luminant Power has concluded that the analyses have adequately accounted for operation of the plant at the uprated power level and were performed using acceptable analytical models. The evaluation demonstrates that the reactor protection and safety systems will ensure that the specified acceptable fuel design limits are met and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of the inadvertent ECCS event. Based on this, it is concluded that the plant will continue to meet the requirements of GDCs -10, -15, and -26 following implementation of the uprated power level. Therefore, the proposed SPU program is acceptable with respect to the inadvertent ECCS at power event.

2.8.5.5.4 References

1. WCAP-14882 and WCAP-15234, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999 and May 1999, respectively.

Table 2.8.5.5-1	
Time Sequence of Events – Unit 1 Inadvertent ECCS	
Event	Time into Transient (seconds)
Inadvertent SI Actuation	0.0
Reactor Trip due to SI Actuation	0.0
Turbine Trip from Reactor Trip	0.0
Main Steam Safety Valve (MSSV) Actuation	58.1
Operator initiates action to control T_{avg} by opening 3 of 4 ARVs	450.0
Maximum Pressurizer Water Volume Reached	468.0
ARVs Are Fully Opened	490.0
SI Injection Terminated and Heaters Turned Off (Operator Action)	780.0
End of Transient	2,380.0

Table 2.8.5.5-2	
Time Sequence of Events – Unit 2 Inadvertent ECCS	
Event	Time into Transient (seconds)
Inadvertent SI Actuation	0.0
Reactor Trip due to SI Actuation	0.0
Turbine Trip from Reactor Trip	0.0
MSSV Actuation	80.1
Operator initiates action to control T_{avg} by opening 3 of 4 ARVs	450.0
ARVs Are Fully Opened	490.0
SI Injection Terminated and Heaters Turned Off (Operator Action)	780.0
Maximum Pressurizer Water Volume Reached	788.0
End of Transient	2,380.0

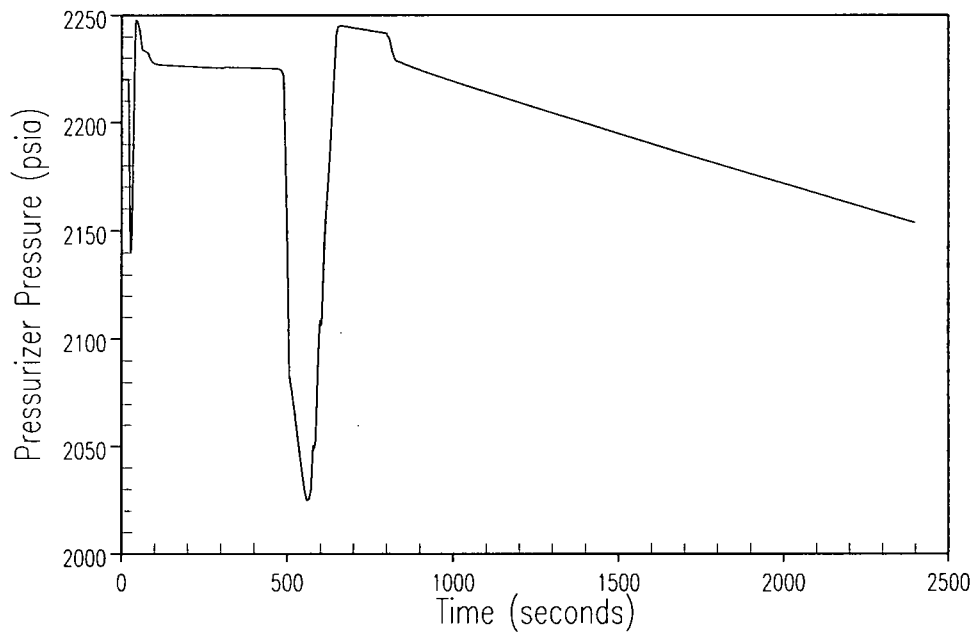
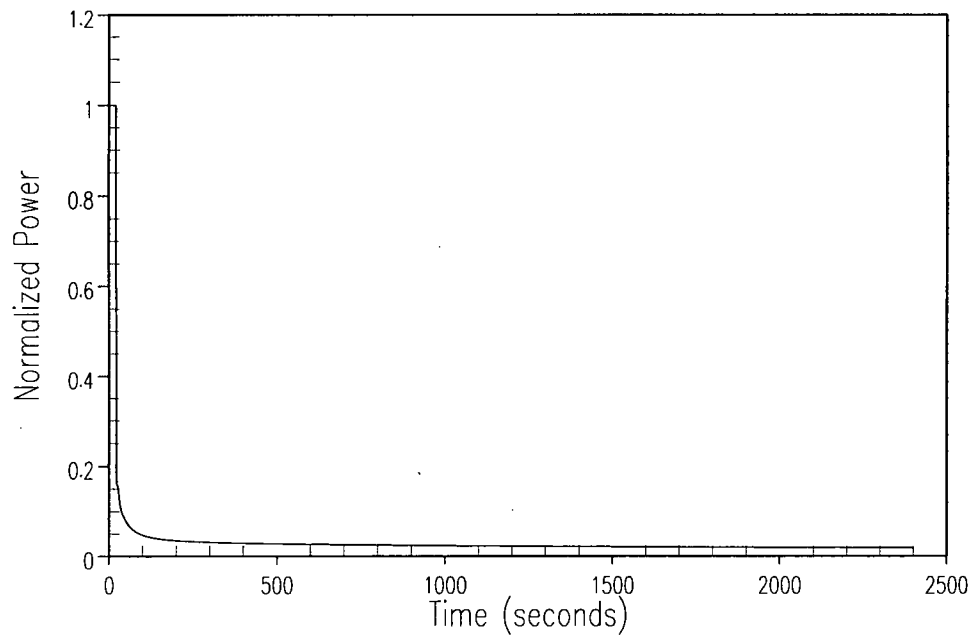


Figure 2.8.5.5-1 Unit 1 Inadvertent ECCS – Power and Pressurizer Pressure Versus Time

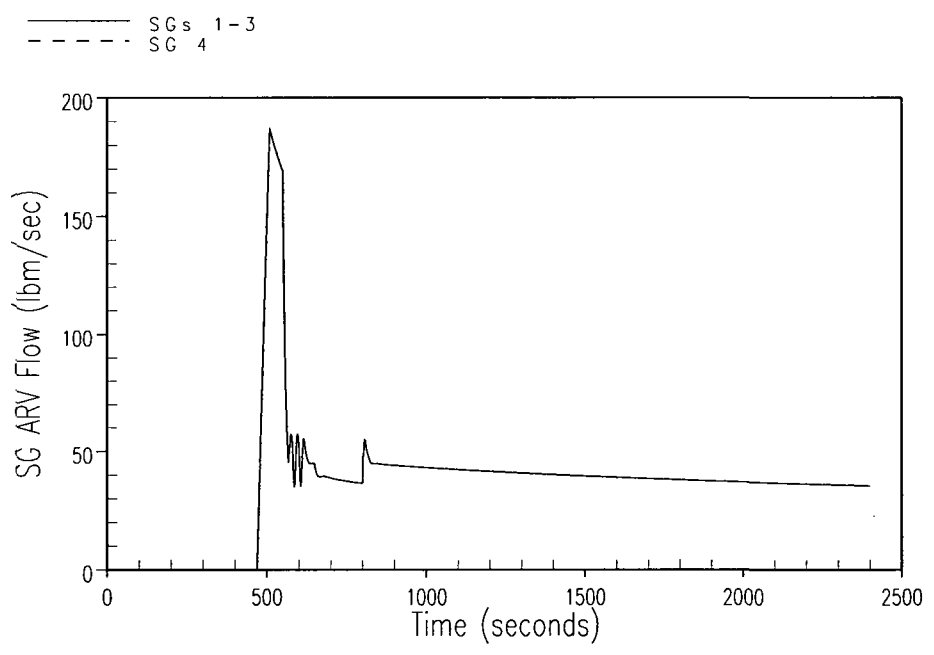
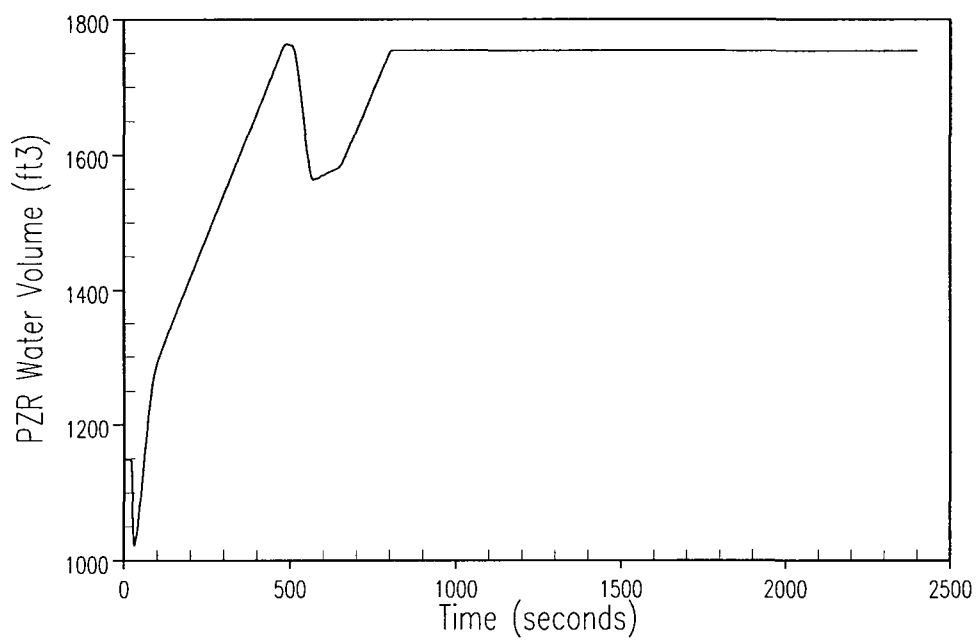


Figure 2.8.5.5-2 Unit 1 Inadvertent ECCS – Pressurizer Volume and ARV Flow Rate Versus Time

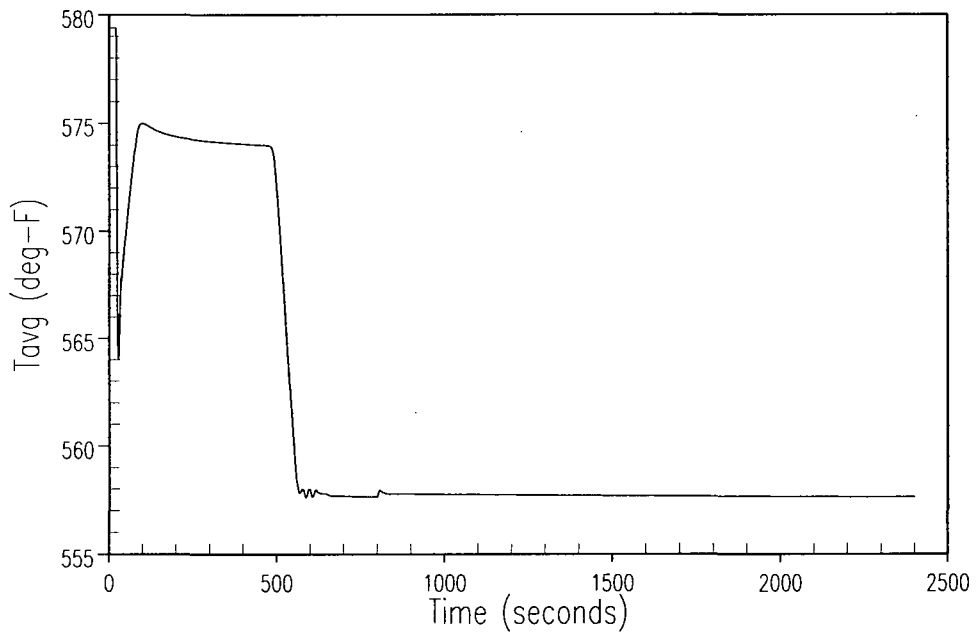
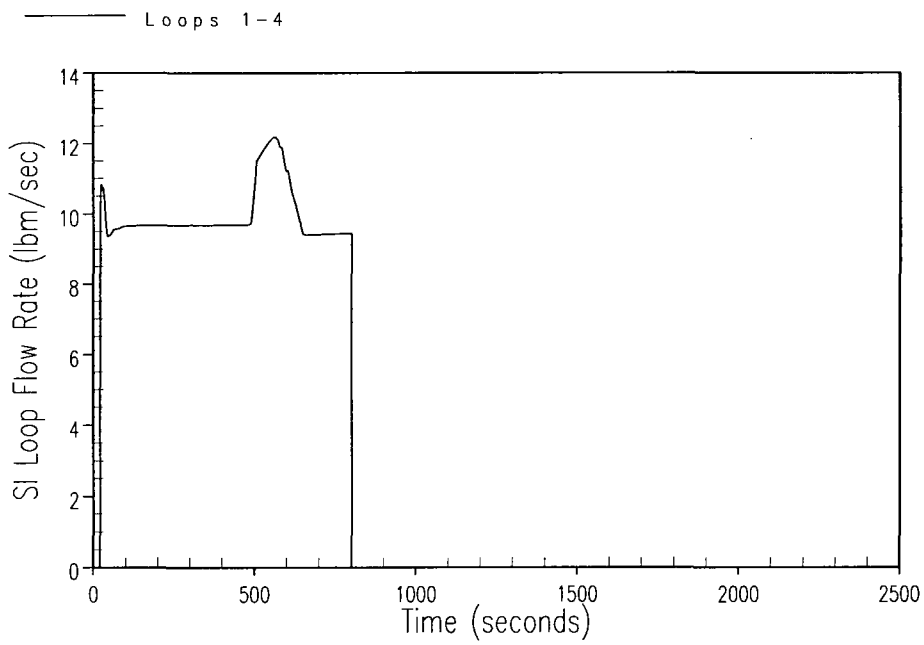


Figure 2.8.5.5-3 Unit 1 Inadvertent ECCS – SI Flow Rates and RCS Average Temperature Versus Time

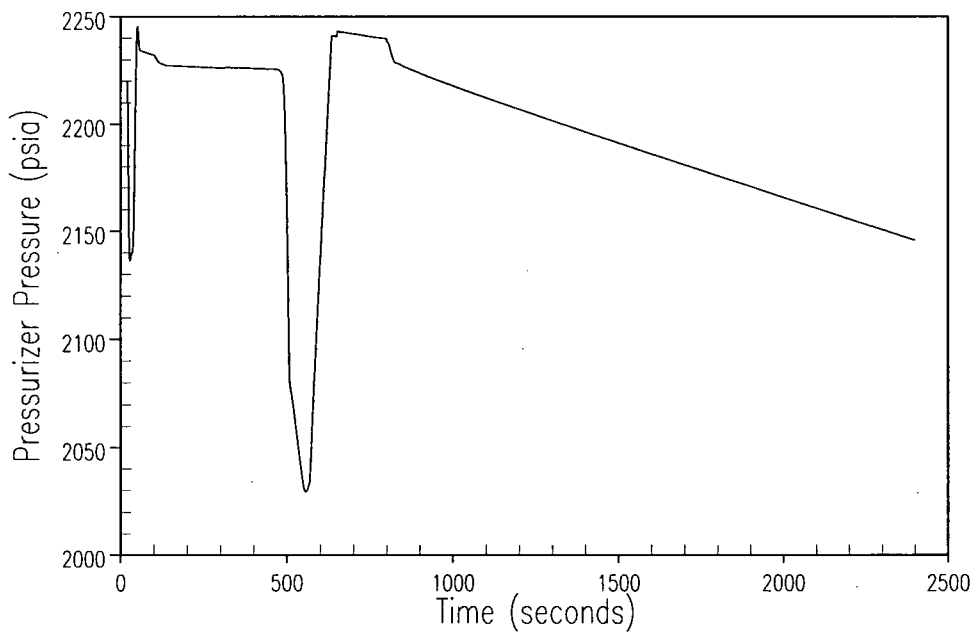
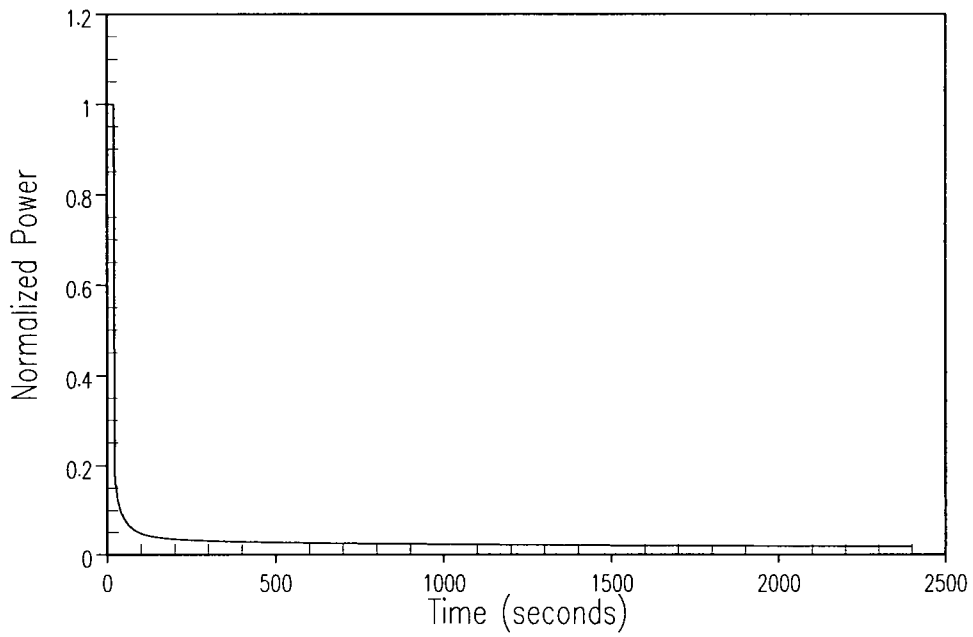


Figure 2.8.5.5-4 Unit 2 Inadvertent ECCS – Power and Pressurizer Pressure Versus Time

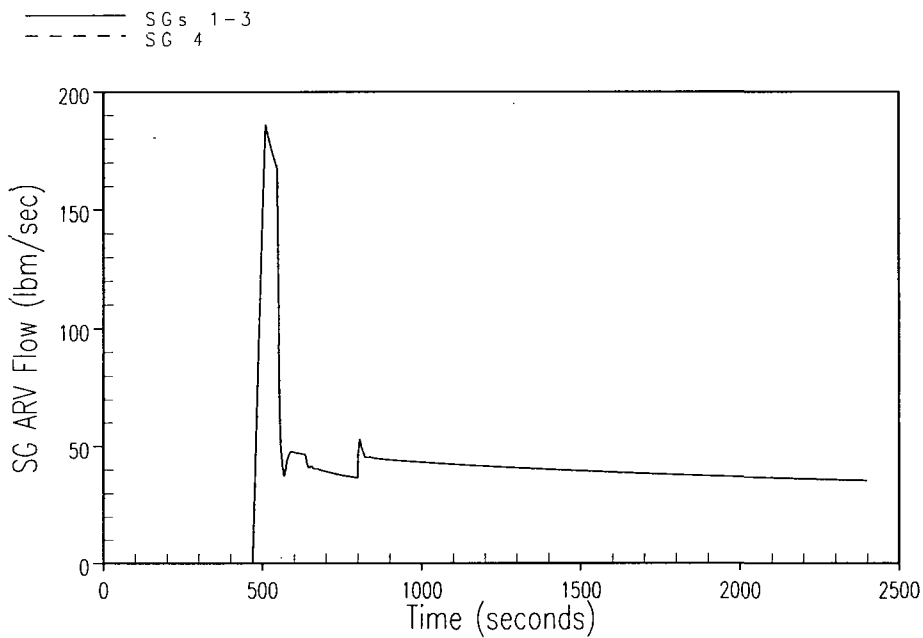
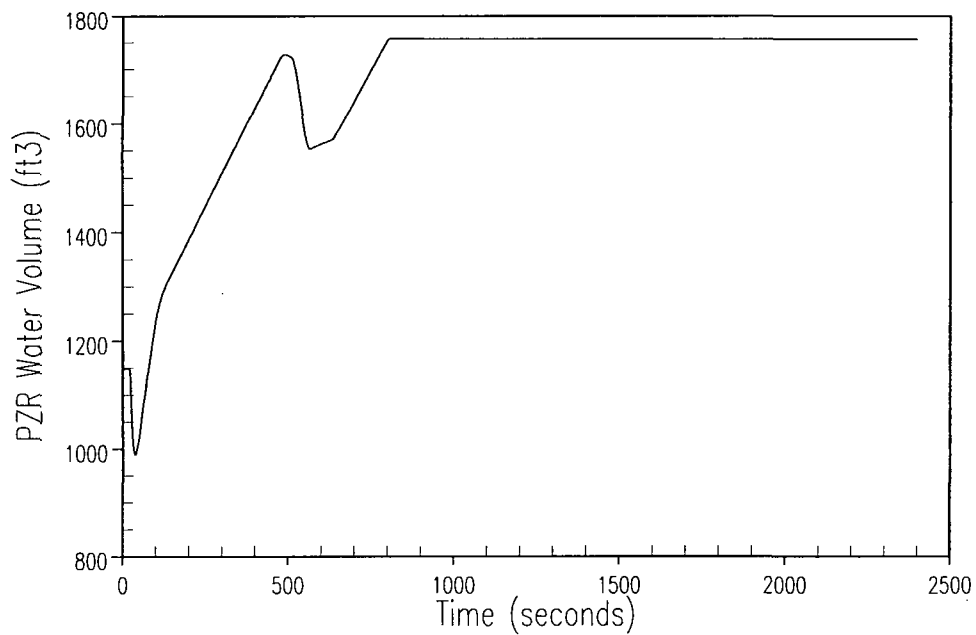


Figure 2.8.5.5-5 Unit 2 Inadvertent ECCS – Pressurizer Volume and ARV Flow Rate Versus Time

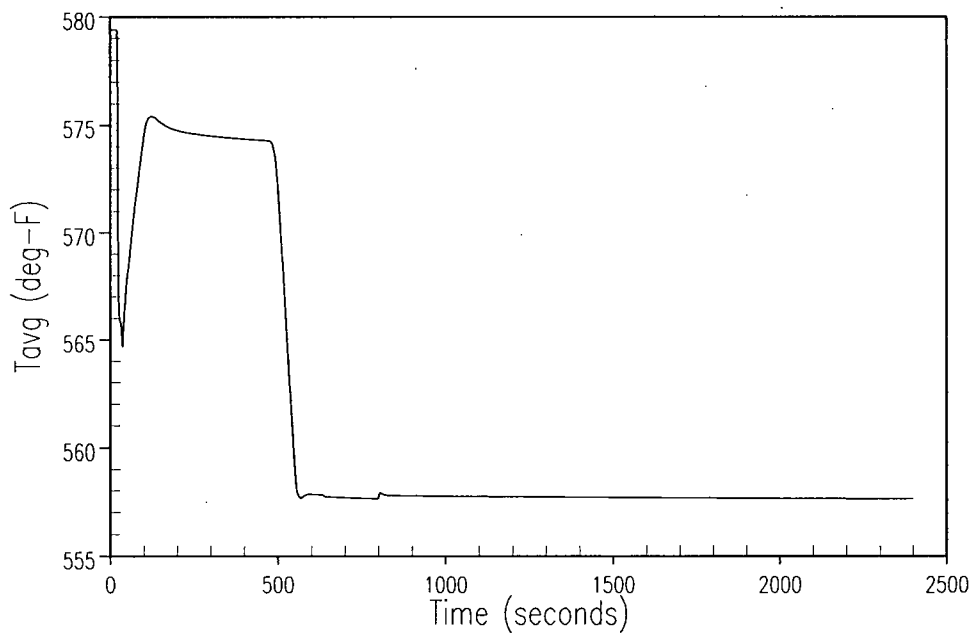
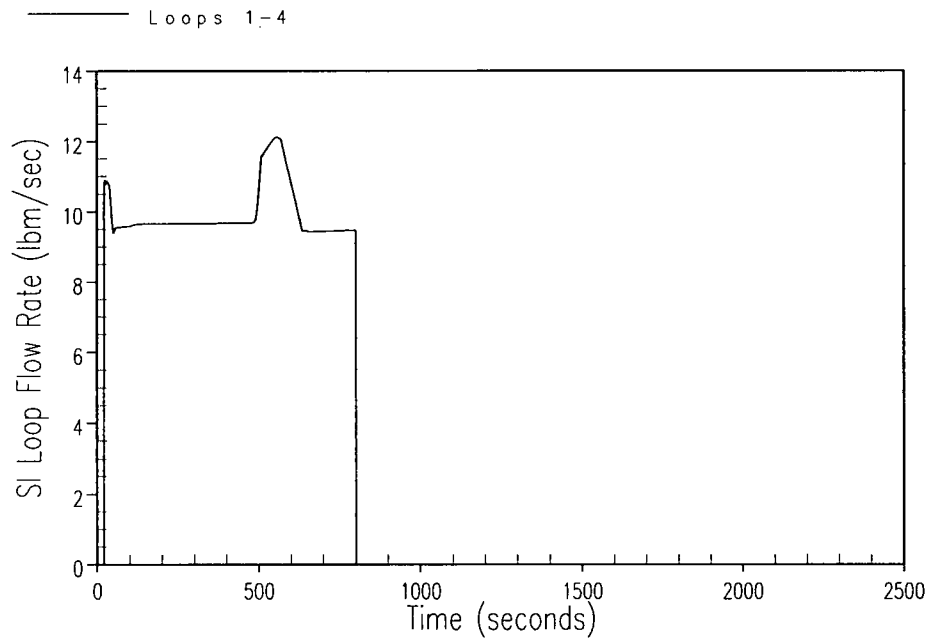


Figure 2.8.5.5-6 Unit 2 Inadvertent ECCS – SI Flow Rates and RCS Average Temperature Versus Time

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Pressurizer Pressure Relief Valve Opening

2.8.5.6.1.1 Regulatory Evaluation

The inadvertent opening of a pressurizer relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system (RCS) pressure. A reactor trip normally occurs due to low RCS pressure. The review covered:

- The sequence of events
- The analytical model used for analyses
- The values of parameters used in the analytical model
- The results of the transient analyses

The acceptance criteria are based on:

- General Design Criterion (GDC)-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences.
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including anticipated operational occurrences.
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under normal conditions of operation, including anticipated operational occurrences, specified acceptable fuel design limits are not exceeded.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR Section 3.1), the design bases of Comanche Peak Nuclear Power Plant (CPNPP) are measured against the Nuclear Regulatory Commission (NRC) GDC for Nuclear Power Plants, Appendix A to 10 CFR 50. The adequacy of the CPNPP design relative to the GDC is discussed in the FSAR Section 3.1.2 and 3.1.3.

Specifically, the adequacy of CPNPP design relative to conformance to:

- GDC-10, Reactor Design, is described in FSAR Section 3.1.2.1.

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II).
2. To ensure return of the reactor to a safe state following infrequent faults (Condition III) with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.
3. To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).

Note that the term “fuel damage” as used in Item 2 above is defined as penetration of the fission product barrier (that is, the fuel rod cladding). Also note that American National Standards Institute ANSI N18.2-1973 expands the definitions of the four conditions enumerated in Items 1 through 3 above.

FSAR Chapter 4 discusses the design bases and design evaluation of reactor components, including the fuel, reactor vessel internals, and reactivity control systems. Details of the control and protection systems instrumentation design and logic are discussed in FSAR Chapter 7. This information supports the accident analyses of FSAR Chapter 15, which show that the acceptable fuel design limits are not exceeded for Conditions I and II occurrences.

- GDC-15, Reactor Coolant System Design, is described in FSAR Section 3.1.2.6.

The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components.

FSAR Chapter 5 discusses the RCS design.

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- GDC-26, Reactivity Control System Redundancy and Capability, is described in FSAR Section 3.1.3.7.

Two reactivity control methods are provided. These are rod control cluster assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the full-length control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCA are presented in FSAR Chapter 4. The operation is discussed in FSAR Chapter 7. The means of controlling the boric acid concentration are described in FSAR Chapter 9. Performance analyses under accident conditions are included in FSAR Chapter 15.

FSAR Section 15.6.1 states that an accidental depressurization of the RCS could occur as result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, and therefore allows a much more rapid depressurization upon opening, the most severe core conditions are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure which could reach the hot leg saturation pressure without reactor protection system intervention. The pressure continues to decrease throughout the transient. With a positive moderator temperature coefficient, the effect of the pressure decrease would be to increase power via the moderator density feedback, but the rod control system (if in automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

Initial operating conditions of maximum core power, maximum reactor coolant average temperature and minimum reactor coolant pressure plus uncertainties are assumed. Plant characteristics and initial conditions are discussed in FSAR Section 15.0.3. FSAR Section 15.6.1.3 concludes that the results of the analysis show that the pressurizer low pressure and the overtemperature N-16 reactor protection system signals provide adequate protection against the RCS depressurization event.

2.8.5.6.1.2 Technical Evaluation

2.8.5.6.1.2.1 Introduction

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief valve. To conservatively bound this scenario, the Westinghouse methodology models the failure of a pressurizer safety valve since a safety valve is sized to relieve approximately twice the steam flow of a pressurizer power-operated relief valve (PORV) and will allow a much more rapid depressurization upon opening. The depressurization resulting from an open safety valve is also much more rapid than would occur from the accidental actuation of pressurizer spray. Therefore, the failure of a pressurizer safety valve yields the most severe core conditions resulting from an accidental depressurization of the RCS. It should be noted that a stuck-open pressurizer safety valve is not an event of moderate frequency as a control system failure would be. A stuck-open safety valve is considered to be a small-break loss-of-coolant accident (LOCA) during which the RCS cannot be isolated, whereas the failure of a PORV can be overridden by the closure of the PORV block valve. Nonetheless, the results of this analysis are shown to comply with the acceptance criteria for an event of moderate frequency.

Initially, the event results in a rapidly decreasing RCS pressure, which could reach hot leg saturation conditions without reactor protection system intervention. If saturated conditions are reached, the rate of depressurization is slowed considerably. However, the pressure continues to decrease throughout the event. The power remains essentially constant throughout the initial stages of the transient.

The reactor may be tripped by the following reactor trip system signals:

- Low pressurizer pressure
- Overtemperature N-16

2.8.5.6.1.2.2 Input Parameters, Assumptions, and Acceptance Criteria

To produce conservative results in calculating the departure from nucleate boiling ratio (DNBR) during the transient, the following assumptions were made:

- The accident was analyzed using the Revised Thermal Design Procedure (RTDP) (Reference 1). Initial core power, pressurizer pressure, and RCS temperature were assumed to be at their nominal values, consistent with steady-state full-power operation. Reactor coolant minimum measured flow was modeled. Uncertainties in initial conditions were included in the DNBR safety analysis limit as described in Reference 1. The initial core power level assumed is 3,612 MWt.
- A zero moderator coefficient of reactivity was assumed. This is conservative for beginning-of-life (BOL) operation in order to provide a conservatively low amount of negative reactivity feedback due to changes in moderator temperature.

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- A small (absolute value) Doppler coefficient of reactivity is assumed, such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator feedback.
 - The spatial effect of voids resulting from local or subcooled boiling was not considered in the analysis with respect to reactivity feedback or core power shape. In fact, it should be noted that the power peaking factors were kept constant at their design values, while the void formation and resulting core feedback effects would result in considerable flattening of the power distribution. Although this would significantly increase the calculated DNBR, no credit was taken for this effect.
 - The analysis performed assumes that the rod control system is in automatic. However, no rod motion occurs during the transient because the conditions do not change enough to demand any rod motion from the rod control system. Therefore, the transient results are identical with or without automatic rod control.

Based on its frequency of occurrence, the accidental depressurization of the RCS accident is considered a Condition II event as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973. The following items summarize the acceptance criteria associated with this event:

- The critical heat flux should not be exceeded. This was met by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.
- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures. Note that since this event is a depressurization event, these limits are not challenged. Both primary and secondary pressures decrease for the entire duration of the event.

2.8.5.6.1.2.3 Description of Analyses and Evaluations

The purpose of this analysis was to demonstrate that the reactor trip system functions and mitigates the consequences of the RCS depressurization event. This analysis is concerned with the transient from initiation through just past the time of reactor trip. With respect to long-term post-accident recovery, it is assumed that operators follow approved plant procedures to bring the plant to a safe post-accident condition.

The accident was analyzed by using the detailed digital computer code RETRAN (Reference 2). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

2.8.5.6.1.2.4 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 2.8.5.6.1-1 through 2.8.5.6.1-8. Figures 2.8.5.6.1-1 and 2.8.5.6.1-2 illustrate the nuclear power transients following the depressurization for Unit 1 and Unit 2, respectively. Nuclear power remains essentially unchanged until the reactor trip occurs on low pressurizer pressure. The pressurizer pressure transients are illustrated in Figures 2.8.5.6.1-3 (Unit 1) and 2.8.5.6.1-4 (Unit 2). Pressure decreases continuously throughout the transient. However, pressure decreases more rapidly after core heat generation is reduced via the reactor trip. Illustrated in Figures 2.8.5.6.1-5 (Unit 1) and 2.8.5.6.1-6 (Unit 2) are the loop average temperature transients. The loop average temperature decreases slowly until the reactor trip occurs. The DNBR decreases initially, but increases rapidly following the reactor trip as demonstrated in Figures 2.8.5.6.1-7 (Unit 1) and 2.8.5.6.1-8 (Unit 2). The DNBR remains above the limit value of 1.61 throughout the transient.

The calculated sequences of events for both units are shown in Table 2.8.5.6.1-1. The calculated minimum DNBR values are provided in Table 2.8.5.6.1-2.

The results of the analysis show that the low pressurizer pressure reactor trip system function provides adequate protection against the RCS depressurization event since the minimum DNBR remains above the safety analysis limit throughout the transient. Therefore, no cladding damage or release of fission products to the RCS is predicted for this event.

The results of the analysis performed for the accidental depressurization of the RCS for the core power level of 3,612 MWt support the implementation of the SPU at CPNPP.

2.8.5.6.1.3 Conclusion

The analysis of the inadvertent opening of a pressurizer pressure relief valve event has been reviewed and Luminant Power has concluded that the analysis has adequately accounted for plant operation at the uprated power level and was performed using acceptable analytical models. It is further concluded that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of this event. Based on this, it is concluded that the plant will continue to meet the current licensing basis requirements with respect to GDCs -10, -15, and -26 following implementation of the SPU. Therefore, the SPU is acceptable with respect to the inadvertent opening of a pressurizer pressure relief valve event.

2.8.5.6.1.4 References

1. WCAP-11397, "Revised Thermal Design Procedure," April 1989.
2. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

Table 2.8.5.6.1-1		
Time Sequence of Events – Accidental Depressurization of the RCS		
Event	Time (seconds) (Unit 1)	Time (seconds) (Unit 2)
Inadvertent Opening of One Pressurizer Safety Valve	0.0	0.0
Low Pressurizer Pressure Reactor Trip Setpoint Reached	42.8	42.4
Rods Begin to Drop	44.8	44.4
Minimum DNBR Occurs	45.5	45.0

Table 2.8.5.6.1-2		
Results - Accidental Depressurization of the RCS		
Minimum DNBR (Unit 1)	Minimum DNBR (Unit 2)	DNBR Safety Analysis Limit
1.923	1.921	1.61

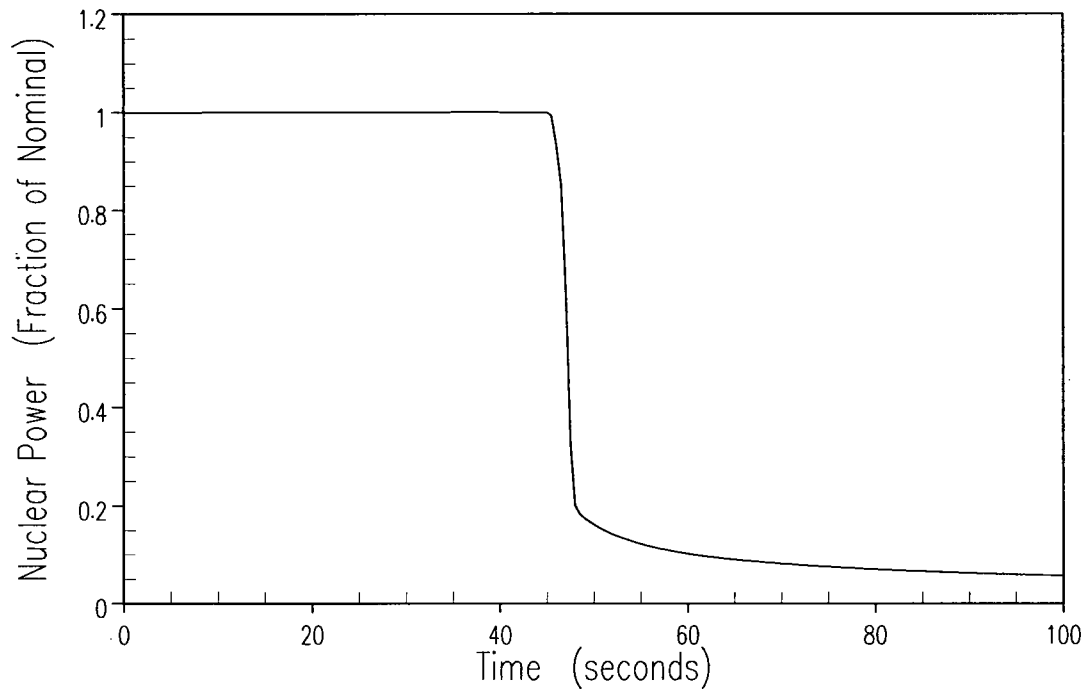


Figure 2.8.5.6.1-1 RCS Depressurization – Nuclear Power Versus Time (Unit 1)

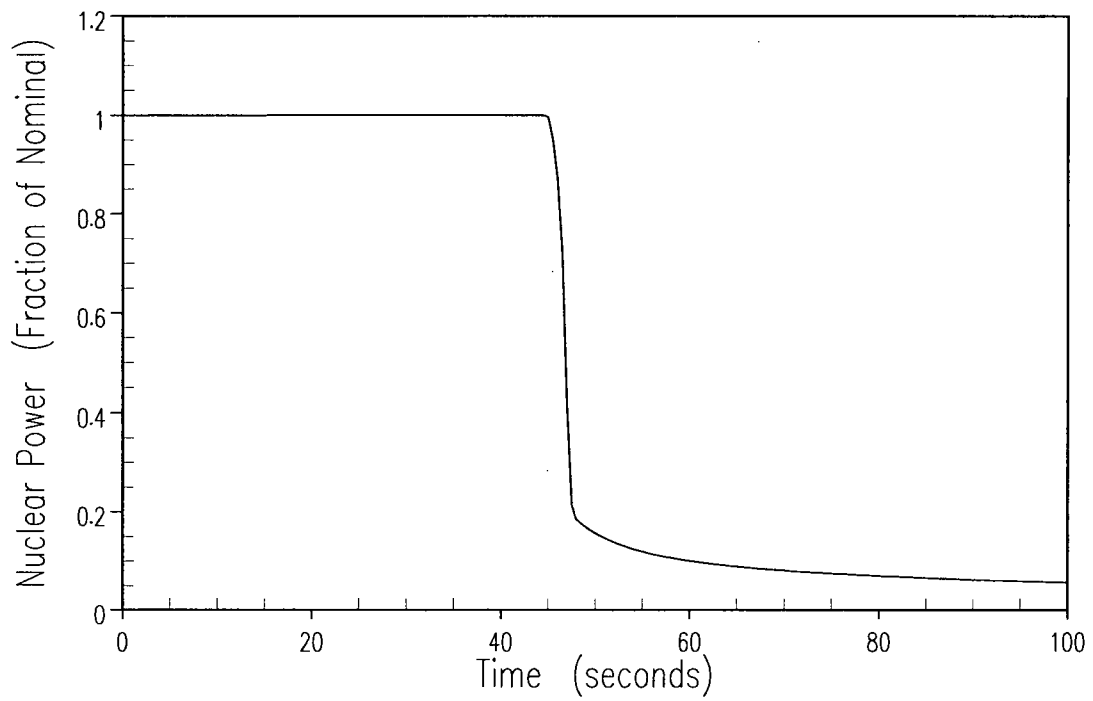


Figure 2.8.5.6.1-2 RCS Depressurization – Nuclear Power Versus Time (Unit 2)

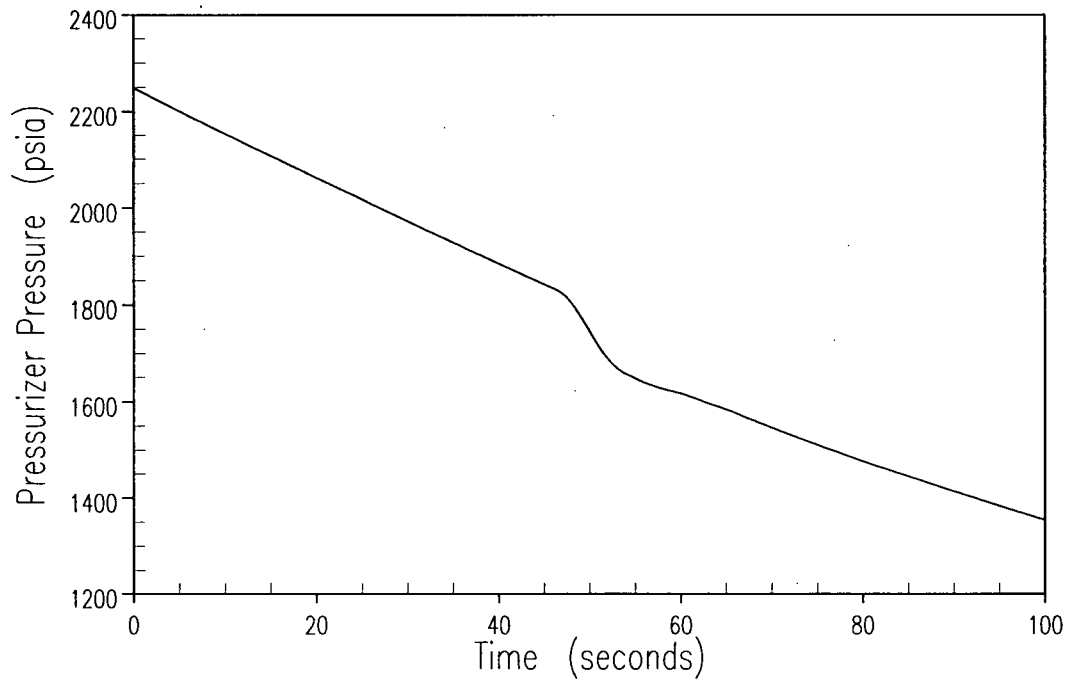


Figure 2.8.5.6.1-3 RCS Depressurization – Pressurizer Pressure Versus Time (Unit 1)

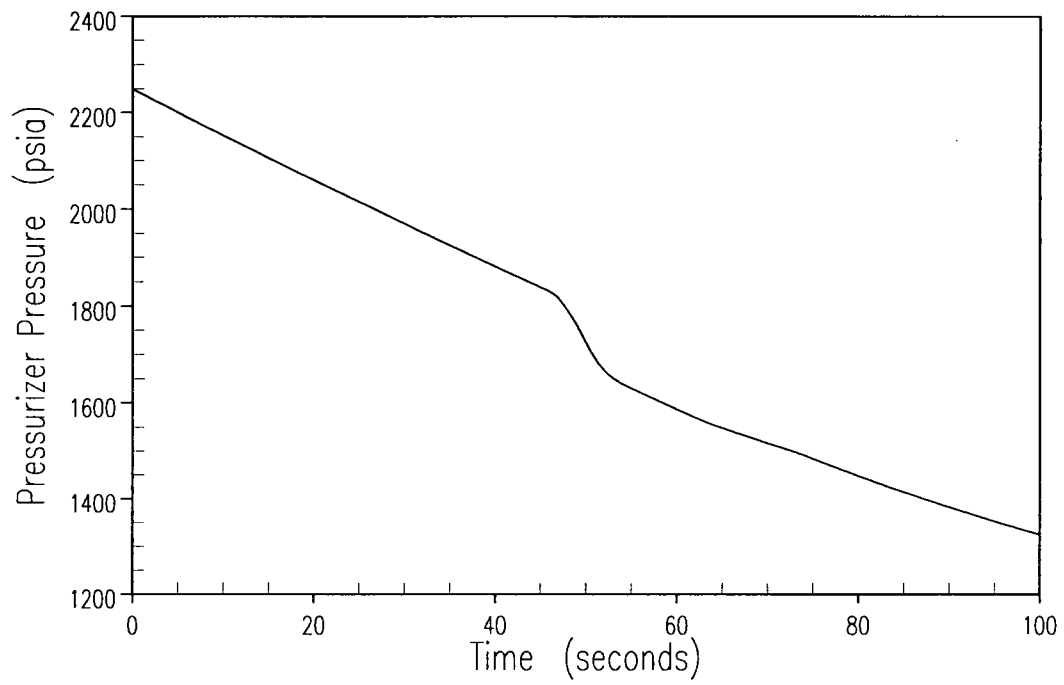


Figure 2.8.5.6.1-4 RCS Depressurization – Pressurizer Pressure Versus Time (Unit 2)

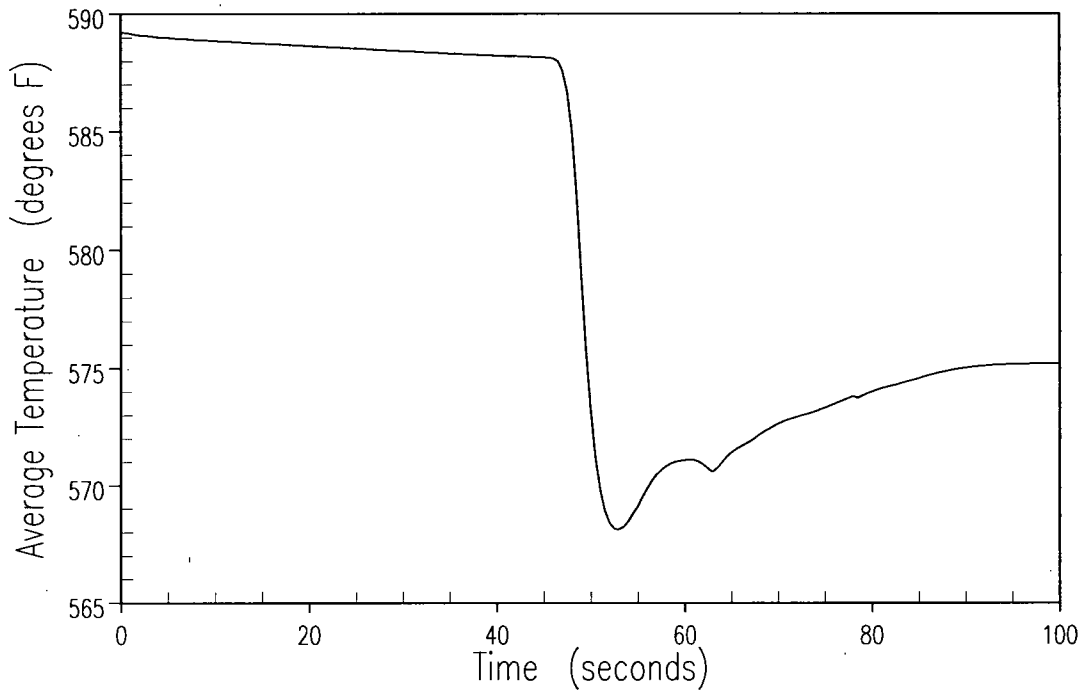


Figure 2.8.5.6.1-5 RCS Depressurization – Indicated Loop Average Temperature Versus Time (Unit 1)

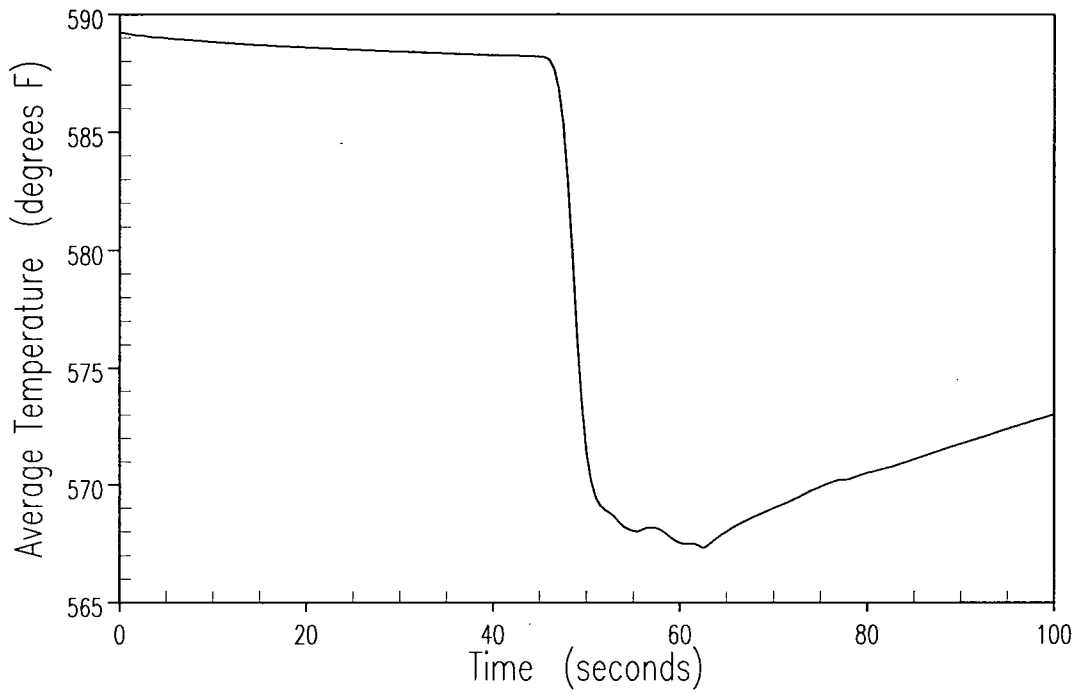


Figure 2.8.5.6.1-6 RCS Depressurization – Indicated Loop Average Temperature Versus Time (Unit 2)

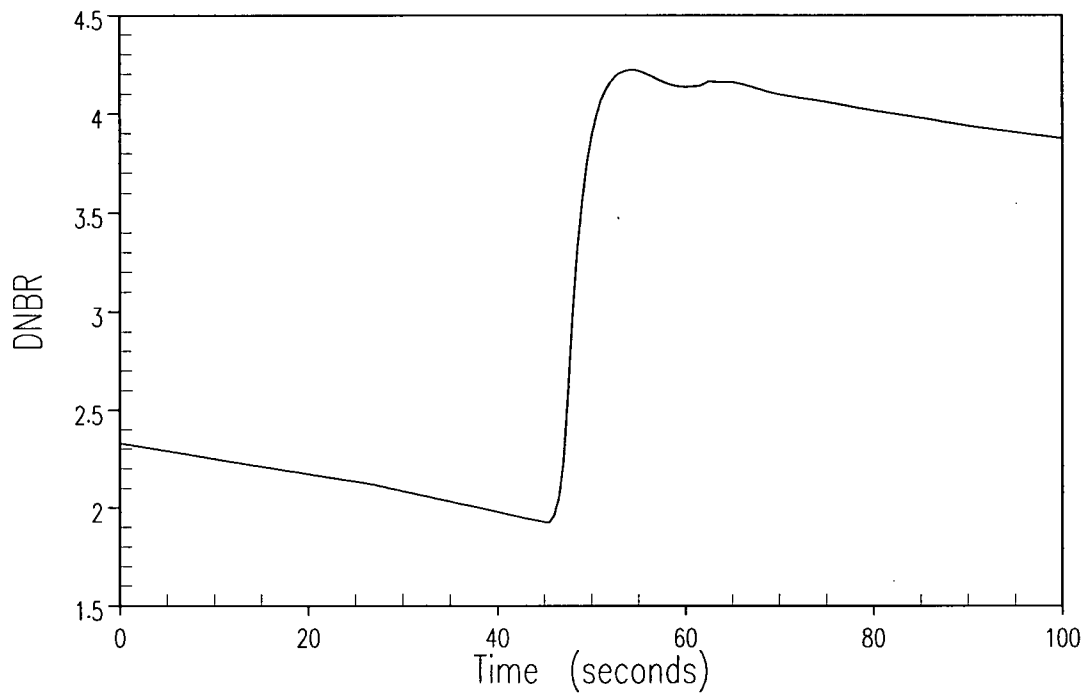


Figure 2.8.5.6.1-7 RCS Depressurization – DNBR Versus Time (Unit 1)

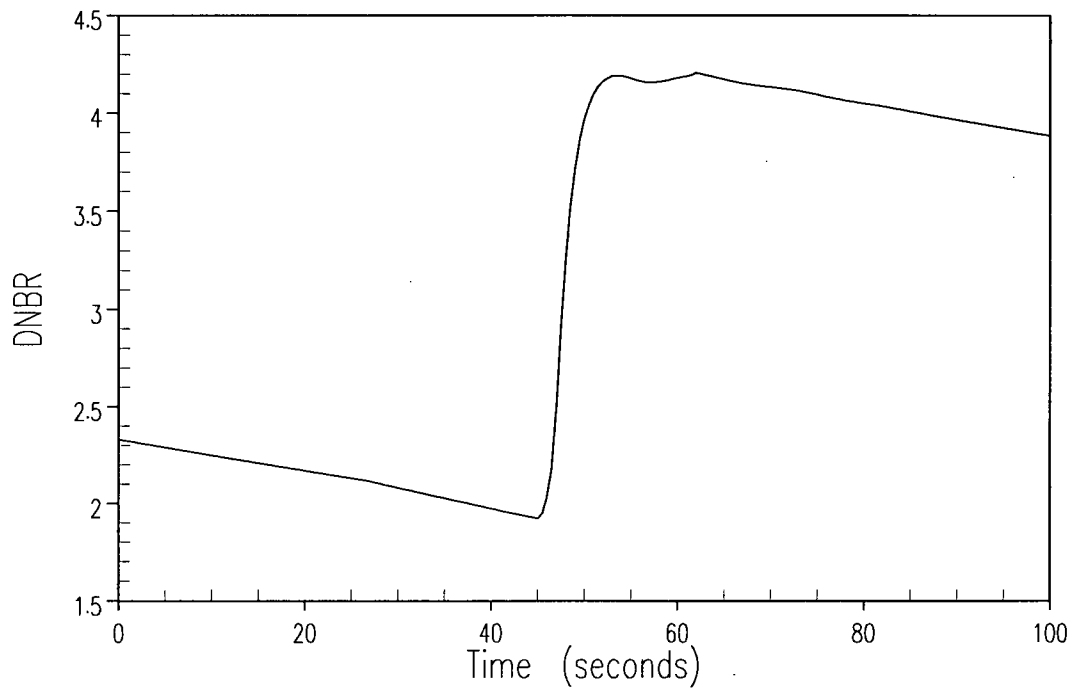


Figure 2.8.5.6.1-8 RCS Depressurization – DNBR Versus Time (Unit 2)

2.8.5.6.2 Steam Generator Tube Rupture

2.8.5.6.2.1 Regulatory Evaluation

A steam generator tube rupture (SGTR) event causes a direct release of radioactive material contained in the primary coolant to the environment through the ruptured SG tube and main steam safety valves (MSSVs) or atmospheric relief valves (ARVs). Reactor protection and engineered safety features (ESFs) are actuated to mitigate the accident and restrict the offsite dose to within the guidelines of 10 CFR 100. The review covered:

- The postulated initial core and plant conditions
- The method of thermal-hydraulic analysis
- The sequence of events
- The assumed reactions of reactor system components
- The functional and operational characteristics of the reactor protection system
- The operator actions consistent with the plant's emergency operating procedures
- The results of the accident analysis

A single failure of a mitigating system is assumed for this event. The review of the SGTR is focused on the thermal-hydraulic analysis for the SGTR in order to determine whether 10 CFR 100 is satisfied with respect to radiological consequences, which are discussed in Licensing Report (LR) subsection 2.9.6, Radiological Consequences of a Steam Generator Tube Rupture; and to confirm that the ruptured steam generator does not experience an overfill. Preventing steam generator overfill is necessary to prevent the release of water to the environment through the MSSVs or ARVs and to preclude the possibility of failure of main steam lines.

Current Licensing Basis

As noted in the Final Safety Analysis Report (FSAR) Section 15.6.3, the SGTR accident analysis includes analyses performed to demonstrate margin-to-overfill and analyses to ensure that possible radiological dose consequences are within allowable guidelines. The dose analysis requires thermal-hydraulic calculations be performed to determine the amount of reactor coolant discharged to the ruptured steam generator and the amounts of steam released from the steam generators. The FSAR analyses were analyzed following the methodology of RXE-88-101, (Reference 1), using the RETRAN02 code to calculate the margin-to-overfill and mass release data. The Westinghouse Owners Group (WOG) SGTR subcommittee, of which Luminant Power was a participant, was formed to address Nuclear Regulatory Commission (NRC) questions regarding assumptions used in SGTR safety analyses that arose as a result of the January 1982 SGTR event at the R. E. Ginna Station. The WOG SGTR subcommittee produced WCAP-10698 and its supplement (References 2 and 3) documenting a methodology for analyzing the consequences of an SGTR. In Reference 1, Luminant Power developed a plant-specific application of the WOG methodology documented in References 2 and 3.

2.8.5.6.2.2 Technical Evaluation

The evaluation of the design basis SGTR event demonstrated that the current design is acceptable to support the uprate operation.

2.8.5.6.2.2.1 Introduction

The SGTR analysis is described in the FSAR, Section 15.6.3. The SGTR accident analysis included analyses performed to demonstrate margin-to-overfill and analyses to ensure that possible radiological dose consequences are within allowable guidelines. The dose analysis required that thermal-hydraulic calculations be performed to determine the amount of reactor coolant discharged to the ruptured steam generator, and the amounts of steam released from the steam generators. The effects of limiting single failures and the times for required operator actions were explicitly included in the analyses. Only the results of the limiting margin-to-overfill and mass-release cases are discussed in the FSAR.

The uprate analyses were performed using the methodology developed in WCAP-10698 and its supplement (References 2 and 3), but with the RETRAN-02 computer code (Reference 4).

The analysis included an analyzed nuclear steam supply system (NSSS) power level of 3,628 MWt, and a full-power T_{avg} operating range from 574.2° to 589.2°F and up to 10-percent steam generator tube plugging as well as a main feedwater temperature range from 390° to 450.3°F. All cases were analyzed with a loss-of-offsite power.

Note that in order to demonstrate margin to overfill, Unit 1 is limited to a minimum T_{avg} of 580.0°F. A lower T_{avg} would result in overfilling the ruptured steam generator. Unit 2 demonstrated margin to overfill with a T_{avg} of 574.2°F.

The margin-to-overfill transient was analyzed until the ruptured steam generator secondary-side and reactor coolant system (RCS) pressures equalized, at which time the ruptured tube flow was considered to be terminated. The margin-to-overfill analysis examined both CPNPP units, including unit-specific single failures and operating conditions. Only the results of the limiting margin-to-overfill case are presented.

The mass-release case determines the primary-to-secondary break flows and steam releases for the SGTR radiological consequences analysis. This case is analyzed through tube rupture flow isolation and cooldown to residual heat removal system (RHRS) in-service conditions to obtain the total steam releases from the intact and ruptured steam generators. (At this point the plant proceeds to Mode 5 (cold shutdown) conditions using the RHRS without additional steam release.) The mass release analysis considered both CPNPP units. It was determined that Unit 1 bounds Unit 2 for the purposes of the mass release analysis. Only the results of the limiting mass release case are presented.

The radiological consequences analysis is presented in LR subsection 2.9.6, Radiological Consequences of a Steam Generator Tube Rupture.

2.8.5.6.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

Design Basis Accident

The accident modeled is a double-ended break of one steam generator tube located at the top of the tubesheet on the outlet-cold-leg-side of the steam generator. The location of the break on the cold side of the steam generator results in higher primary-to-secondary leakage than a break on the hot side of the steam generator. However, the break flow flashing fraction was conservatively calculated for use in the radiological consequences analysis assuming that all of the break flow came from the hot leg side of the steam generator. The combination of these conservative assumptions regarding the break location results in a conservative calculation of the radiological consequences. It was also assumed that loss-of-offsite power occurred at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip. The condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam was released to the atmosphere through the steam generator ARVs.

Single Failure Considerations

The effects of single failures in margin-to-overfill and mass-release analyses were investigated in WCAP-10698 and its Supplement 1 (References 2 and 3). The limiting single failures for CPNPP SGTR analyses are described below.

The limiting single failure for margin-to-overfill considerations for Unit 1 is an ARV on one intact steam generator failing to open. The limiting single failure for margin-to-overfill considerations for Unit 2 is the failure of a DC bus, resulting in a failure to open the ARVs on two intact steam generators. The failure of the ARVs on intact steam generators requires the RCS cooldown to be performed with fewer steam generators, resulting in a longer cooldown and delayed break flow termination.

The limiting single failure for the mass-release analysis for the CPNPP units is the ARV failing to close on the ruptured steam generator (Reference 3). Failure of this ARV causes an uncontrolled release directly to the atmosphere and the uncontrolled depressurization of the ruptured steam generator resulting in increased primary-to-secondary flow. Pressure in the ruptured steam generator remains less than the RCS until the failed ARV is isolated and recovery actions are completed.

Operator Actions Assumed

Important operator actions in the CPNPP Emergency Operating Procedures (EOPs) were explicitly modeled in the analysis. These actions were intended to terminate flow through the SGTR before proceeding to long-term cooldown. The operator actions modeled in the uprate analysis and the associated times were consistent with or conservative compared to those currently incorporated in the analyses presented in FSAR Section 15.6.3. These action times consisted of two components: initiation times (for the operator to start actions) and plant/system response times (for the plant conditions to reach performance objectives such as temperature,

pressure, flow, etc., required by the recovery action). The latter times were determined from the thermal-hydraulic transient analyses of the SGTR accident. The operator action times are summarized in Table 2.8.5.6.2-1.

The operator actions that were modeled include:

- Controlling excessive auxiliary feedwater (AFW) flow to the ruptured steam generators.

In the analysis, this function is modeled by assuming the operators isolate the turbine-driven AFW (TDAFW) flow at 3 minutes past reactor trip. The TDAFW flow is isolated consistent with CPNPP procedure EOP-0.0A/(B), "Reactor Trip or Safety Injection," which calls for throttling the AFW flow to steam generators with levels above just-on-span.

- Identifying the ruptured steam generator.

Several means are available to the operator to identify a ruptured steam generator. The predominant indications are an unexpected rapid increase in the ruptured steam generator's narrow-range level following the reactor trip, high radiation from a steam generator blowdown radiation monitor, or high radiation from a steam line radiation monitor. The analysis modeled a loss-of-offsite power concurrent with reactor trip. The radiation monitors are non-safety related and are not credited for accident mitigation. This results in the assumed unavailability of the radiation monitors, leaving only the ruptured steam generator level increase as a means of identifying the ruptured steam generator.

- Isolating the steam flow from the ruptured steam generator and throttling auxiliary feedwater flow to the ruptured steam generator.

Isolating the ruptured steam generator minimizes radiological releases and reduces the possibility of overfilling by minimizing the accumulation of feedwater. This action also enables the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary-to-secondary flow. It was assumed that the ruptured steam generator would be isolated when the level in the steam generator reached between being just on span and 50 percent on the narrow-range instrument (modeled as 44-percent narrow-range level for Unit 1 and 30-percent narrow-range level for Unit 2), or after an operator action time of 13 minutes, whichever was longer.

- Cooling down the RCS by dumping steam from the intact steam generators.

The RCS is cooled down as rapidly as possible to a temperature less than the saturation temperature corresponding to the ruptured steam generator's pressure. The cooldown is performed using the intact steam generators' ARVs since neither the steam dump valves nor the condenser were available following the assumed loss-of-offsite power. The

cooldown continues until RCS subcooling at the ruptured steam generator pressure is 20°F, plus an allowance of 25°F for instrument uncertainty.

- Depressurizing the RCS after cooldown to minimize break flow and restore pressurizer level.

After the RCS cooldown, safety injection is terminated since it is the principal contributor to tube rupture flow. Depressurizing the RCS is required to ensure an adequate RCS inventory and reliable pressurizer level indication prior to stopping injection. Since offsite power was assumed to be lost at the time of reactor trip, the reactor coolant pumps were not running, and thus normal pressurizer spray was not available. It was assumed that the operator depressurized the RCS using a pressurizer power-operated relief valve (PORV). The operator continues to depressurize until any of the following is satisfied:

- RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 13 percent
 - Pressurizer level is greater than 75 percent
 - RCS subcooling is less than the 25°F allowance for subcooling instrument uncertainty
- Terminating safety injection to prevent re-pressurization of the RCS and terminate primary to secondary flow.

Safety injection is terminated when all of the following are satisfied:

- The RCS pressure stabilizes or starts to increase.
- The RCS subcooling is greater than the 25°F allowance for subcooling instrument uncertainty.
- Secondary heat sink is available.
- The pressurizer level is greater than 13 percent.

Following termination of tube rupture flow, the operator is required to perform additional actions to bring the plant to Mode 5 (cold-shutdown) conditions. The operator actions are defined in the CPNPP EOPs. Only two of the actions were explicitly considered in the analysis.

The operator is required to cool the RCS to the RHRS in-service temperature by feeding and steaming the intact steam generators. The SGTR long-term mass-release analysis assumed the operator performs this action by dumping steam to the atmosphere via the ARVs. Although other preferable cooldown methods (such as steam dump to the condenser to minimize activity releases) are identified in the CPNPP EOPs, steam dump to the atmosphere was modeled to maximize activity releases.

Cooldown and depressurization of the ruptured steam generator is performed after the RCS is cooled to the RHRS in-service temperature. With a loss-of-offsite power, the operator releases steam from the ruptured steam generator to the atmosphere. (This method is conservative for radiological calculations since it maximizes the activity released from the plant.) The operator maintains equal pressure between the RCS and ruptured steam generator secondary side using the PORV as needed until the RHRS is brought online.

Explicit operator action times were not defined since cooldown can proceed more gradually after tube rupture flow is terminated.

Input Parameters and Initial Conditions

Parameters and initial conditions common to the margin-to-overfill and mass release analyses were:

- Both units were at 100.6-percent rated thermal power (RTP). Both units were operating within the T_{avg} window, which ranged from 574.2° to 589.2°F. The mass release analysis modeled the high T_{avg} since that maximized secondary steam releases. The margin-to-overfill analysis modeled the low T_{avg} as that minimized secondary steam releases and maximized the mass flow rate through the break. Note that, in order to demonstrate margin to overfill, Unit 1 is limited to a minimum T_{avg} of 580.0°F. A lower T_{avg} would result in overfilling the ruptured steam generator. Unit 2 demonstrated margin to overfill with a T_{avg} of 574.2°F. Other initial conditions are summarized in Table 2.8.5.6.2-2.
- Reactor trip occurred when the overtemperature N-16 setpoint was reached. No reactor trip delay was assumed since it maximized the secondary-side inventory in the ruptured steam generator and steam releases from all steam generators. It was also assumed that loss-of-offsite power occurred at the time of reactor trip.
- The turbine automatically tripped following a reactor trip. Zero delay was assumed since it minimized the steam flow to the turbine, and maximized the secondary-side water inventory in the ruptured steam generator and steam releases from all steam generators.
- The condenser was unavailable for steam dump following reactor trip. All subsequent steam relief was through the ARVs, and MSSVs, if needed.
- A low ARV setpoint of 1,140 psia was used since control at lower steam generator pressures caused a greater primary-to-secondary side pressure differential and tube rupture flow.
- Maximum safeguards safety injection flows were modeled. This assumption conservatively increased the break flow through the ruptured tube.
- Auxiliary feedwater was automatically started following reactor trip and loss-of-offsite power.

-
- Operation of charging and letdown systems and pressurizer heaters were not credited. Operating these systems delays the reactor trip, which reduces the severity of the analyzed transient.
 - Conservatively high decay heat rates were used. The increased heat input resulted in greater tube rupture flow after reactor trip due to the longer time needed for removing heat and depressurizing the RCS.

For the margin-to-overfill cases:

- The initial water mass in the steam generators corresponded to the nominal steam generator level plus uncertainty. For Unit 1, the initial water mass corresponded to a level of 77 percent on the narrow-range level, which corresponded to a nominal level of 67-percent plus 10-percent uncertainty. For Unit 2, the initial water mass corresponded to a level of 82 percent on the narrow-range level, which corresponded to a nominal level of 64-percent plus 18-percent uncertainty. A higher initial mass in the ruptured steam generator is conservative with respect to reducing the margin to overfill. (The total fluid mass shown in Table 2.8.5.6.2-2 corresponds to the mass in the Unit 1 steam generators at T_{avg} of 580.0°F at full power, with 10-percent tube plugging and a feedwater temperature of 390°F.)
- The turbine runback on overtemperature N-16 at a rate of 10-percent power per minute prior to reactor trip was simulated but not credited for delaying reactor trip, for a maximum of 3 minutes. Turbine runback increased the secondary water mass with reduced load, because the feedwater controller attempts to maintain steam generator level as power decreased before the trip.
- The maximum available AFW flow was modeled to maximize the mass of water in the ruptured steam generator at the time of isolation. The intact steam generators received the minimum AFW flow. This increased the steam releases from the intact steam generators, which decreased the steam release from the ruptured steam generator.

For the mass release analyses:

- A turbine runback was not assumed since it delays reactor trip and increases secondary mass. An earlier reactor trip results in greater steam releases to the atmosphere from all steam generators.
- The steam generator water mass corresponded to 57 percent on the narrow-range level. This mass represented the full-power, nominal steam generator water level with a -10-percent instrument uncertainty applied. A lower initial mass in the ruptured steam generator increases the predicted offsite doses. (The value shown in Table 2.8.5.6.2-2 corresponds to T_{avg} at 589.2°F with 0-percent tube plugging and feedwater temperature of 450.3°F.)
- The minimum AFW flow, distributed equally, was modeled to maximize the steam releases.

Acceptance Criteria

No regulatory acceptance criteria were used for the margin-to-overfill and mass release analyses. Both analyses were performed using conservative assumptions to demonstrate the ability of the operator to limit the system transient and establish parameters for providing a bounding radiological consequence assessment.

In order to demonstrate that water release from the ruptured steam generator did not have to be considered in the radiological consequences assessment, the margin-to-overfill analysis was performed to demonstrate that the secondary side of the ruptured steam generator did not completely fill with water. The available secondary side volume of a single CPNPP Unit 1 steam generator is 5,328 ft³. The available secondary-side volume of a single Unit 2 steam generator is 5,955 ft³. Margin-to-overfill was demonstrated, provided the transient calculated steam generator secondary side water volume was less than 5,328 ft³ for Unit 1 and less than 5,955 ft³ for Unit 2.

The radiological consequences analysis acceptance criteria for the SGTR are discussed in LR subsection 2.9.6, Radiological Consequences of a Steam Generator Tube Rupture.

2.8.5.6.2.2.3 Description of Analyses and Evaluations

The margin-to-overfill analyses were performed using the methodology in WCAP-10698 (Reference 2) with plant-specific parameters. The ruptured steam generator's secondary-side water volume was calculated as a function of time to demonstrate that overfill did not occur. The analysis was performed from the start of the rupture until break flow was terminated at equalization of primary-and-secondary pressures. The methodology included the explicit modeling of operator actions in the CPNPP EOPs required for mitigation of the SGTR accident.

The mass release analyses were performed using the methodology in WCAP-10698 and its Supplement 1 (References 2 and 3). The plant response, the integrated primary-to-secondary break flow, and the steam releases to the condenser and to the atmosphere up to the time the tube rupture flow was terminated were all calculated using RETRAN-02 results. When calculating the amount of break flow that flashed to steam, 100 percent of the break flow was assumed to come from the hot leg side of the break.

The steam release from the time of tube rupture flow termination until 2 hours, and from 2 to 11 hours, were determined from mass-and-energy balances using the RCS's and intact steam generators' conditions. Following termination of the tube rupture flow, the intact steam generators' ARVs were assumed to cool down the plant at less than the maximum allowable rate of 100°F/hour to an RHRS in-service temperature of 350°F.

The ruptured steam generator was assumed to be depressurized to the RHRS in-service pressure of 365 psia immediately after the RCS cooldown. The amount of steam released was determined from mass-and-energy balances. No changes in thermodynamic conditions were assumed from termination of the tube rupture flow until depressurization was started since the ruptured steam generator was isolated. Steam releases from all steam generators are

considered terminated when a single train of RHR is able to remove decay heat. This was assumed to occur at a conservatively long value of 11 hours.

2.8.5.6.2.2.4 SGTR Results

Only the results for the limiting margin-to-overfill and mass release cases were presented.

SGTR Margin-to-Overfill Transient Analysis

Results are presented for the worst-case margin-to-overfill analysis. The worst case, considering both units, the range of T_{avg} , and the range of tube plugging was Unit 1—10-percent tube plugging with a T_{avg} of 580.0°F. The analysis showed that 580.0°F was the lowest temperature to demonstrating margin to overfill for Unit 1. (Unit 2 demonstrated margin to overfill at a T_{avg} of 574.2°F.) The sequence of events is summarized in Table 2.8.5.6.2-3 and Figures 2.8.5.6.2-1 to 2.8.5.6.2-7 show primary- and secondary-side responses until the SGTR flow was terminated.

To ensure proper initialization of the RETRAN-02 model, 100 seconds of steady-state operation were modeled prior to initiating the break, and all times listed include this 100 seconds. Once the break was initiated, the reactor coolant flow to the secondary side through the ruptured tube immediately caused the pressurizer level and pressure to decrease, as shown in Figures 2.8.5.6.2-1 and 2.8.5.6.2-2. The continued decrease in pressurizer pressure caused the overtemperature N-16 setpoint to be reached at 287 seconds, followed by immediate reactor and turbine trips. The reactor coolant pumps tripped due to the assumed loss-of-offsite power at the time of reactor trip. Immediately following reactor trip, the temperature differential across the hot and cold legs decreased as core power decayed. The temperature differential then increased as shown in Figure 2.8.5.6.2-4 as the pumps coasted down and natural circulation flow developed.

With the steam dump valves closed after trip (due to the loss-of-condenser vacuum resulting from the assumed loss-of-offsite power at the time of reactor trip), the secondary-side pressures in all steam generators increased rapidly to the ARV setpoint as shown in Figure 2.8.5.6.2-3. The pressurizer level and pressure dropped more rapidly, and safety injection was actuated via the low-pressurizer pressure setpoint at 300 seconds (see Figures 2.8.5.6.2-1 and 2.8.5.6.2-2 and Table 2.8.5.6.2-3).

Turbine-driven auxiliary feedwater flow was isolated at 3 minutes past reactor trip (Table 2.8.5.6.2-1) at 467 seconds. The operator isolated the ruptured steam generator by isolating steam flow and throttling the remaining motor-driven auxiliary feedwater flow at 13 minutes after break initiation (see Table 2.8.5.6.2-3). The operator actions were assumed at 13 minutes after break initiation since the ruptured steam generator's narrow-range level had previously returned to greater than 44 percent. After AFW isolation, the increase in fluid mass in the ruptured steam generator (shown in Figure 2.8.5.6.2-6) was due to the ruptured tube flow.

There was a 5-minute operator delay time before initiating the cooldown (see Table 2.8.5.6.2-1) at 1,180 seconds (See Table 2.8.5.6.2-3). The ARV on one intact steam generator was assumed to fail closed at the start of the cooldown. The cooldown was then completed with the ARVs on the remaining two intact steam generators. The subsequent reduction in the available intact steam generators' pressure is shown in Figure 2.8.5.6.2-3, and the resulting cooldown of the RCS temperature is shown in Figure 2.8.5.6.2-4. The pressurizer level and pressure also decreased during this cooldown, as shown in Figures 2.8.5.6.2-1 and 2.8.5.6.2-2. The cooldown was continued until RCS subcooling at the ruptured steam generator pressure was 20°F, plus an allowance of 25°F for instrument uncertainty. The cooldown was completed at 1,980 seconds (see Table 2.8.5.6.2-3).

The available intact steam generators' ARVs were later re-opened to dump steam and maintain an adequate RCS subcooling margin. When the ARVs were opened, the increased energy transfer from the primary to the secondary side also aided in the depressurization of the RCS to the ruptured steam generator's pressure (see Figures 2.8.5.6.2-2 and 2.8.5.6.2-3).

The operator began to depressurize the RCS using the pressurizer PORV at 2,100 seconds after a 2-minute delay (see Table 2.8.5.6.2-1). Depressurization was terminated at 2,180 seconds when the RCS subcooling was reduced below the 25°F allowance for instrument uncertainty. The depressurization reduced pressurizer pressure and the break flow and increased safety injection flow to refill the pressurizer, as shown in Figures 2.8.5.6.2-1 and 2.8.5.6.2-2.

A 2-minute delay was imposed prior to termination of safety injection flow (see Table 2.8.5.6.2-1). Safety injection was terminated in the analysis at that time because the safety injection termination criteria were satisfied. The RCS pressure was allowed to increase to 50 psi above the ruptured steam generator pressure to ensure that the RCS pressure was increasing when safety injection was terminated. The operator terminated safety injection at 2,300 seconds because the safety injection termination criteria were satisfied and the RCS pressure began to decrease, as shown in Figure 2.8.5.6.2-2. The primary-to-secondary flow continued until the RCS and ruptured steam generator pressures equalized at approximately 3,045 seconds.

The primary-to-secondary break flow rate and water volume in the ruptured steam generator are shown in Figure 2.8.5.6.2-5 and 2.8.5.6.2-7, respectively. Figure 2.8.5.6.2-7 shows a bounding value of 23 ft³ margin-to-overfill relative to the steam generator's total volume of 5,328 ft³. For Unit 2 the margin-to-overfill is 185 ft³. Therefore, it was concluded that overfill of the ruptured steam generator would not occur for a design basis SGTR for CPNPP.

SGTR Mass Release Transient Analysis

The maximum mass release occurred for Unit 1 with a steam generator tube plugging level of 0 percent, and with the reactor initially operating with a T_{avg} at 589.2°F. The sequence of events is summarized in Table 2.8.5.6.2-4, and the primary- and secondary-side responses appear in Figures 2.8.5.6.2-8 to 2.8.5.6.2-19. Total mass releases for use in the dose analyses are summarized in Table 2.8.5.6.2-5.

The mass release and margin-to-overfill results were similar until 13 minutes from break initiation. The mass release transient modeled a low initial secondary inventory and minimum AFW flow with actuation delayed conservatively. As a result, the ruptured steam generator level did not reach 44 percent until 1,077 seconds. Isolating the ruptured steam generator was therefore delayed until 1,077 seconds, consistent with Table 2.8.5.6.2-1. At 1,077 seconds, the ruptured steam generator's ARV was assumed to fail open. Steam releases from the ruptured steam generator are shown in Figure 2.8.5.6.2-16. The failure of the ARV caused the steam generator to rapidly depressurize, and the primary-to-secondary flow through the ruptured tube to increase (see Figures 2.8.5.6.2-10 and 2.8.5.6.2-13). The ruptured steam generator's depressurization caused the RCS pressure and temperature to decrease. The operator identified and locally closed the block valve for the failed ARV after 36 minutes (see Table 2.8.5.6.2-1). The depressurization of the ruptured steam generator stopped at 3,237 seconds, and its pressure began to increase, as shown in Figure 2.8.5.6.2-10.

There was a 5-minute operator action delay time imposed prior to initiating cooldown after the failed ARV's block valve was closed (see Table 2.8.5.6.2-1). The cooldown was performed using the intact steam generators' ARVs to dump steam to the atmosphere, and continued until the RCS subcooling at the ruptured steam generator pressure at the start of cooldown was 20°F, plus an allowance of 25°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator when the cooldown was initiated, the RCS had to be cooled to a lower temperature to satisfy the cooldown criterion. The net effect was that the cooldown period was longer, relative to the overfill case. The cooldown was completed at 5,082 seconds. The reductions in the intact steam generators' pressure and the RCS temperature during the cooldown period are shown in Figures 2.8.5.6.2-10, 2.8.5.6.2-11, and 2.8.5.6.2-12, respectively. The intact steam generators' ARVs were later reopened (see Figure 2.8.5.6.2-17) to maintain RCS temperature and subcooling margin.

The RCS depressurization began later than the limiting margin-to-overfill case. After a 2-minute delay (see Table 2.8.5.6.2-1), the operator used the pressurizer PORV to depressurize, starting at 5,202 seconds. Depressurization was terminated at 5,358 seconds, when the RCS pressure was less than the ruptured steam generator's pressure and the pressurizers level was above 13 percent. During depressurization, safety injection flow refilled the pressurizer while break flow was reduced, as shown in Figures 2.8.5.6.2-8 and 2.8.5.6.2-13, respectively.

At this point, a 2-minute operator delay (see Table 2.8.5.6.2-1) was assumed before shutting down safety injection at 5,478 seconds. Like the overfill analysis, safety injection was terminated after that 2 minutes because the criteria were satisfied. The RCS pressure began to decrease, as shown in Figure 2.8.5.6.2-9. Figure 2.8.5.6.2-13 shows that the primary-to-secondary flow continued until the RCS and ruptured steam generator pressures equalized at 7,156 seconds. Figures 2.8.5.6.2-18 and 2.8.5.6.2-19 show the transient ruptured steam generator water volume and mass. The peak ruptured steam generator water volume is well below the available volume of 5,328 ft³.

The integrated flashing break flow was 25,482 lbm. Figures 2.8.5.6.2-14 and 2.8.5.6.2-15 show the flashing fraction and integrated flashed break flows, respectively.

Following termination of the tube rupture flow, the RCS was cooled down using the intact steam generators. The steam releases are presented in Table 2.8.5.6.2-5. Since the condenser was in service until reactor trip, any radioactivity released to the atmosphere before reactor trip was through the condenser air ejector. After reactor trip, the releases were assumed to be via the ARVs. Table 2.8.5.6.2-5 indicates that 215,400 lbm of steam was released to the atmosphere from the ruptured steam generator within the first 2 hours (that is, the ruptured steam generator was isolated within this interval). After 2 hours, 20,400 lbm of steam was released to the atmosphere from the ruptured steam generator, when it was depressurized after the RCS was cooled to the RHRS in-service temperature. A total of 307,900 lbm of reactor coolant flowed through the tube rupture before break flow was terminated.

The analysis performed to calculate the mass transfer data for input to the radiological consequences analysis has been completed and data tabulated for the limiting case. The results of the analysis were used as input to the radiological consequences analysis presented in LR subsection 2.9.6, Radiological Consequences of a Steam Generator Tube Rupture.

2.8.5.6.2.3 Conclusion

Luminant Power has reviewed the analysis of the SGTR accident and concludes that the analysis has adequately accounted for the plant operation at the uprate power level and was performed using acceptable analytical methods and approved computer codes. Luminant Power further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the ruptured steam generator. Therefore, Luminant Power finds the uprate acceptable with respect to the SGTR event.

2.8.5.6.2.4 References

1. RXE-88-101, "Design Basis Analysis of a Postulated Steam Generator Tube Rupture Event for Comanche Peak Steam Electric Station," Unit 1, March 1988.
2. WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," August 1987.
3. WCAP-10698 Supplement 1, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," March 1986.
4. WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

Table 2.8.5.6.2-1**Operator Action Times For Design Basis SGTR Analysis**

Action	Time
Isolate TDAFW Flow to all Steam Generators	3 minutes from reactor trip
Identify and Isolate Ruptured Steam Generator	Maximum of 13 minutes past break initiation or the calculated time to reach the midpoint of just-on-span and 50% span. For Unit 1, this is 44% span. For Unit 2, this is 30% span.
Identify and Isolate the Failed-Open ARV (mass release analysis only)	36 minutes from the time of ARV failure
Operator Action Time to Initiate Cooldown	5 minutes from complete ruptured steam generator isolation
Cooldown	Calculated time for RCS cooldown
Operator Action Time to Initiate Depressurization	2 minutes from end of cooldown
Depressurization	Calculated time for RCS depressurization
Operator Action time to Initiate Safety Injection Termination	Maximum of 2 minutes from end of depressurization or time to satisfy safety injection termination criteria
Pressure Equalization	Calculated time for equalization of RCS and ruptured steam generator pressures

**Table 2.8.5.6.2-2
Plant Parameters Used in SGTR Analysis**

	SGTR Overfill Analysis	SGTR Dose Analysis
Initial RCS pressure (psia)	2,280	2,220
Initial Steam Generator Water Mass (lbm)	117,500	71,000
Reactor Trip Delay (sec)	0.0	0.0
Turbine Trip Delay	0.0	0.0
Pressurizer Pressure Setpoint for Safety Injection (psia)	2,000	2,000
Steam Generator Atmospheric Relief Valve Setpoint (psia)	1,140	1,140
Safety Injection System Pump Delay (sec)	0.0	0.0
AFW Delay (sec)	0.0	60 sec from safety injection
AFW Flow Ruptured steam generator Intact steam generators	Maximum Minimum	Minimum Minimum
AFW Temperature (°F)	100	100
Decay Heat	120% American Nuclear Society (ANS)	120% ANS
Safety Injection Flow	Maximum safeguards	Maximum safeguards
ARV Capacity Ruptured steam generator (1,200 psia reference pressure) Intact steam generators (1,200 psia reference pressure)	750,000 lbm/hr 750,000 lbm/hr	968,800 lbm/hr 750,000 lbm/hr

Table 2.8.5.6.2-3
Sequence of Events for Margin-to-Overfill Analysis

Event	Time (seconds)
SGTR	100
Reactor Trip	287
AFW Initiated	287
Safety Injection	300
TDAFW to all Steam Generators Isolated	467
Ruptured Steam Generator Isolated	880
RCS Cooldown Initiated	1,180
RCS Cooldown Target Temperature Reached	1,980
Pressurizer PORV Opened	2,100
Pressurizer PORV Closed	2,180
Safety Injection Terminated	2,300
Break Flow Terminated	3,045

Table 2.8.5.6.2-4	
Sequence of Events for Input to Radiological Consequences Analysis	
Event	Time (seconds)
SGTR	100
Reactor Trip	277
Safety Injection	287
AFW Initiated	347
TDAFW to All Steam Generators Isolated	457
Ruptured Steam Generator Isolated	1,077
Ruptured Steam Generator ARV Fails Open	1,077
Ruptured Steam Generator ARV Isolated	3,237
RCS Cooldown Initiated	3,537
Break Flow Flashing Terminated	3,958
RCS Cooldown Target Temperature Reached	5,082
Pressurizer PORV Opened	5,202
Pressurizer PORV Closed	5,358
Safety Injection Terminated	5,478
Break Flow Terminated	7,156

**Table 2.8.5.6.2-5
Mass Releases
Total Mass Flow (Pounds)**

	Time Period			
	Start of Event to Time of Reactor Trip ^(1,2)	Time of Reactor Trip to Time at Which Break Flow is Terminated ⁽¹⁾	Time at Which Break Flow is Terminated to 2 Hours	2 Hours to Time at Which RCS Reaches RHR In-Service Conditions ⁽¹⁾
Ruptured Steam Generator – Condenser – Atmosphere	314,500 0	0 215,400	0 0	0 20,400
Intact Steam Generators – Condenser – Atmosphere	937,700 0.0	0 365,600	0 33,100	0 1,302,800
Total Break Flow	9,400	298,500	0	0
Flashed Break Flow	1,390	24,092	0	0

Notes:

1. The break is initiated at 100 seconds. Reactor trip occurs at 277 seconds; break flow stops flashing at 3,958 seconds; break flow is terminated at 7,156 seconds; RHR conditions are reached at 11 hours.
2. Pre-trip releases to condenser and feedwater flows include 100 seconds steady-state operation prior to initiation of the break.

Comanche Peak Unit 1 Steam Generator Tube Rupture Margin to Overfill

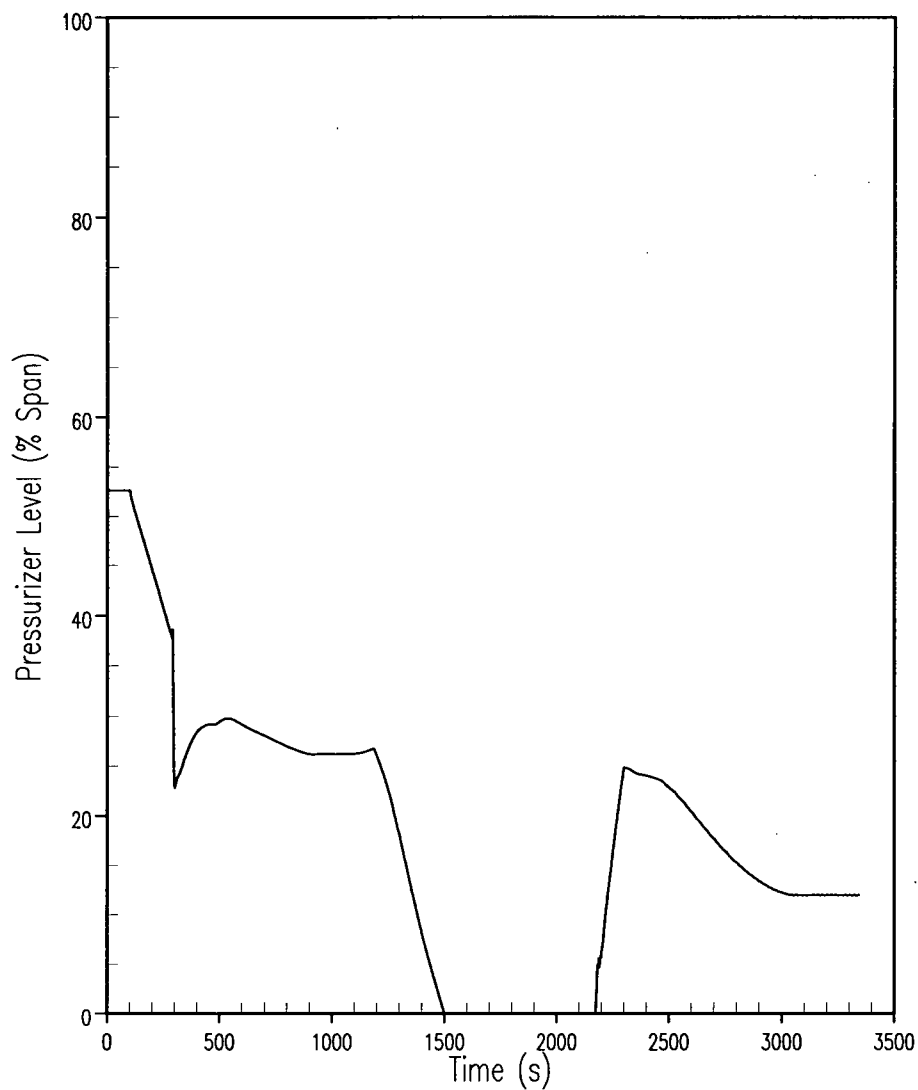


Figure 2.8.5.6.2-1 SGTR (Overfill), Pressurizer Level Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Margin to Overfill

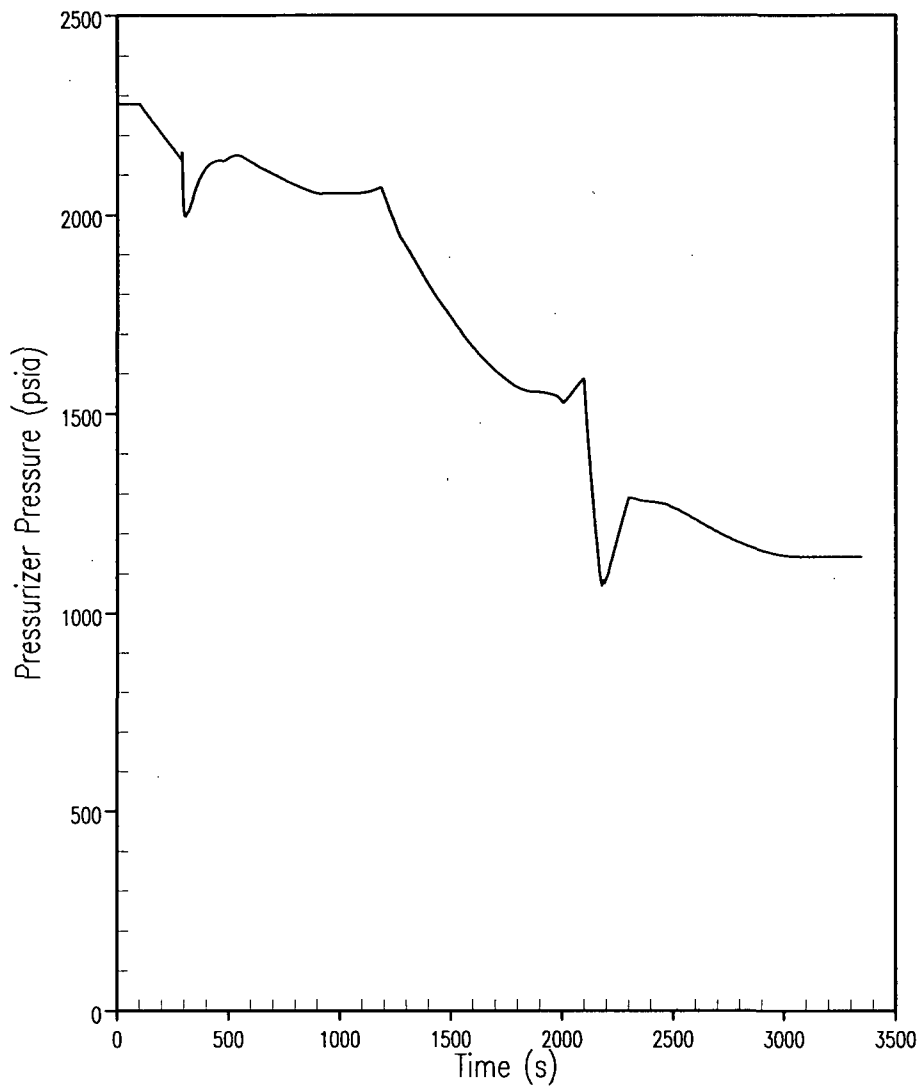


Figure 2.8.5.6.2-2 SGTR (Overfill), Pressurizer Pressure Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Margin to Overfill

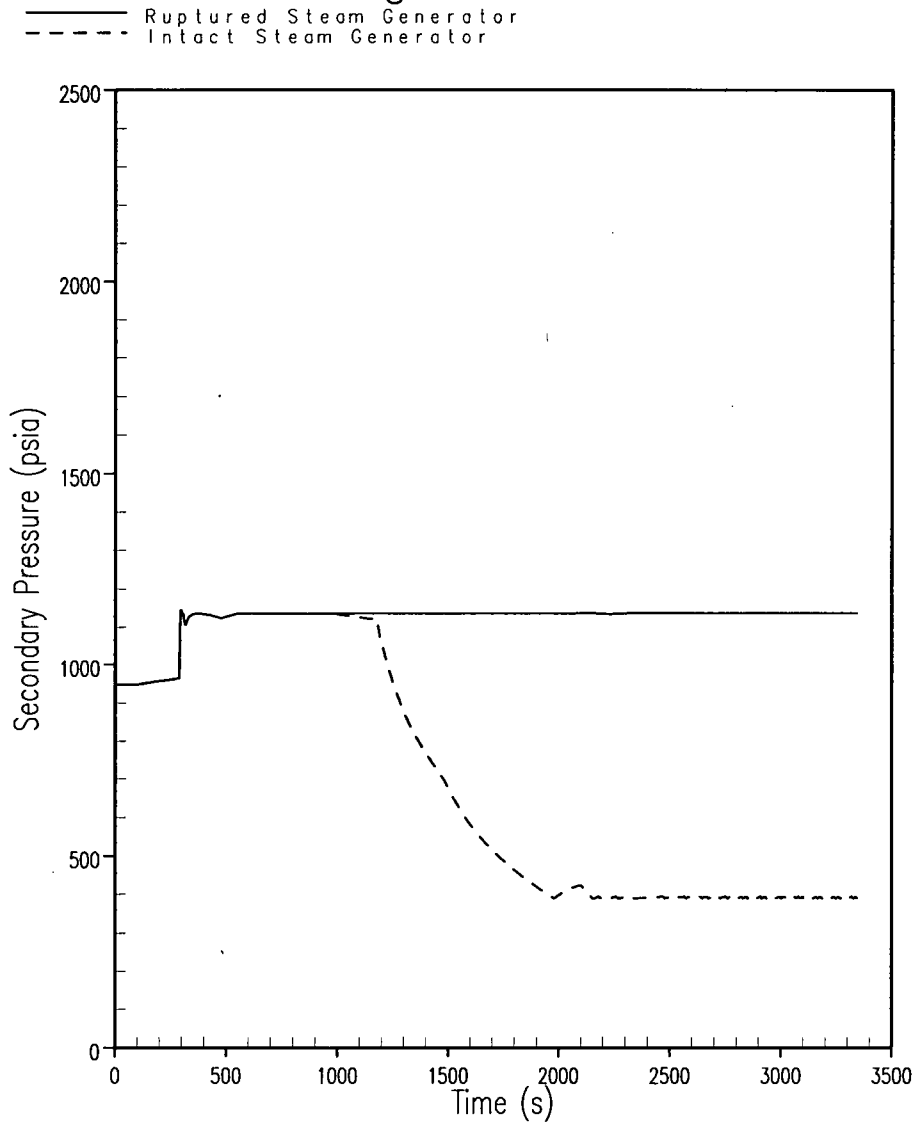


Figure 2.8.5.6.2-3 SGTR (Overfill), Secondary Pressure Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Margin to Overfill

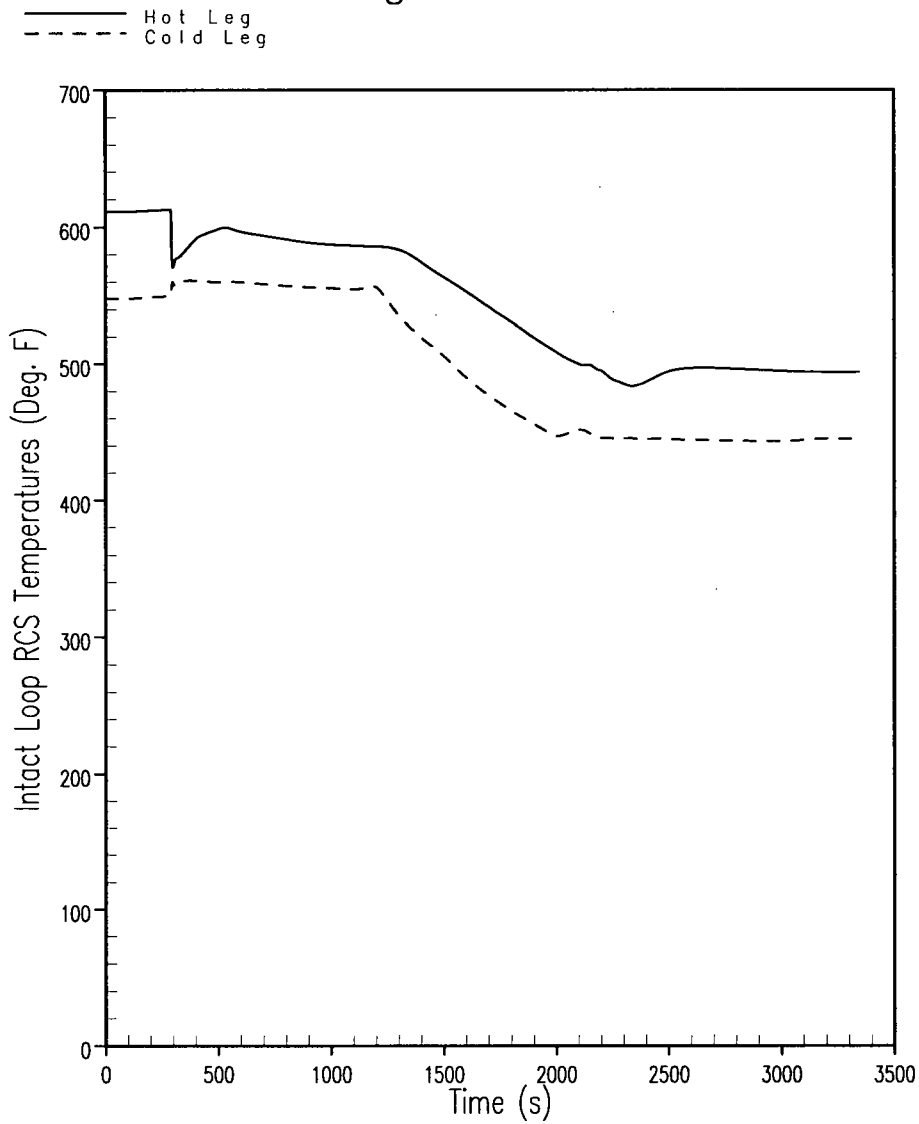


Figure 2.8.5.6.2-4 SGTR (Overfill), Intact Loop RCS Temperatures Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Margin to Overfill

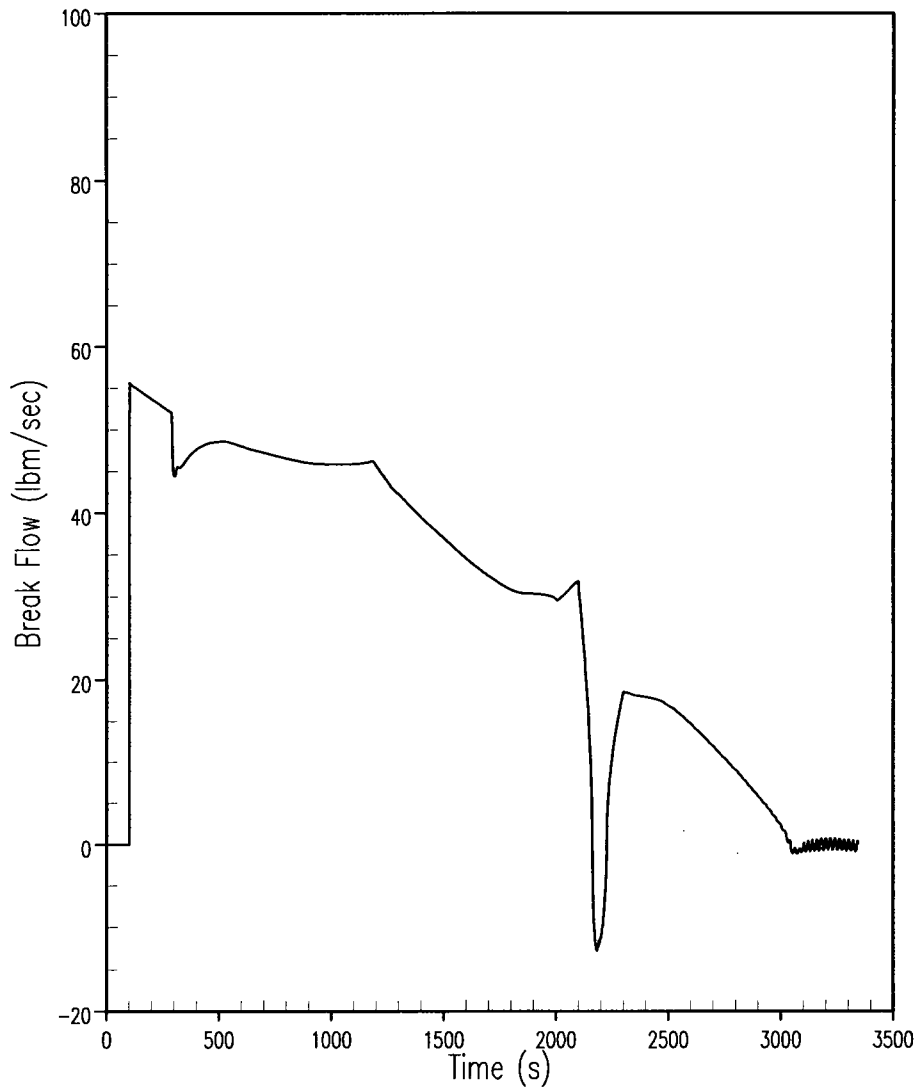


Figure 2.8.5.6.2-5 SGTR (Overfill), Break Flow Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Margin to Overfill

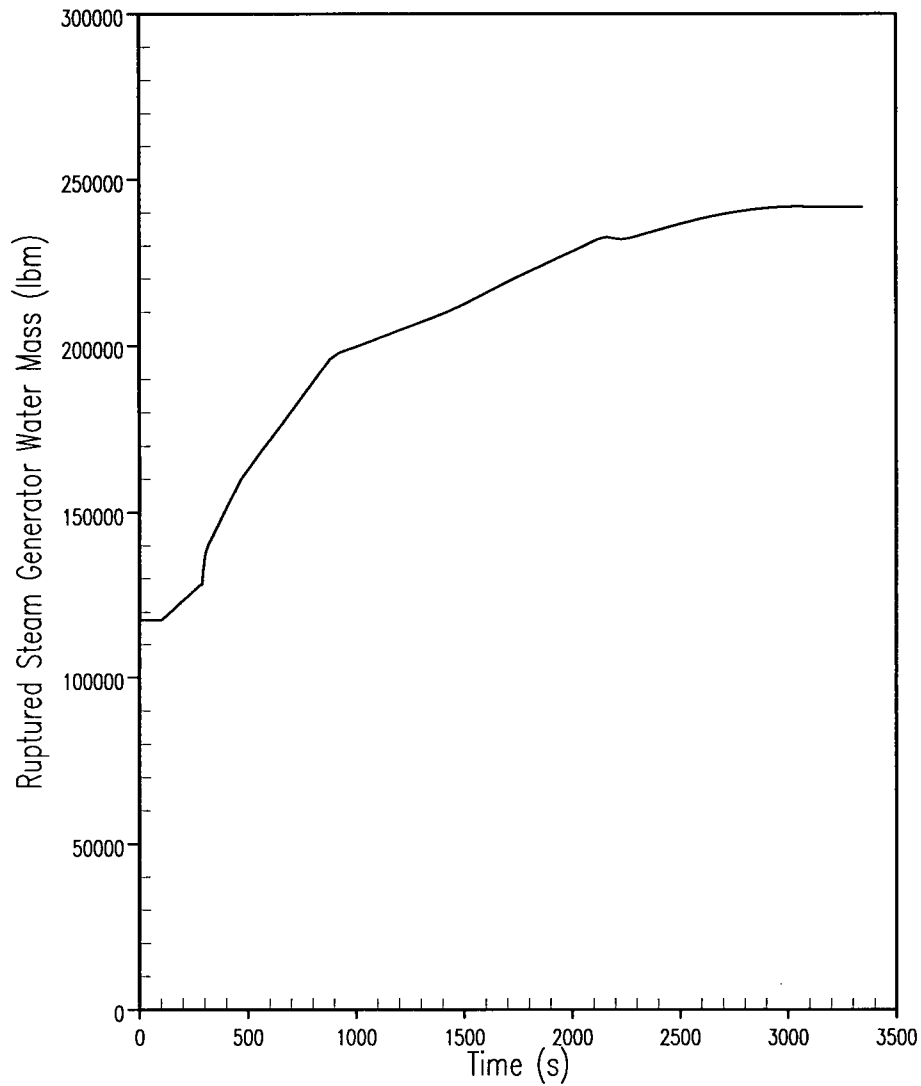


Figure 2.8.5.6.2-6 SGTR (Overfill), Ruptured Steam Generator Fluid Mass Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Margin to Overfill

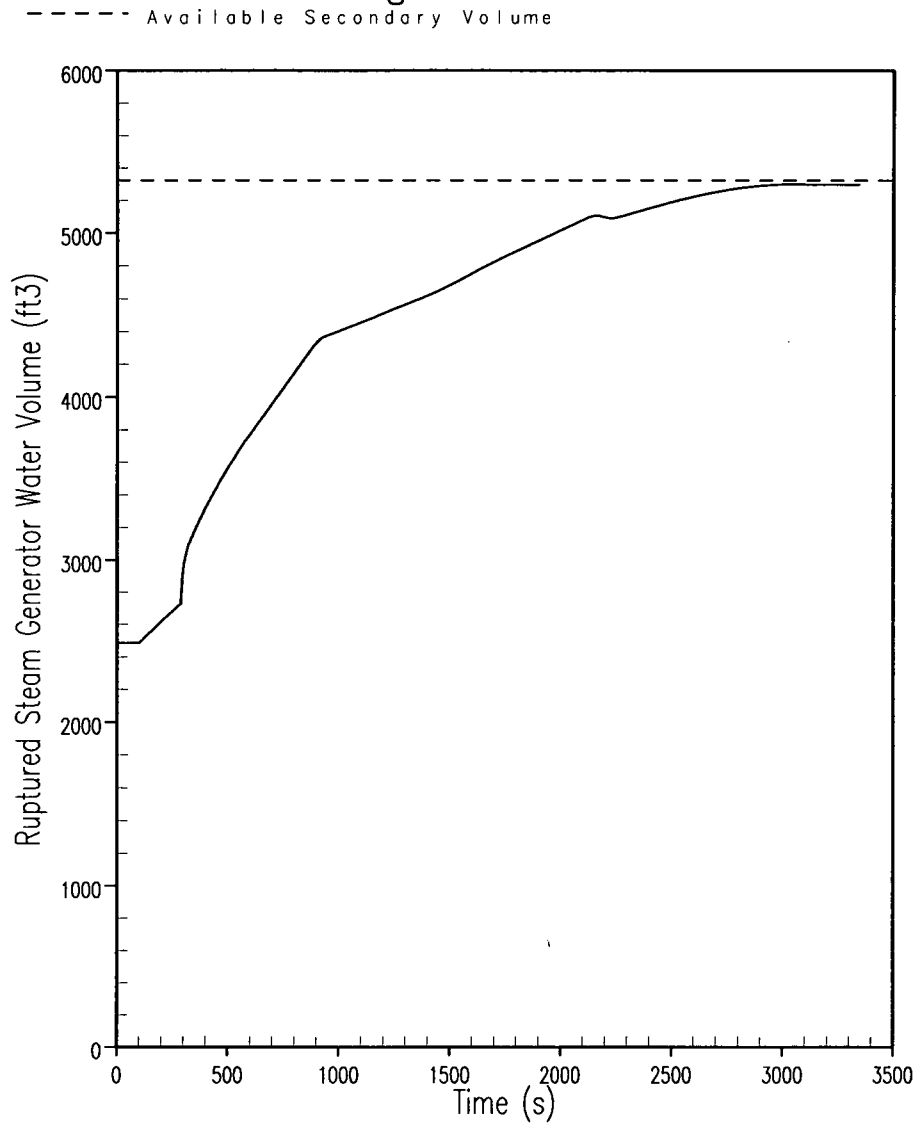


Figure 2.8.5.6.2-7 SGTR (Overfill), Ruptured Steam Generator Water Volume Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

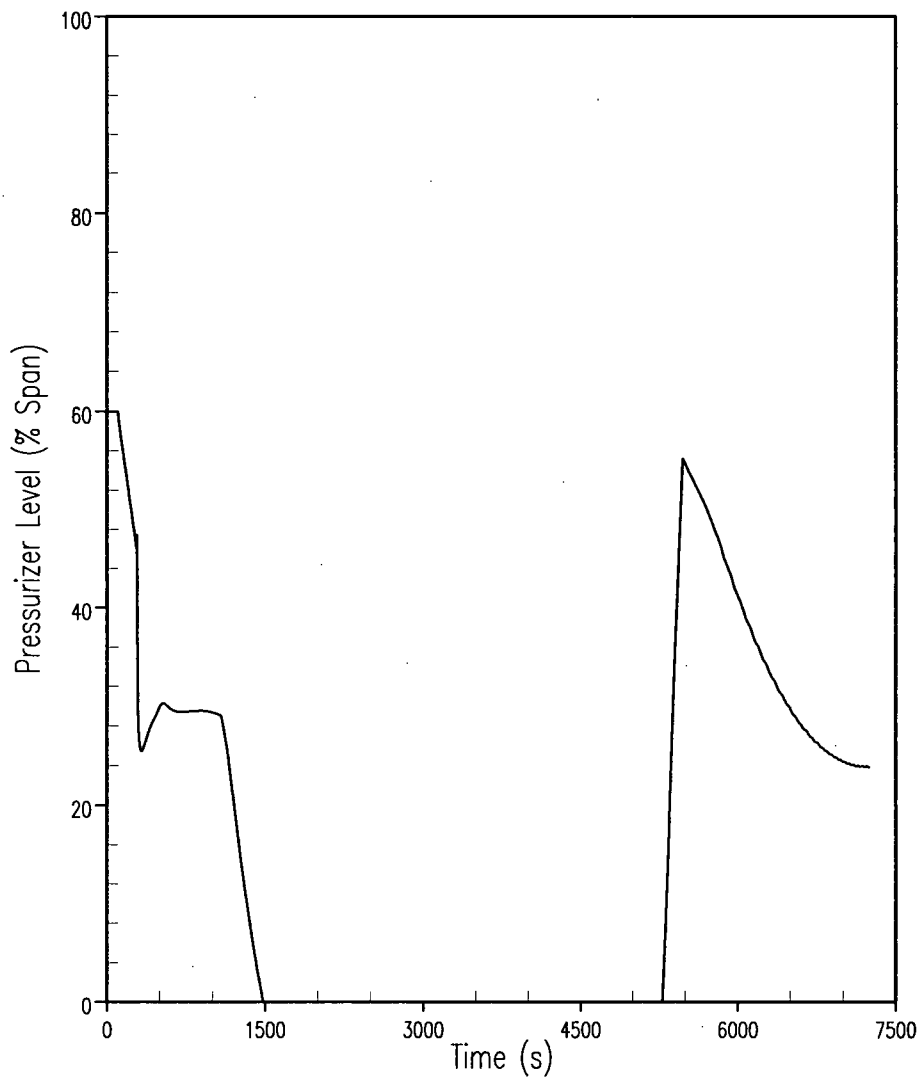


Figure 2.8.5.6.2-8 SGTR (Mass Release), Pressurizer Level Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

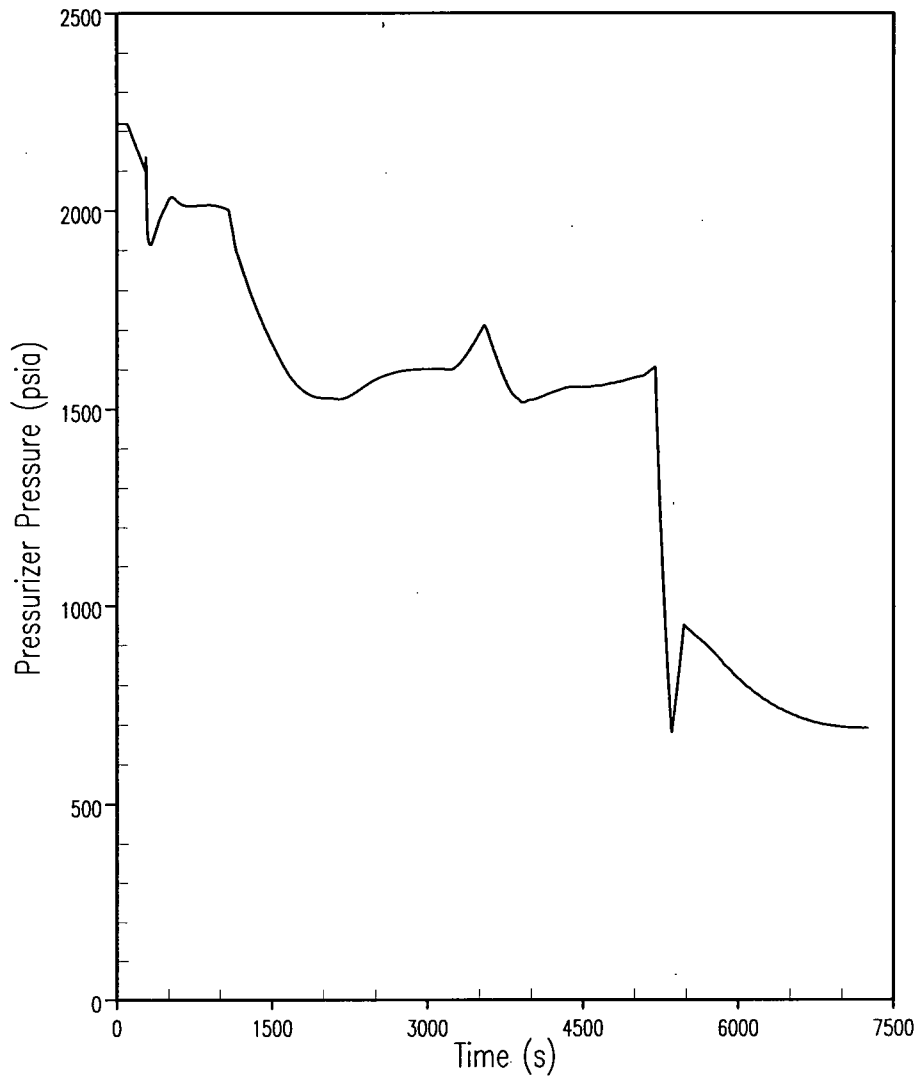


Figure 2.8.5.6.2-9 SGTR (Mass Release), Pressurizer Pressure Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

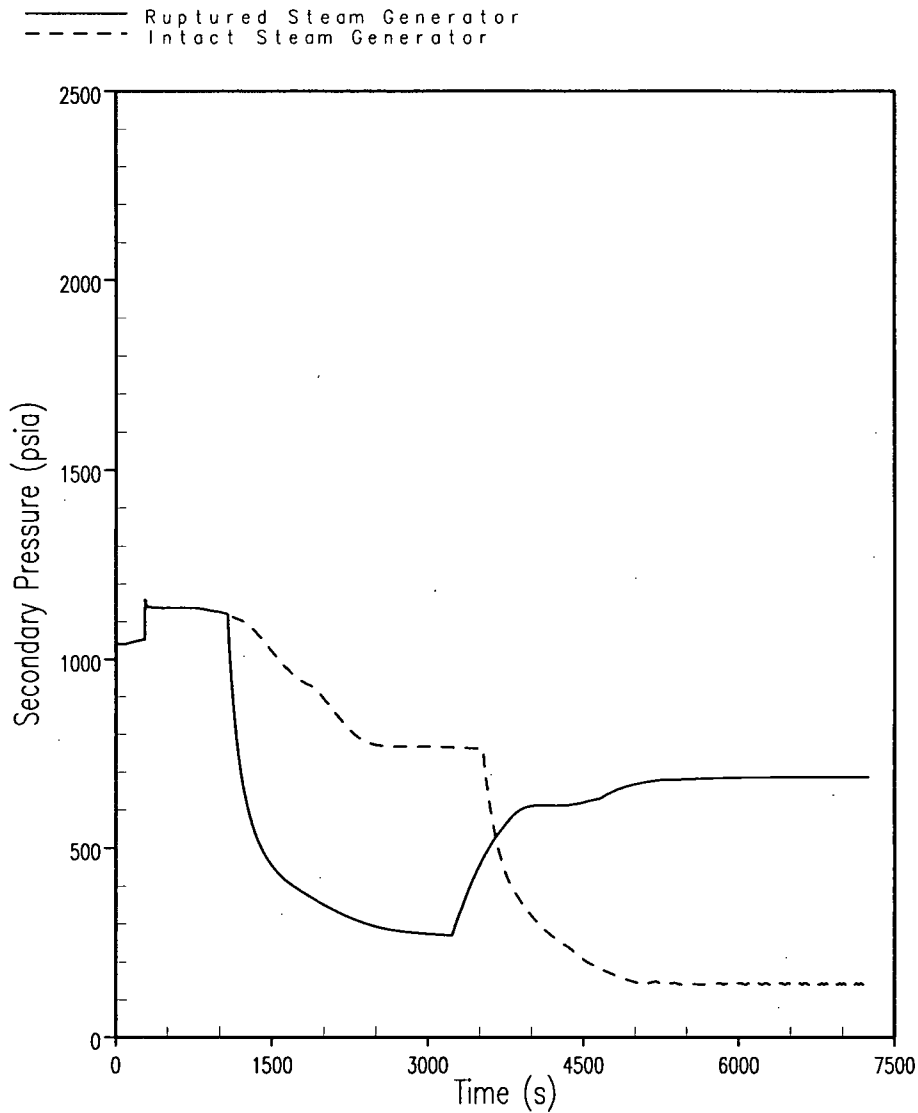


Figure 2.8.5.6.2-10 SGTR (Mass Release), Secondary Pressure Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

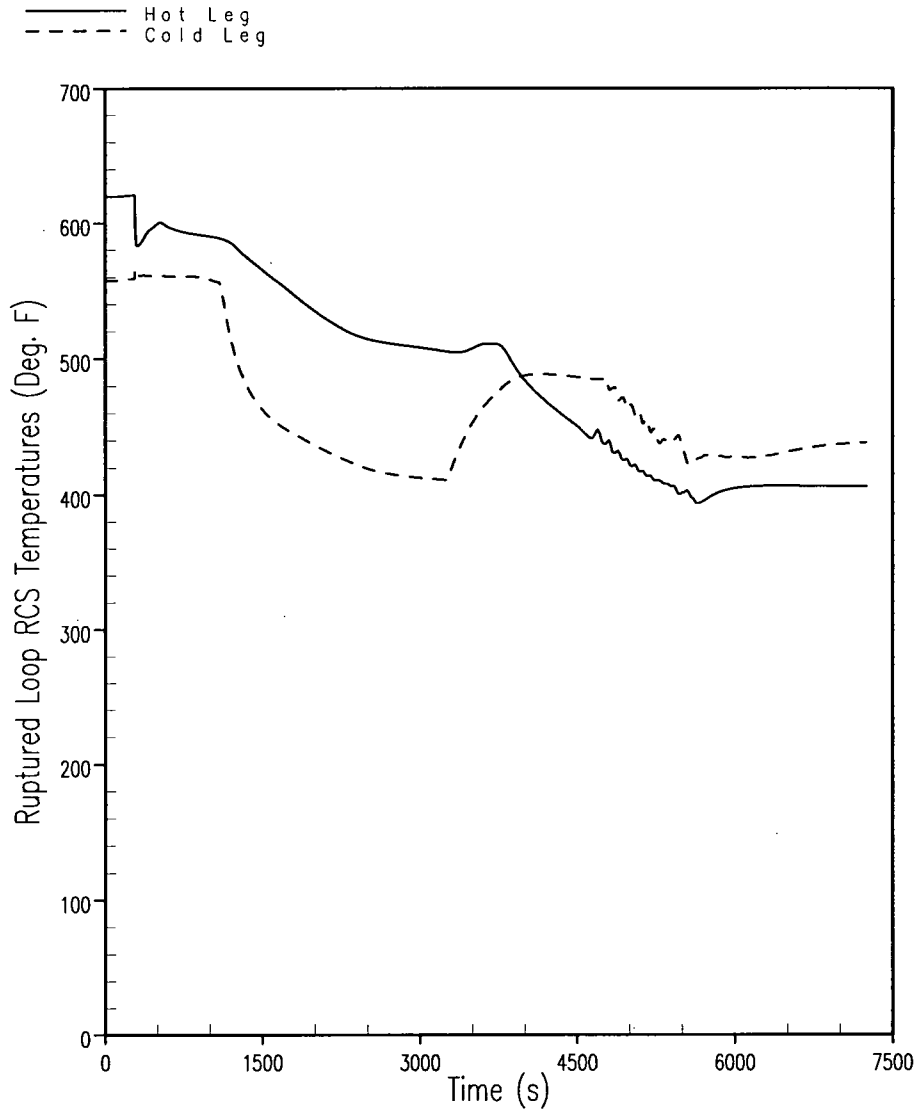


Figure 2.8.5.6.2-11 SGTR (Mass Release), Ruptured Loop RCS Temperatures Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

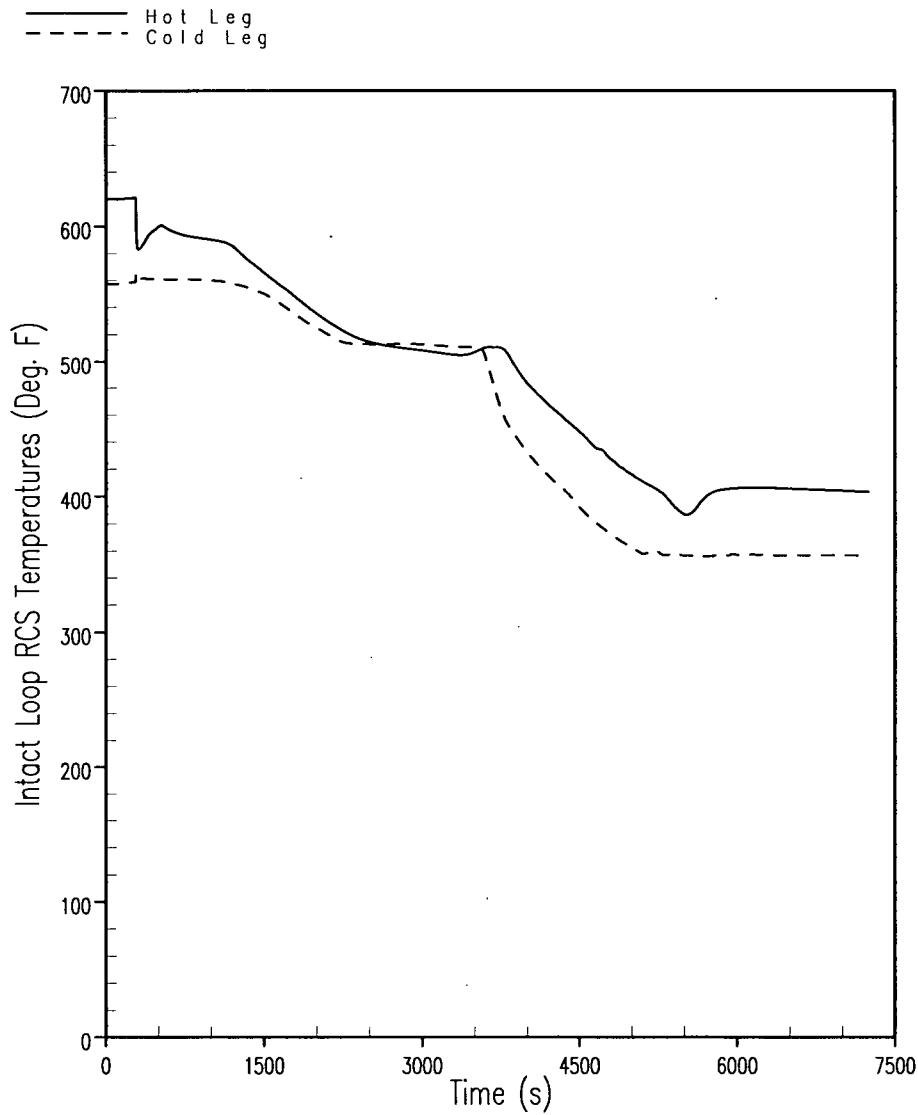


Figure 2.8.5.6.2-12 SGTR (Mass Release), Intact Loop RCS Temperatures Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

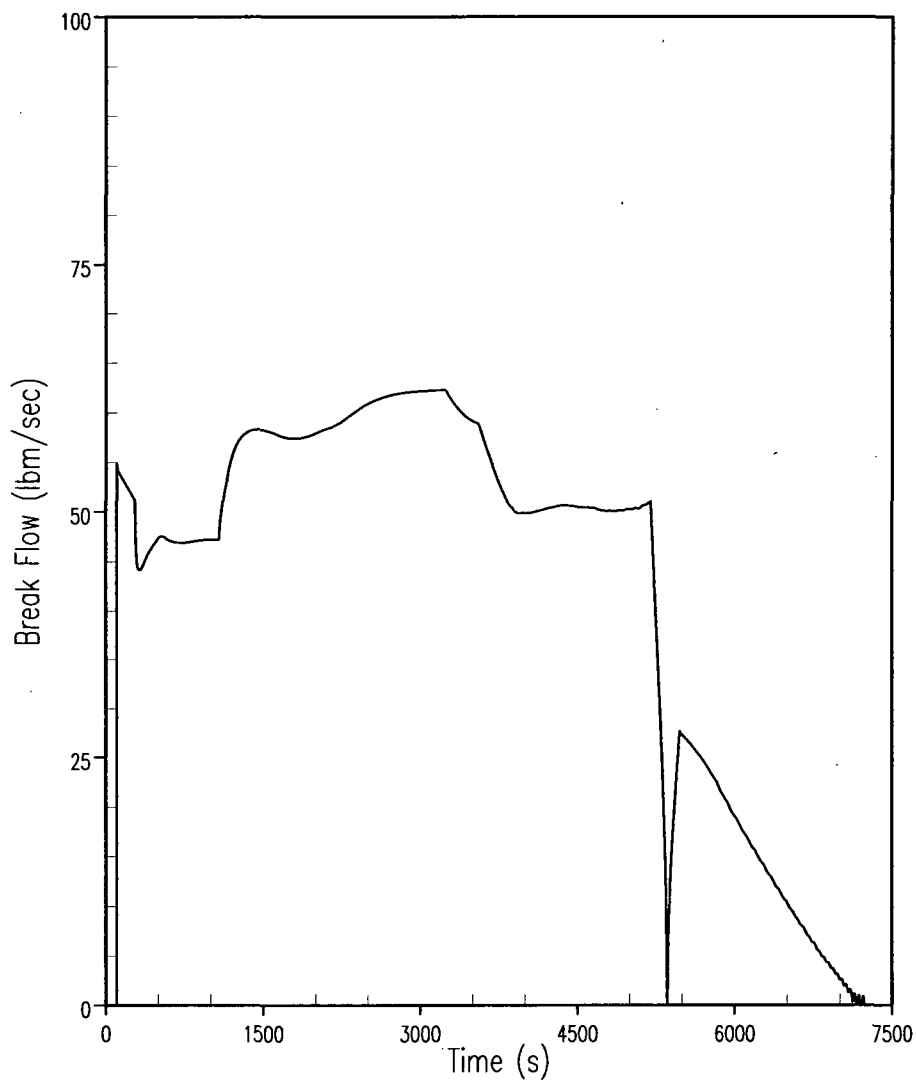


Figure 2.8.5.6.2-13 SGTR (Mass Release), Break Flow Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

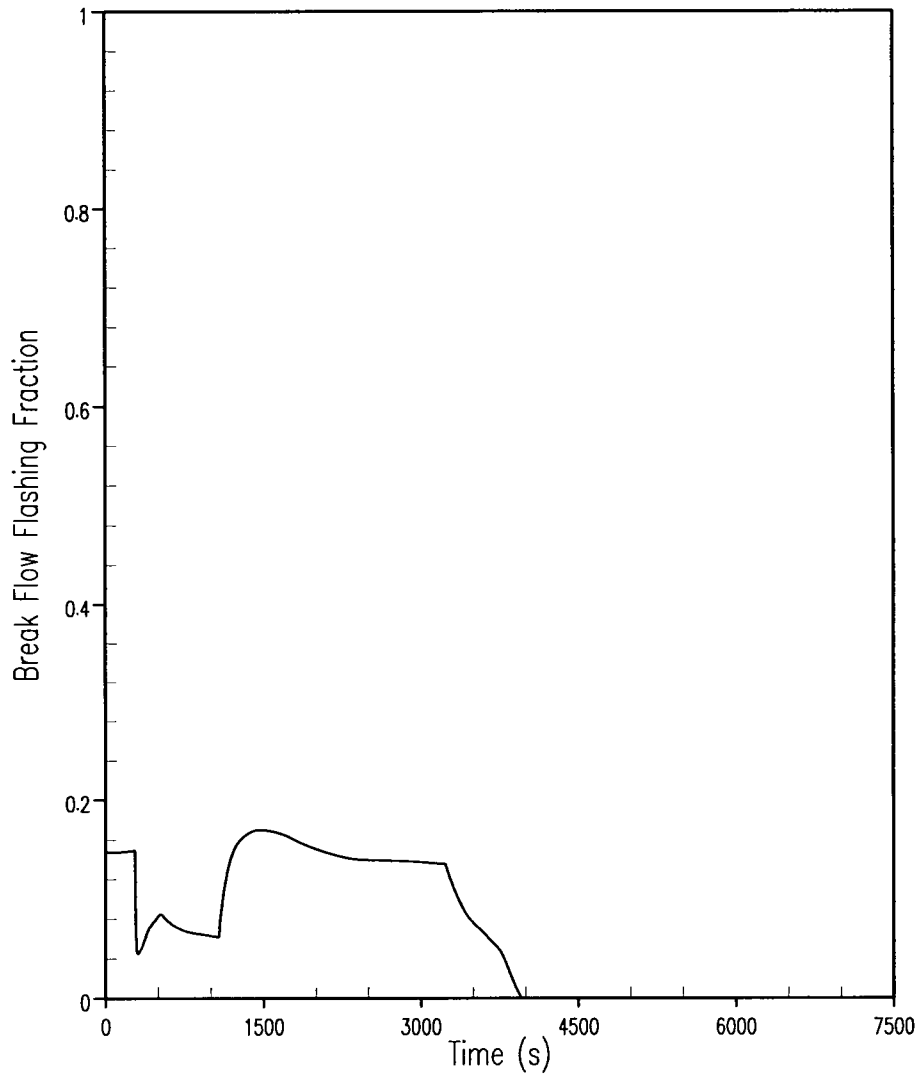


Figure 2.8.5.6.2-14 SGTR (Mass Release), Break Flow Flashing Fraction Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

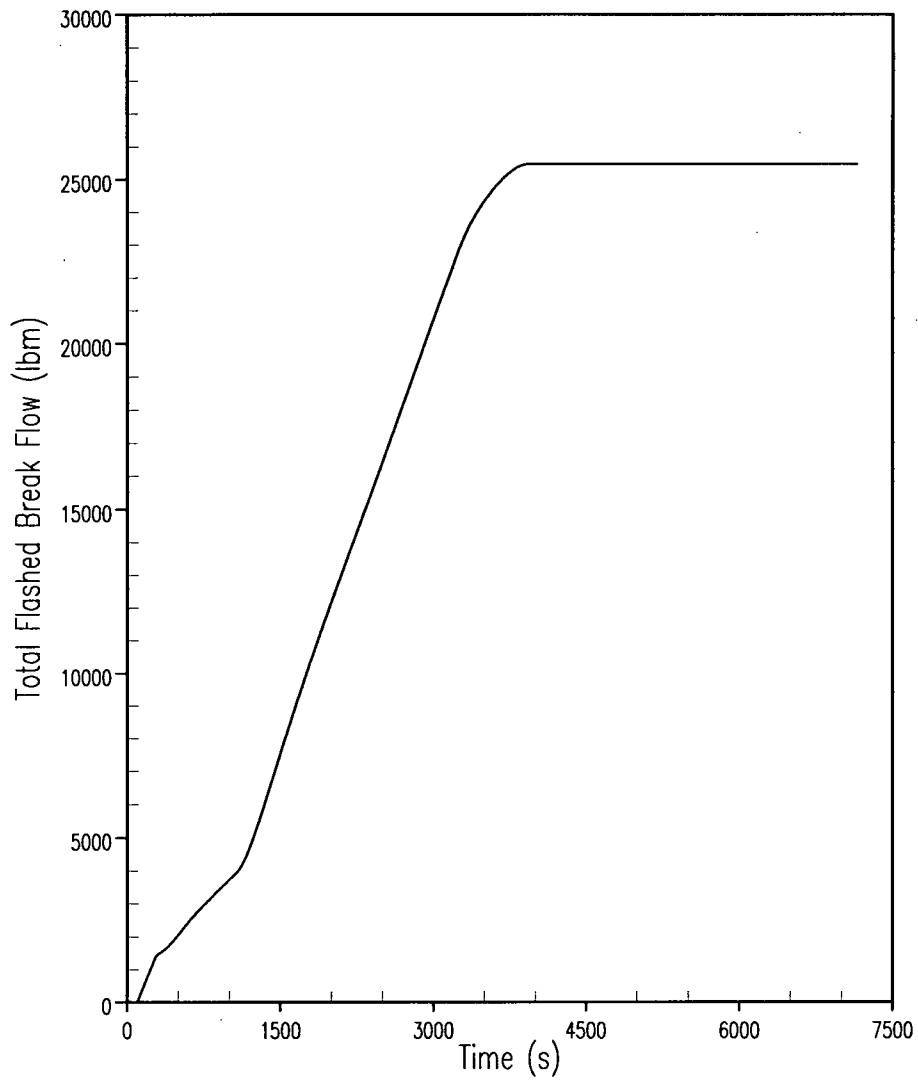


Figure 2.8.5.6.2-15 SGTR (Mass Release), Total Flashed Break Flow Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

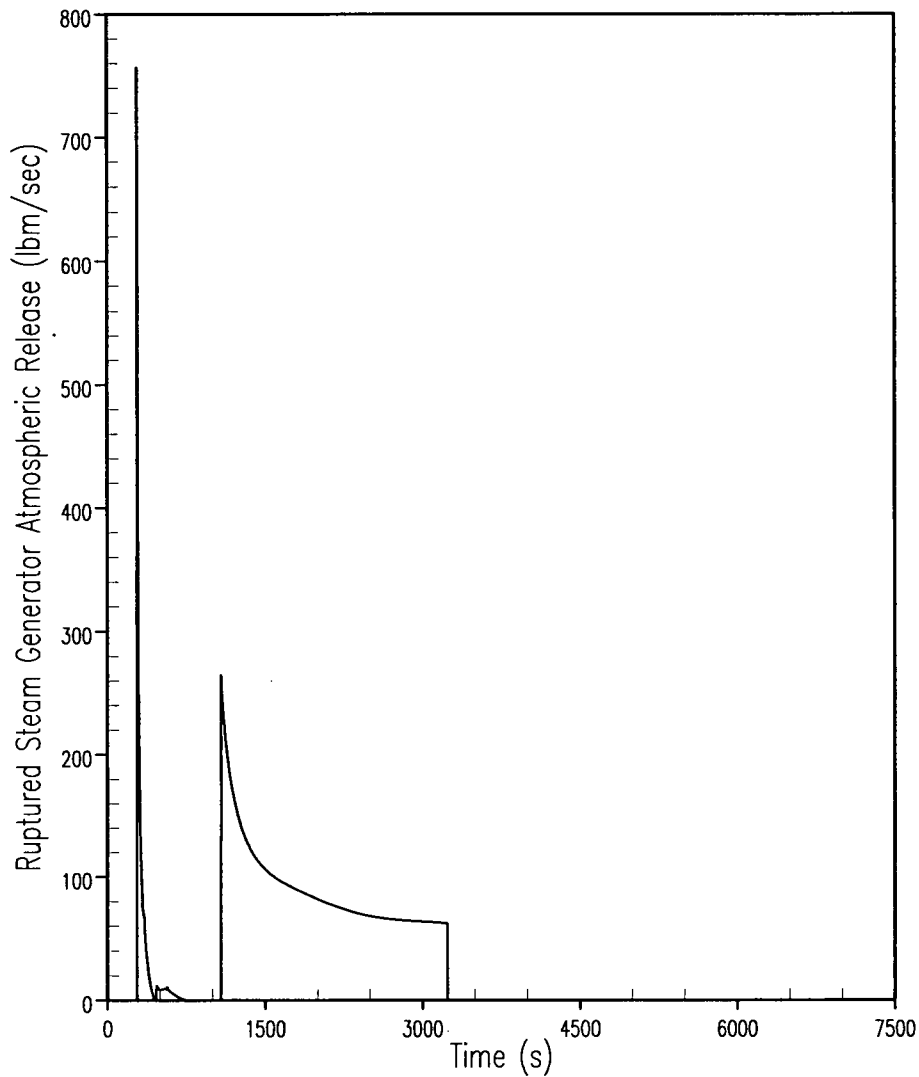


Figure 2.8.5.6.2-16 SGTR (Mass Release), Ruptured Steam Generator Mass Release Rate to the Atmosphere Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

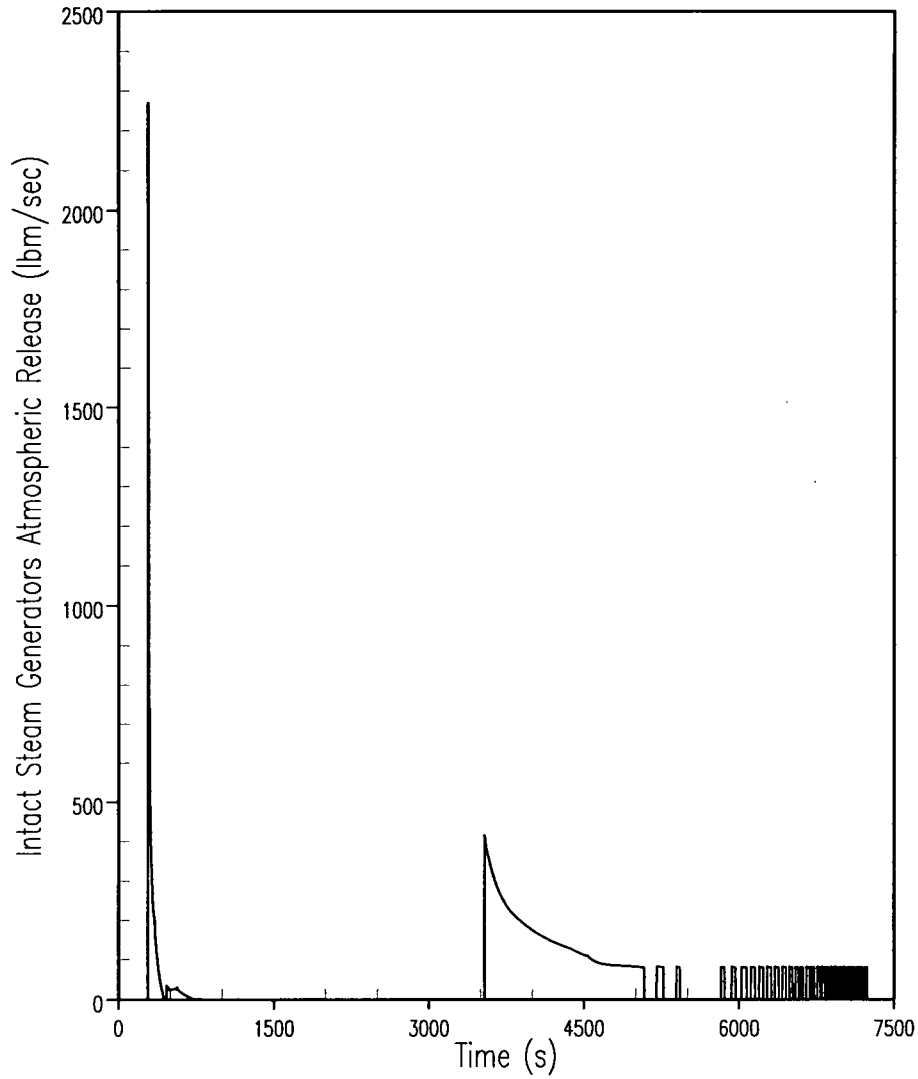


Figure 2.8.5.6.2-17 SGTR (Mass Release), Intact Steam Generator Mass Release Rate to the Atmosphere Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

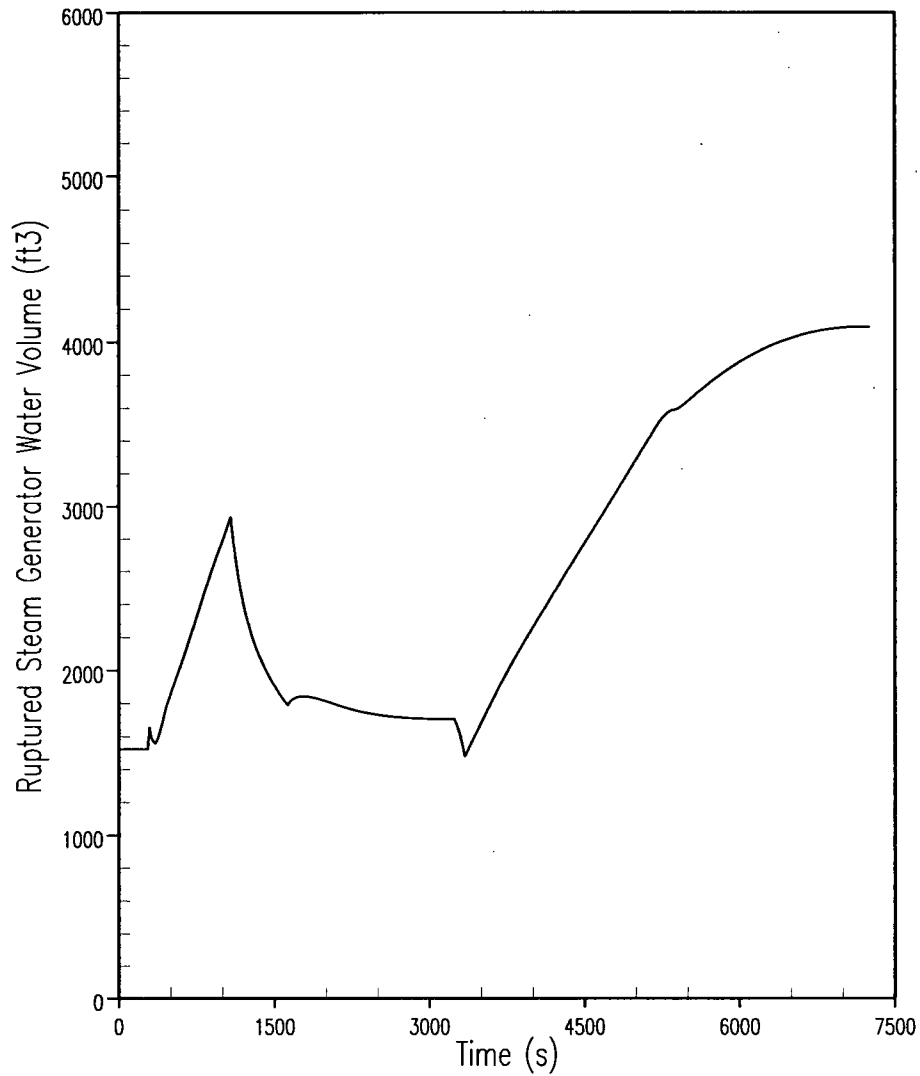


Figure 2.8.5.6.2-18 SGTR (Mass Release), Ruptured Steam Generator Water Volume Versus Time

Comanche Peak Unit 1 Steam Generator Tube Rupture Mass Release For Doses

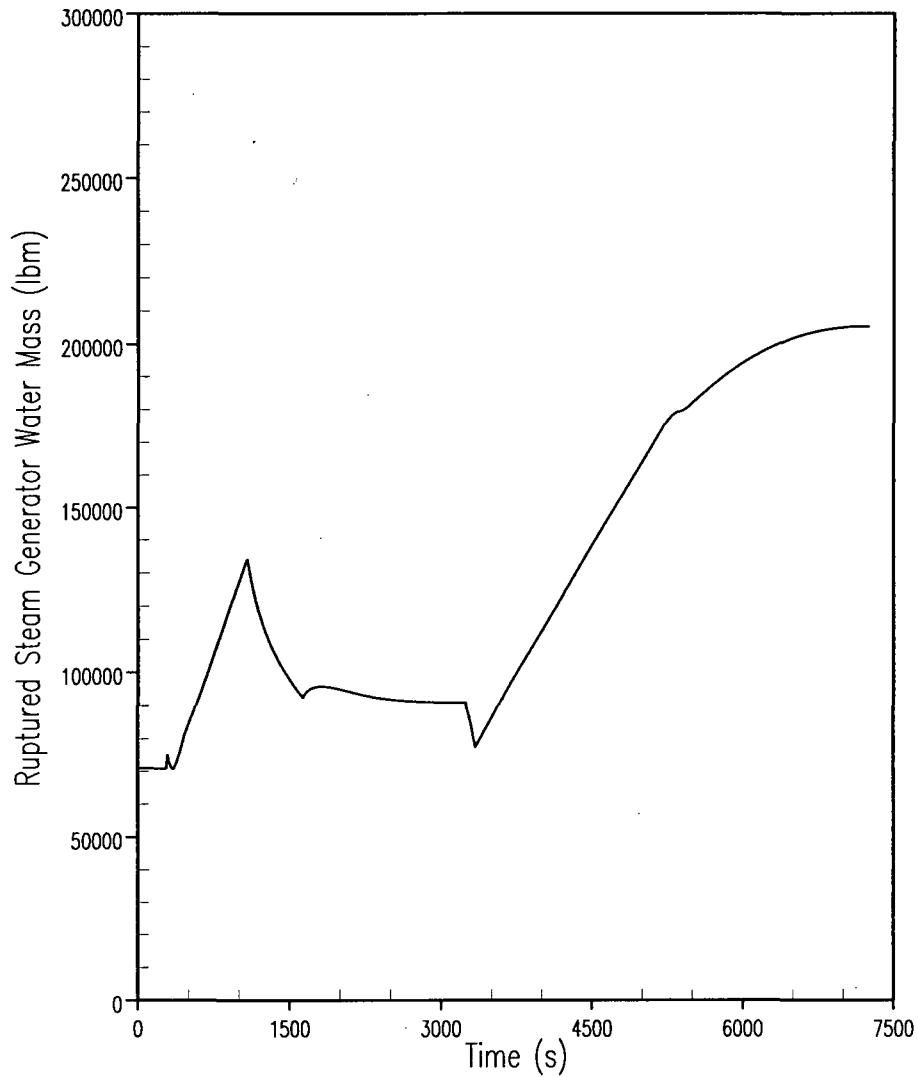


Figure 2.8.5.6.2-19 SGTR (Mass Release), Ruptured Steam Generator Water Mass Versus Time

2.8.5.6.3 Emergency Core Cooling System and Loss-of-Coolant Accidents

2.8.5.6.3.1 Regulatory Evaluation

Loss-of-coolant accidents (LOCAs) are postulated accidents that would result in the loss of reactor coolant from piping breaks in the reactor coolant pressure boundary (RCPB) at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection system (RPS) and emergency core cooling system (ECCS) are provided to mitigate these accidents.

The acceptance criteria are based on:

- 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance
- 10 CFR 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models by the ECCS after the blowdown phase of a LOCA
- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer
- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained
- GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel and cladding damage that could interfere with continued effective core cooling will be prevented.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the CPNPP design relative to:

- GDC-4, Environmental and Missile Design Bases, is described in the FSAR Section 3.1.1.4.

The station's SSCs important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents including LOCAs. Environmental conditions are described in FSAR Section 3.11.

- GDC-27, Combined Reactivity Control Systems Capability, is described in FSAR Section 3.1.3.8.

The facility is provided with a means of making the core subcritical and maintaining it at that level under any anticipated conditions and with an appropriate margin for contingencies. These means are discussed in detail in FSAR Chapters 4 and 9. Combined use of the rod cluster control assemblies and the chemical shim permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. Upon trip for this determination, the single highest worth control cluster is assumed to be stuck full-out upon trip.

- GDC-35, Emergency Core Cooling, is addressed in FSAR Section 3.1.4.6.

An ECCS is provided to cope with any LOCAs in the plant design basis. Abundant cooling water is available in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to ensure that clad metal-water reaction is limited to less than 1 percent. Adequate design provisions are made to assure performance of the required safety functions even with a single failure.

Details of the capability of the systems are included in FSAR Section 6.3. An evaluation of the adequacy of the safety functions is included in FSAR Chapter 15. Performance evaluations are conducted in accordance with 10 CFR Part 50, Section 50.46 and Appendix K.

2.8.5.6.3.2 Large-Break LOCA

This section discusses the large-break best-estimate LOCA (BELOCA) analysis to support the SPU program for CPNPP.

2.8.5.6.3.2.1 Introduction – LBLOCA

The LBLOCA is described in Licensing Report (LR) subsection 2.8.5.6.3.2.3 for a major rupture of the RCPB. A major rupture (large break) is defined as a breach in the RCPB with a total cross-sectional area greater than 1.0 ft².

A best-estimate LOCA analysis has been completed for the CPNPP Units 1 and 2. This license amendment request (LAR) for CPNPP requests approval to apply the Westinghouse BELOCA analysis methodology.

Westinghouse recently underwent a program to revise the statistical approach used to develop the peak cladding temperature (PCT) and oxidation results at the 95th percentile. This method is still based on the Code Qualification Document (CQD) technology (Reference 1) and follows the steps in the Code Scaling, Applicability, and Uncertainty (CSAU) methodology (Reference 2). However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The Automated Statistical Treatment of Uncertainty Method (ASTRUM) methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The approved ASTRUM evaluation model is documented in WCAP-16009 (Reference 3).

LR subsections 2.8.5.6.3.2.2 through 2.8.5.6.3.2.4 summarize the application of the Westinghouse ASTRUM BELOCA evaluation model to the CPNPP analysis of the BELOCA event. The analysis was performed in compliance with all the NRC conditions and limitations as identified in WCAP-16009 (Reference 3). Luminant Power and its vendor, Westinghouse Electric Company LLC, continue to have ongoing processes that ensure that LOCA analysis input values conservatively bound the as-operated plant values for those parameters.

2.8.5.6.3.2.2 Input Parameters, Assumptions, and Acceptance Criteria – LBLOCA

Table 2.8.5.6.3.2-1 lists the major plant parameter assumptions used in an as-approved ASTRUM BELOCA analysis. The major plant parameter assumptions for CPNPP Unit 1 and 2 are included in the safety analysis transition submittal to the NRC. The acceptance criteria are discussed in LR subsection 2.8.5.6.3.2.4.

2.8.5.6.3.2.3 Description of Analyses – Large Break LOCA

Westinghouse developed an uncertainty methodology called ASTRUM (Reference 3). This method is based on the CQD methodology (Reference 1) and follows the steps in the CSAU methodology (Reference 2). However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in WCAP-16009 (Reference 3). The ASTRUM methodology remains applicable to three- and four-loop PWRs, as well as 2-loop Westinghouse plants with upper plenum injection (UPI).

The three 10 CFR 50.46 criteria (PCT, LMO, and CWO) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size).

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13.3 of WCAP-16009 (Reference 3), as applicable to the ASTRUM methodology. Section 13.3 of WCAP-16009 (Reference 3) was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report.

2.8.5.6.3.2.4 Results – LBLOCA

The description of the limiting PCT, LMO, and CWO transients and the results for the CPNPP ASTRUM analyses are included in the safety analysis transition submittal to the NRC (TXX-07107 and TXX-07108).

10 CFR 50.46 Requirements

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

- (b)(1) The limiting PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level. The resulting PCT for the limiting CPNPP case confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., “Peak Clad Temperature less than 2,200°F,” is demonstrated.
- (b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95th percentile LMO at the 95-percent confidence level. The resulting LMO (considering both transient and pre-transient oxidation) for the limiting CPNPP case confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., “Local Maximum Oxidation of the cladding less than 17 percent,” is demonstrated.
- (b)(3) The limiting CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The resulting CWO for the limiting CPNPP case confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., “Core-Wide Oxidation less than 1 percent,” is demonstrated.
- (b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for best-estimate LOCA applications. The approved methodology (Reference 1) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the 44 assemblies in the low-power channel. This situation is not calculated to occur for CPNPP. Therefore, acceptance criterion (b)(4) is satisfied.
- (b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at these plants to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology (Reference 3).

2.8.5.6.3.2.5 References

1. WCAP-12945, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1, and WCAP-14747, "Code Qualification Document for Best-Estimate LOCA Analysis," 1998.
2. NUREG/CR-5249, "Qualifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident," 1989.
3. WCAP-16009, "Realistic Large Break LOCA Evaluation Methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," 2005.
4. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 4, 1974.

Table 2.8.5.6.3.2-1	
Major Plant Parameter Assumptions Used in Best-Estimate LBLOCA ASTRUM Analysis	
Parameter	Bias
Plant Physical Description	
• Steam Generator Tube Plugging	Maximum
Plant Initial Operating Conditions	
• Reactor Power	Maximum (nominal + uncertainties)
• Peaking Factors	Max F_Q Max $F_{\Delta H}$
• Axial Power Distribution	PBOT/PMID Envelop
Fluid Conditions	
• T_{avg}	Nominal \pm uncertainties
• Pressurizer Pressure	Nominal \pm uncertainties
• Reactor Coolant Flow	Nominal
• Accumulator Temperature	Nominal \pm uncertainties
• Accumulator Pressure	Nominal \pm uncertainties
• Accumulator Water Volume	Nominal \pm uncertainties
• Accumulator Boron Concentration	Minimum
Accident Boundary Conditions	
• Single Failure Assumptions	Loss of one ECCS train
• Safety Injection Flow	Minimum
• Safety Injection Temperature	Nominal \pm uncertainties
• Safety Injection Initiation Delay Time	Maximum delay with offsite power Maximum delay without offsite power
• Containment Pressure	Minimum

2.8.5.6.3.3 Small-Break LOCA

2.8.5.6.3.3.1 Introduction

A loss-of-coolant accident (LOCA) is defined as a rupture of the reactor coolant system (RCS) piping or of any line connected to the system. The small-break LOCA (SBLOCA) includes all postulated pipe ruptures with a total cross-sectional area less than 1.0 ft². The SBLOCAs analyzed in this section are for those breaks beyond the capability of a single charging pump

resulting in the actuation of the emergency core cooling system (ECCS). The analysis was performed to demonstrate conformance with the 10 CFR 50.46 requirements for the conditions associated with the CPNPP Units 1 and 2 SPU Program.

2.8.5.6.3.3.2 Input Parameters, Assumptions and Acceptance Criteria

Table 2.8.5.6.3.3.2-1 lists the key input parameters and assumptions used in a NOTRUMP-EM SBLOCA analysis. The key input parameters and assumptions for CPNPP are included in the Safety Analysis Transition submittal to the Nuclear Regulatory Commission (NRC).

The acceptance criteria for the SBLOCA analysis are specified in 10 CFR 50.46, as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2,200°F.
2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. (Note that this criterion is not addressed as part of the short-term SBLOCA analysis; the post-LOCA long term cooling analysis addresses this acceptance criterion.)

2.8.5.6.3.3.3 Description of Analyses

The SBLOCA analysis was performed for the CPNPP Unit 1 and 2 SPU Program using the 1985 Westinghouse SBLOCA Evaluation Model with NOTRUMP (NOTRUMP-EM) (References 1 – 3), including NRC approved changes to the methodology as described in References 4 and 5. Westinghouse obtained generic NRC approval of the NOTRUMP computer code's modeling capabilities and solution techniques (Reference 1) and the use of the NOTRUMP computer code for licensing applications (Reference 2) in 1985. NRC approval of additional modeling details (Reference 3), such as limiting break location was obtained in 1986. The NOTRUMP-EM was later revised (Reference 4) and granted generic NRC approval for an improved condensation model and related changes in safety injection modeling assumptions for safety injection to the reactor coolant system (RCS) cold legs. Most recently, the NRC generically approved updates to the NOTRUMP-EM to include the ability to model annular fuel pellets (Reference 5) in the fuel rod heatup calculations.

2.8.5.6.3.3.4 Results

The results for the CPNPP NOTRUMP-EM SBLOCA analysis are included in the Safety Analysis Transition submittal to the NRC (TXX-07107 and TXX-07108).

2.8.5.6.3.3.5 References

1. WCAP-10079, "NOTRUMP - A Nodal Transient Small Break and General Network Code," August 1985.
2. WCAP-10054, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
3. WCAP-11145, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," October 1986.
4. WCAP-10054, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.
5. WCAP-14710, "1-D Heat Conduction Model for Annular Fuel Pellets," May 1998.

Table 2.8.5.6.3.3.2-1

Key Plant Parameters and Assumptions Used in Appendix K SBLOCA NOTRUMP-EM Analysis

Parameter	Bias
A. Core Parameters	
Analyzed Core Power Level	Maximum (Nominal + Uncertainties)
Total Core Peaking Factor, F_O	Maximum
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	Maximum
Hot Assembly Average Rod Power, P_{HA}	Maximum
Axial Offset	Maximum
B. Reactor Coolant System	
Pressurizer Pressure	Maximum (Nominal + Uncertainties) ⁽¹⁾
C. Reactor Protection System	
Reactor Trip Setpoint	Minimum
D. Steam Generators	
Steam Generator Tube Plugging	Maximum ⁽¹⁾
E. Safety Injection (SI)	
Limiting Single Failure	Loss of one Emergency Diesel Generator
SI Water Temperature	Maximum
SI Delay Time	Maximum
Safety Injection Flow Rates	Minimum
F. Accumulators	
Water/Gas Temperature	Maximum
Cover Gas Pressure	Minimum
<p>Notes:</p> <p>1. These parameters are analyzed at the maximum value, but are generally inconsequential to SBLOCA analysis results.</p>	

2.8.5.6.3.4 Post-LOCA Subcriticality

2.8.5.6.3.4.1 Regulatory Evaluation

In support of the CPNPP Units 1 and 2 SPU Program, post-LOCA subcriticality sump boron calculations were performed. The methodology used to demonstrate CPNPP compliance with the requirements of 10 CFR 50.46 Paragraph (b), Item (5), is documented in WCAP-8339 (Reference 1). Reference 1 states that the core will remain subcritical post-LOCA by borated water from the various injected emergency core cooling system (ECCS) water sources. Post-LOCA sump boron calculations demonstrate that the core will remain subcritical upon entering and during the sump recirculation phase of ECCS injection. Containment sump boron concentration calculations were used to develop a core reactivity limit that was confirmed as part of the Westinghouse Reload Safety Evaluation Methodology (Reference 2).

2.8.5.6.3.4.2 Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The input parameters and assumptions used in the sump boron calculations are given in Table 2.8.5.6.3.4-1.

The sump boron concentration calculational model is based on the following assumptions:

- The calculation of the sump mixed mean boron concentration assumes minimum mass and minimum boron concentrations for significant boron sources and maximum mass and minimum boron concentration for significant dilution sources.
- Boron is mixed uniformly in the sump. The post-LOCA sump inventory is made up of constituents that are equally likely to return to the containment sump; that is, selective holdup in containment is neglected.
- The sump mixed mean boron concentration is calculated as a function of the pre-trip RCS conditions.

There are no specific acceptance criteria when calculating the post-LOCA sump boron concentration. However, the resulting sump boron concentration, which is calculated as a function of the pre-LOCA RCS boron concentration, is reviewed for each cycle-specific core design to confirm that adequate boron exists to maintain subcriticality in the long-term post-LOCA.

Description of Analyses and Evaluations

With respect to post-LOCA criticality, a post-LOCA subcriticality boron limit curve was developed for the SPU plant conditions. Provided that the cycle-specific maximum critical boron concentration remains below the post-LOCA sump boron concentration limit curve (for all

rods out, no Xenon, 68 – 212°F), the core will remain subcritical post-LOCA, and decay heat can be removed for the extended period required by the remaining long-lived radioactivity.

Results

A post-LOCA subcriticality boron limit curve was developed for the SPU plant conditions. The CPNPP SPU Post-LOCA subcriticality boron limit curve is shown in Figure 2.8.5.6.3.4-1.

2.8.5.6.3.2.4.3 Conclusion

Cycle-specific reload safety evaluations will ensure that the core will remain subcritical post-LOCA, thus addressing the GDC-27 requirement that the capability to cool the core is maintained.

2.8.5.6.3.2.4.4 References

1. WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model - Summary," June 1974.
2. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

Table 2.8.5.6.3.4-1	
CPNPP Units 1 and 2 SPU Input Parameters	
Parameter	SPU Value
RWST Boron Concentration, Minimum (ppm)	2,400
Accumulator Boron Concentration, Minimum (ppm)	2,300
RWST Volume, Assumed Minimum (gallons)	440,300

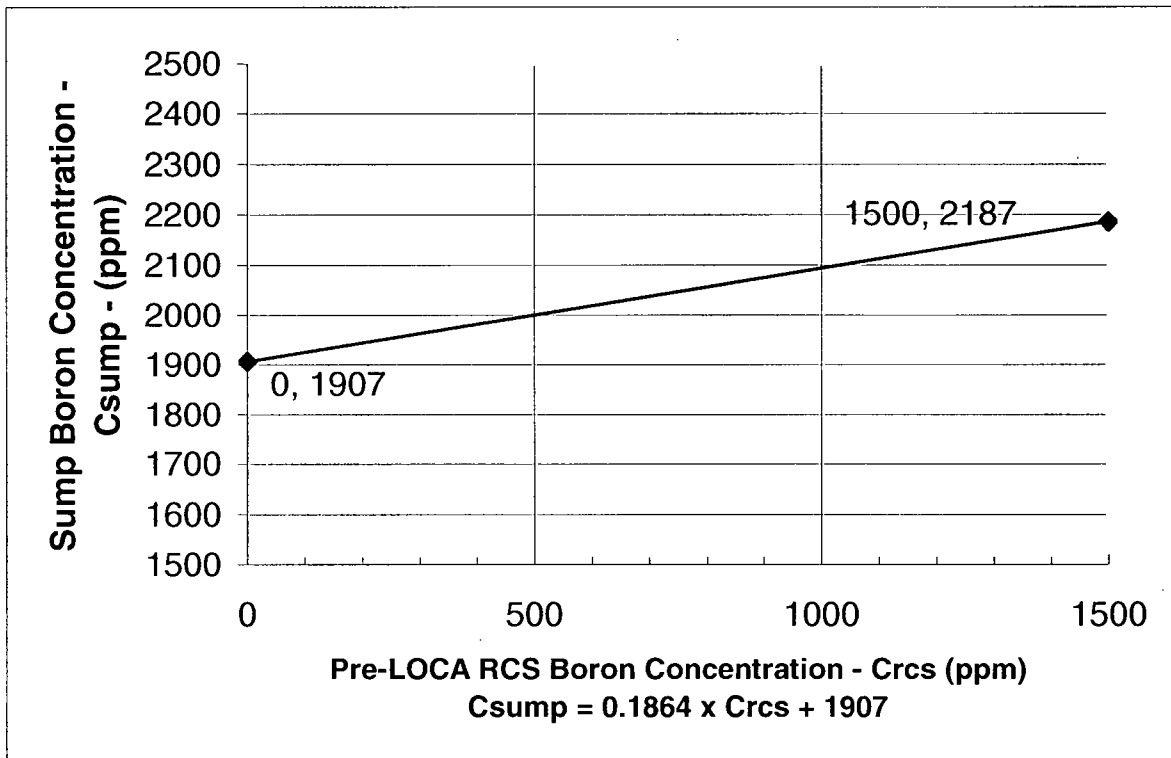


Figure 2.8.5.6.3.4-1 CPNPP SPU Post-LOCA Subcriticality Boron Limit Curve

2.8.5.6.3.5 Post-LOCA Long-Term Cooling

2.8.5.6.3.5.1 Regulatory Evaluation

In support of the SPU, a post-LOCA long-term cooling analysis was performed. There are two aspects to a long-term cooling analysis – the potential for boric acid precipitation and maintaining long-term decay heat removal. This analysis satisfies the requirements of 10 CFR 50.46 Paragraph (b), Item (4) and 10 CFR 50.46, Paragraph (b), Item (5).

The injection and sump recirculation ECCS modes are described in FSAR Section 6.3. Boric acid precipitation during long term-cooling is addressed in FSAR Section 6.3.2.5.4. Operator actions to prevent boric acid precipitation are described in the FSAR Appendices A and B. The switchover from injection mode to cold leg recirculation mode and the switchover from cold leg recirculation mode to hot leg recirculation mode are described in FSAR Section 6.3.2.8 and Table 6.3-7.

2.8.5.6.3.5.2 Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The major inputs to the boric acid precipitation calculation include core power assumptions and assumptions for boron concentrations and water volume/masses for significant contributors to the containment sump. The input parameters used in the CPNPP SPU boric acid precipitation calculations are given in Table 2.8.5.6.3.5-1.

The boric acid precipitation calculation model is based on the following assumptions and meets NRC guidance as presented in Reference 3 and is consistent with the interim methodology reported in Reference 4.

- The boric acid concentration in the core region is computed over time with consideration of the effect of core voiding on liquid mixing volume. Voiding is calculated using the Modified Yeh Correlation described in Reference 1.
- The core mixing volume used in the calculations considered the potential negative effects of loop pressure drop.
- The boric acid concentration limit is the experimentally determined boric acid solubility limit as reported in Reference 2 and summarized in Table 2.8.5.6.3.5-2 and Figure 2.8.5.6.3.5-1. For large breaks and large small breaks, the effect of containment or RCS pressure above atmospheric pressure is not credited and the boric acid solubility limit at 218°F (boiling point of saturated boric acid solution at atmospheric conditions) is assumed. For breaks where RCS depressurization is not complete or breaks where the RCS might be at elevated pressures at hot leg switchover time, the solubility limit associated with the saturation temperature of water at the associated elevated pressure is credited.

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- The liquid mixing volume used in the calculation includes 50 percent of the lower plenum volume and is consistent with the results of the Mitsubishi Heavy Industries (MHI) BACCHUS PWR vessel mixing tests summarized in Reference 7.
 - For SBLOCA scenarios, the analysis does not assume a specific start time for cooldown/depressurization emergency procedures, nor does it assume depressurization to some minimum pressure at hot leg switchover time. Nevertheless, for the purpose of defining expected scenarios, it is anticipated that operators begin cooldown/depressurization within one hour of the initiation of the event.
 - The effect of containment sump pH additives on increasing the boric acid solubility limit is not credited.
 - The boric acid concentration of the makeup containment sump water during recirculation is a calculated sump mixed mean boron concentration. The calculation of the sump mixed mean boron concentration assumes maximum mass and maximum boron concentrations for significant boron sources, and minimum mass and maximum boron concentrations for significant dilution sources.
 - ECCS flow and enthalpy changes that may occur during the switchover from injection mode to sump recirculation are not part of the long-term cooling analysis scope and were considered in the SBLOCA analysis.

In addition to the above assumptions, NRC requirements pertaining to the decay heat generation rate for both boric acid accumulation and decay heat removal (which is based on the 1971 ANS Standard for an infinite operating time with 20-percent uncertainty) is utilized as an input to prepare the boric acid precipitation calculation. The assumed core power includes a multiplier to address instrument uncertainty as identified by Section 1.A of 10 CFR 50, Appendix K.

The acceptance criteria for the long-term cooling analysis are demonstrated by the ability to keep the core cool after a LOCA and calculating a hot leg switchover (HLSO) time with methods, plant design assumptions, and operating parameters are consistent with the interim methodology reported in Reference 4. The FSAR, Technical Specifications, and Emergency Operating Procedures (EOPs) have been revised to support the maximum time to establish simultaneous hot leg and cold leg injection.

ECCS recirculation flows are evaluated by comparing minimum safety injection pump flows to the flows necessary to dilute the core and replace core boiloff, thus keeping the core quenched.

Description of Analyses and Evaluations

There are two aspects to a long-term cooling analysis – the potential for boric acid precipitation to occur and decay heat removal. The purpose of the boric acid precipitation analysis is to demonstrate that the maximum boric acid concentration in the core remains below the solubility limit, thereby preventing the precipitation of boric acid in the core. If boric acid were to

precipitate in the core region, the precipitate might prevent water from remaining in contact with the fuel cladding and, consequently, result in the core temperature not being maintained at an acceptably low value. The boric acid precipitation analysis determines the appropriate time for switching some or all ECCS recirculation flow to the hot leg and verifies that there is sufficient dilution flow through the core to dilute the core and prevent boric acid buildup.

Prior to sump recirculation, core cooling is addressed by the LBLOCA analysis that demonstrates core reflood and stable and sustained quench, and by the SBLOCA analysis that demonstrates core recovery. After an SBLOCA, RCS refill, depressurization and entry into shutdown cooling, or depressurization and indefinite sump recirculation occurs. With the switch to sump recirculation, long-term cooling is addressed by demonstrating that the core remains covered with two-phase mixture in the long term, thereby ensuring that the core temperature is maintained at an acceptably low value. Paragraph (b)(5) of 10 CFR 50.46 is satisfied when the fuel in the core is quenched, the switch from injection to recirculation phases is complete, and the recirculation flow is large enough to match the boiloff rate. Prior to hot leg recirculation, the ECCS recirculation flow must be sufficient to remove decay heat. ECCS pump availability and specific flow path alignments may reduce ECCS recirculation flow as compared to the flows available during the injection phase. After the switch to hot leg recirculation, core flow sufficient to dilute the core or prevent boric acid buildup, by definition, exceeds core boiloff and therefore provides core cooling.

The long-term cooling analysis described here supports the Post-LOCA Boric Acid Precipitation Analysis Logic presented in Table 2.8.5.6.3.5-3. The flowchart in Figure 2.8.5.6.3.5-2 shows the applicability of the calculations to the specific post-LOCA scenarios.

Large-Break LOCAs

Large breaks (double-ended guillotine down to approximately 1.0 ft²) rapidly depressurize to very near containment pressure with no operator action. The 14.7 psia boric acid precipitation calculation models this scenario and calculates the boric acid buildup for the limiting condition of a cold leg break. Dilution and core cooling flows are confirmed for 14.7 psia RCS backpressure. After hot leg switchover, the hot leg injected flow provides immediate core dilution for a cold leg break. If the break is in the hot leg, injected ECCS flow to the cold leg is sufficient to prevent the buildup of boric acid in the core after switchover to hot leg recirculation. Therefore, after hot leg switchover, simultaneous hot leg and cold leg injection prevents boric acid precipitation in the long term.

Large breaks that lead to rapid RWST draindown represent the limiting case for recirculation flow requirements. At the start of sump recirculation, ECCS flows are evaluated.

Large Small-Break LOCAs

Large small breaks (approximately 0.1 - 1.0 ft²) depressurize to relatively low pressures (before the potential for boric acid precipitation) with no operator action. The 120 psia boric acid precipitation calculation models this scenario and calculates the boric acid buildup for the limiting condition of a cold leg break. The 120 psia calculations consider less core voiding, a

lower h_{fg} (heat of vaporization), and do not credit SI subcooling to reduce core boiloff. After hot leg switchover, as with large breaks, the hot leg injected flow provides core dilution for cold leg breaks and cold leg injected flow prevents buildup of boric acid in the core for hot leg breaks. Dilution and decay heat removal flows are confirmed as adequate at 120 psia RCS backpressure. Core dilution flow provides effective core cooling.

Small-Break LOCAs

For small breaks (approximately 0.005 - 0.1 ft²), emergency procedures instruct operators to take action to depressurize and cool down the RCS. Although this depressurization and cooldown process typically begins within one hour after the event, the long-term cooling analysis makes no specific assumptions regarding time to depressurize. Depressurization to 120 psia (the threshold for boric acid precipitation concerns) may occur before or after hot leg switchover time. In either case, the boric acid buildup at hot leg switchover time is conservatively represented by that calculated for the 120 psia RCS backpressure scenario since this calculation takes no credit for SI subcooling, nor any beneficial effects of the operator action (such as reduced net core boiloff due to condensation in the steam generators). If 120 psia is reached before hot leg switchover time, the core dilution flow after hot leg switchover, which is confirmed as adequate for 120 psia backpressure, provides effective core dilution. If at hot leg switchover time, the 120 psia has not been reached, boric acid precipitation does not occur so long as the RCS remains above this pressure since water and boric acid are miscible at the saturation temperature for these pressures. Even if the RCS pressure is above 120 psia at 24 hours after the LOCA with no core dilution flow, the total boric acid in the core is well below the saturation capacity at the corresponding saturation temperature. Furthermore, if after 24 hours with no dilution flow, the RCS is at saturation and depressurized at the maximum cooldown rate, the core is diluted prior to reaching the boric acid precipitation point. If subcooled core conditions are reached either before or after hot leg switchover, boric acid precipitation is not a concern since there is no net boiling in the core. If subcooled core entry conditions are not reached, the operators continue to depressurize the RCS under controlled conditions. Sump recirculation continues, decay heat in the core decreases, and core dilution flow prevents the buildup of boric acid. Eventually, subcooled core conditions are reached, the system is put into RHR or it remains in indefinite recirculation cooling.

Very Small-Break LOCAs

For very small breaks (less than approximately 0.005 ft²), emergency procedures instruct operators to take action to depressurize the RCS. Because the break is small, subcooled conditions are reached prior to depressurization to 120 psia (the threshold for boric acid precipitation concerns). Natural circulation, if lost, is quickly restored. While in natural circulation, boric acid precipitation is not a concern because the core region is not stagnant. When subcooled conditions occur, net core boiling ceases and boric acid does not accumulate. Eventually, subcooled core conditions will be reached, the system will be put into RHR or continued natural circulation and sump recirculation will keep the boric acid from accumulating in the core. It is important to note that CPNPP is designed so that high pressure SI provides hot leg recirculation flow. As such, it is not necessary to depressurize the RCS to get effective dilution flow.

Results

To address LBLOCAs, CPNPP SPU post-LOCA boric acid buildup calculations for 14.7 psia were performed. These calculations support a 3-hour switchover time to initiate simultaneous hot leg and cold leg SI injection. Note the boric acid concentration is below the boric acid solubility limit for this scenario up to a 7-hour switchover time. Figure 3 shows the buildup of boric acid versus time and the boric acid solubility limit used for this scenario. Although the boric acid buildup calculations for this scenario apply to RCS pressures of up to 30 psia, the boric acid solubility above the atmospheric boiling point of a saturated boric acid and water solution is not credited. Figure 3 also shows the dilution effect of the hot leg injected flow after simultaneous hot leg and cold leg is established.

To address SBLOCAs, CPNPP SPU post-LOCA boric acid precipitation calculations for 120 psia were performed. These calculations show that there is considerable margin to the boric acid solubility limit for this scenario at the 3-hour switchover time. The 120 psia calculations consider less core voiding, a lower heat of vaporization (h_{fg}), and do not credit SI subcooling to reduce core boiloff. Since the boric acid buildup calculations for this scenario apply to RCS pressures of 30 to 120 psia, the boric acid solubility for the saturation temperature of water at 30 psia was credited. Figure 2.8.5.6.3.5-4 shows the buildup of boric acid versus time and the solubility limit appropriate for this scenario. Figure 2.8.5.6.3.5-4 also shows the dilution effect of the hot leg injected flow after simultaneous hot leg and cold leg is established.

In the unlikely event that the RCS pressure remains above 120 psia at hot leg switchover time while at saturated conditions, boric acid precipitation does not occur since the total boric acid in the core is well below the saturation capacity at the elevated pressure saturation temperature. In order to demonstrate the effectiveness of hot leg dilution flow for this scenario, calculations were performed for a hypothetical condition where there would be no hot leg dilution flow for 24 hours. Figure 2.8.5.6.3.5-5 shows the boric acid concentration in the core with the RCS at 120 psia for 24 hours assuming no steam generator heat removal, no dilution flow, and no benefit of reduced steaming due to SI subcooling. At 24 hours, the boric acid concentration is still below the boric acid solubility limit at the saturation temperature at 120 psia. Figure 2.8.5.6.3.5-5 also shows that if hot leg flow is established at 24 hours and the RCS is at saturation and is then cooled (with corresponding depressurization) at a cooldown rate of 100°F/hr, boric acid precipitation does not occur. The resulting hot leg dilution flow maintains the boric acid concentration in the core well below the solubility limit, even as the solubility limit is reduced due to the RCS cooldown. For CPNPP, hot leg dilution flow is provided by the SI pumps which would, in fact, provide dilution flow at RCS pressures well above 120 psia.

Calculations were performed to support an early switchover to hot leg or simultaneous injection. Two aspects of early switchover were considered – the hot leg entrainment threshold and core cooling. If switchover occurs too early, injected SI in the hot legs might be carried around the loops and might not be available for core cooling and dilution. Entrainment threshold calculations similar to those reported in Reference 5 demonstrated that significant hot leg entrainment would not occur after 75 minutes. Calculations showed that either hot leg or cold leg flows are sufficient to provide core cooling flow at 3 hours after the LOCA.

Assessments were made of the effect of loop pressure drop and downcomer boiling on the core mixing volume by performing calculations similar to those reported to the NRC in References 5 and 6. For CPNPP, the total loop pressure drop loss coefficient with and without locked RCP rotor is approximately $1.3\text{E-}08$ ft/gpm² and $7.1\text{E-}08$ ft/gpm², respectively. In all cases, the core region mixing volume assumed in the boric acid buildup calculation was found to be conservatively small in relation to the collapsed liquid volume that would be based on loop pressure drop and available downcomer head.

The effect of the refilling of the pump suction leg loop seals (due to a break at the top of the cold leg pipe) was also assessed by performing calculations similar to those reported to the NRC in References 5 and 6. For CPNPP, the bottom elevation of the loop seal piping is approximately 6.34 feet below the top of the active fuel. While the simultaneous complete closure of all four loop seals would depress the core mixture to slightly below that associated with the core mixing volume, the expected duration of the depression would be brief. Brief core mixture level depressions would have the benefit of promoting mixing between the core region and lower plenum by cycling liquid back and forth between the core region, lower plenum, and downcomer.

An assessment was made of the effect of boric acid plate-out in the steam generators by performing calculations similar to those reported to the NRC in Reference 6. These calculations show that, with 10-percent entrainment for 1.5 hours, the total boric acid mass entrained would deposit a coating of approximately 0.002 inches over 10 feet of steam generator tubes. This coating would not significantly increase loop resistance or depress the core mixture level.

An assessment was made concerning the potential for boric acid precipitation at the hot leg injection point or at colder regions of the vessel. A simplified demonstration calculation showed that the mixing of injected SI with the highly borated solution in the reactor vessel would not initiate boric acid precipitation at the injection point. This calculation ignored temperature and boric acid gradients and assumed effective mixing with no differentiation between different mixing mechanisms such as diffusion (thermal or molecular) and density-driven convection within the vessel. The assessment also concluded that the heating of the injected water as it travels to the core region (either from the downcomer or hot leg) and the expected density-driven mixing mechanisms in the vessel would make it unlikely that significant temperature or boric acid gradients would exist. These conclusions were consistent with those reported to the NRC in Reference 6.

2.8.5.6.3.5.3 Conclusions

In summary, the CPNPP SPU post-LOCA boric acid precipitation calculations used conservative methodology to initiate switchover to hot leg recirculation. SI flow to the hot leg provides effective core dilution thus precluding boric acid precipitation in the core. This realignment addresses the requirements of 10 CFR 50.46 (b) (4) coolable geometry and 10 CFR 50.46 (b) (5) long-term cooling. ECCS flows during sump recirculation were shown to be sufficient to remove decay heat after a LOCA for SPU plant conditions, provided the ECCS realignment to provide SI flow to the hot legs occurs no sooner than 3 hours following the event. This addresses the requirements of 10 CFR 50.46 (b) (5) long-term cooling. Since the long-term

core cooling analyses for the SPU show that no changes to the CPNPP ECCS are required, General Design Criterion (GDC)-35 requirements continue to be met.

2.8.5.6.3.5.4 References

1. H. C. Yeh, "Modification of Void Fraction Calculation," Proceedings of the Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Volume 1, Taipei, Taiwan, June 6, 1988.
2. P. Cohen, Water Coolant Technology of Power Reactors, Chapter 6, "Chemical Shim Control and pH Effect," ANS-USEC Monograph, 1980 (Originally published in 1969).
3. Letter dated August 1, 2005 from R. A. Gramm, U. S. Nuclear Regulatory Commission to J. A. Gresham, Westinghouse Electric Company, "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, 'Post LOCA Long Term Cooling Model' Due to Discovery of Non-conservative Modeling Assumptions During Calculations Audit".
4. Letter dated October 3, 2006 from Sean E. Peters, Project Manager, Special Projects Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, NRC to Stacey L. Rosenberg, Chief, Special Projects Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, NRC, "Summary of August 23, 2006 Meeting With The Pressurized Water Reactor Owners Group (PWROG) to Discuss the Status of Program to Establish Consistent Criteria For Post Loss-Of-Coolant (LOCA) Calculations."
5. Letter L-05-112, FirstEnergy Nuclear Operating Company to USNRC, "Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173," July 08, 2005.
6. Letter L-05-169, FirstEnergy Nuclear Operating Company to USNRC, "Responses to a Request for Additional Information (RAI dated September 30, 2005) in Support of License Amendment Request Nos. 302 and 173," November 21, 2005.
7. WCAP-16317, "Review and Evaluation of MHI BACCHUS PWR Vessel Mixing Tests," November 2004.

Table 2.8.5.6.3.5-1

CPNPP Post-LOCA Long-Term Cooling Analysis Input Parameters

Parameter	SPU Value
Analyzed Core Power (MWt)	3,612
Analyzed Core Power Uncertainty (percent)	0.6
Decay Heat Standard	1971 ANS, Infinite Operation, plus 20% (10 CFR 50 Appendix K)
H ₃ BO ₃ Solubility Limit (weight percent)	See Table 2.8.5.6.3.5-2
RWST Boron Concentration, Maximum (ppm)	2,600
RWST Volume, Maximum (gallons)	506,368
RWST Temperature, Minimum (°F)	40
Accumulator Boron Concentration, Maximum (ppm)	2,600
Accumulator Tank Volume, Maximum (gallons)	6,597 per tank
Accumulator Tank Temperature, Minimum (°F)	60

**Table 2.8.5.6.3.5-2
Boric Acid Solution Solubility Limit**

Temperature, °F	Pressure, psia	Solubility g H ₃ BO ₃ /100 g of Solution in H ₂ O
P = Atmospheric Pressure		
32	14.7	2.70
41	14.7	3.14
50	14.7	3.51
59	14.7	4.17
68	14.7	4.65
77	14.7	5.43
86	14.7	6.34
95	14.7	7.19
104	14.7	8.17
113	14.7	9.32
122	14.7	10.23
131	14.7	11.54
140	14.7	12.97
149	14.7	14.42
158	14.7	15.75
167	14.7	17.41
176	14.7	19.06
185	14.7	21.01
194	14.7	23.27
203	14.7	25.22
212	14.7	27.53
217.9	14.7	29.27
P = P_{SAT}		
226.0	19.3	31.47
242.8	26.3	36.69
260.1	35.5	42.34
277.3	47.1	48.81
289.9	57.5	54.79
304.7	71.9	62.22
318.9	88.3	70.67
339.8 = Congruent Melting of H₃BO₃		

**Table 2.8.5.6.3.5-3
Post-LOCA Boric Acid Precipitation Analysis Logic**

APPROXIMATE BREAK SIZE (FT ²)	SCENARIO	ANALYSIS
DEG	<u>Large Breaks</u> Large breaks will rapidly depressurize to very near containment pressure.	Represented by 14.7 psia boric acid buildup calculation. Dilution flows confirmed for 14.7 psia RCS backpressure.
1.0		
0.1	<u>Large Small Breaks</u> Large small breaks will depressurize to below 120 psia without operator action.	Represented by 120 psia boric acid buildup calculation. Dilution flows are confirmed at 120 psia RCS backpressure.
	0.1	
0.005	<u>Small Breaks</u> Emergency procedures will instruct operators to take action to depressurize RCS. Eventually the system will be put into RHR or it will remain in indefinite recirculation cooling.	Credit operator action to depressurize the RCS. If the 120 psia is reached before HLSO time, the 120 psia boric acid buildup calculation applies. If 120 psia is not reached before HLSO time, credit higher boric acid solubility limit. If core subcooling conditions are reached, boric acid precipitation is not a concern since there will be no net boiling in the core.
	0.005	
0.001	<u>Very Small Breaks</u> Emergency procedures will instruct operators to take action to depressurize RCS. Subcooled conditions will be reached prior to depressurization to 120 psia (the threshold for boric acid precipitation concerns). Eventually, the system will be put in RHR or it will remain in indefinite recirculation cooling.	Natural circulation, if lost, will be quickly restored. While in natural circulation, boric acid precipitation is not a concern because the core region will not be stagnant.
	0.001	
0.0	<u>Leaks</u> Charging System has make-up capacity.	

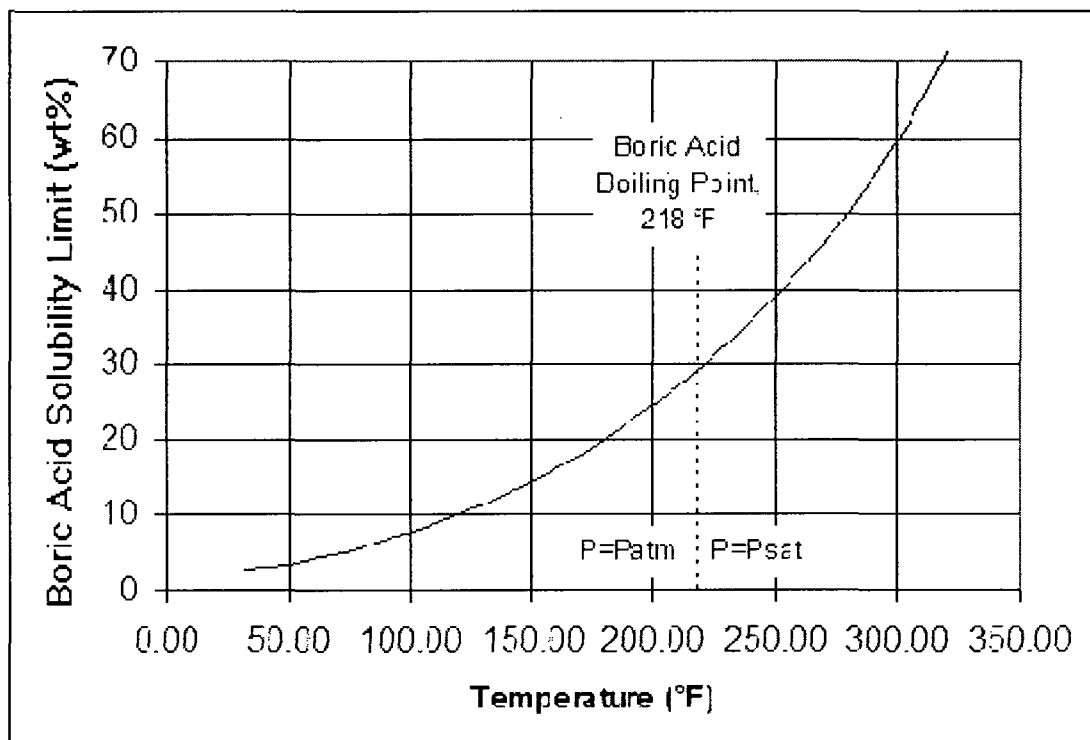


Figure 2.8.5.6.3.5-1 Boric Acid Solubility Limit

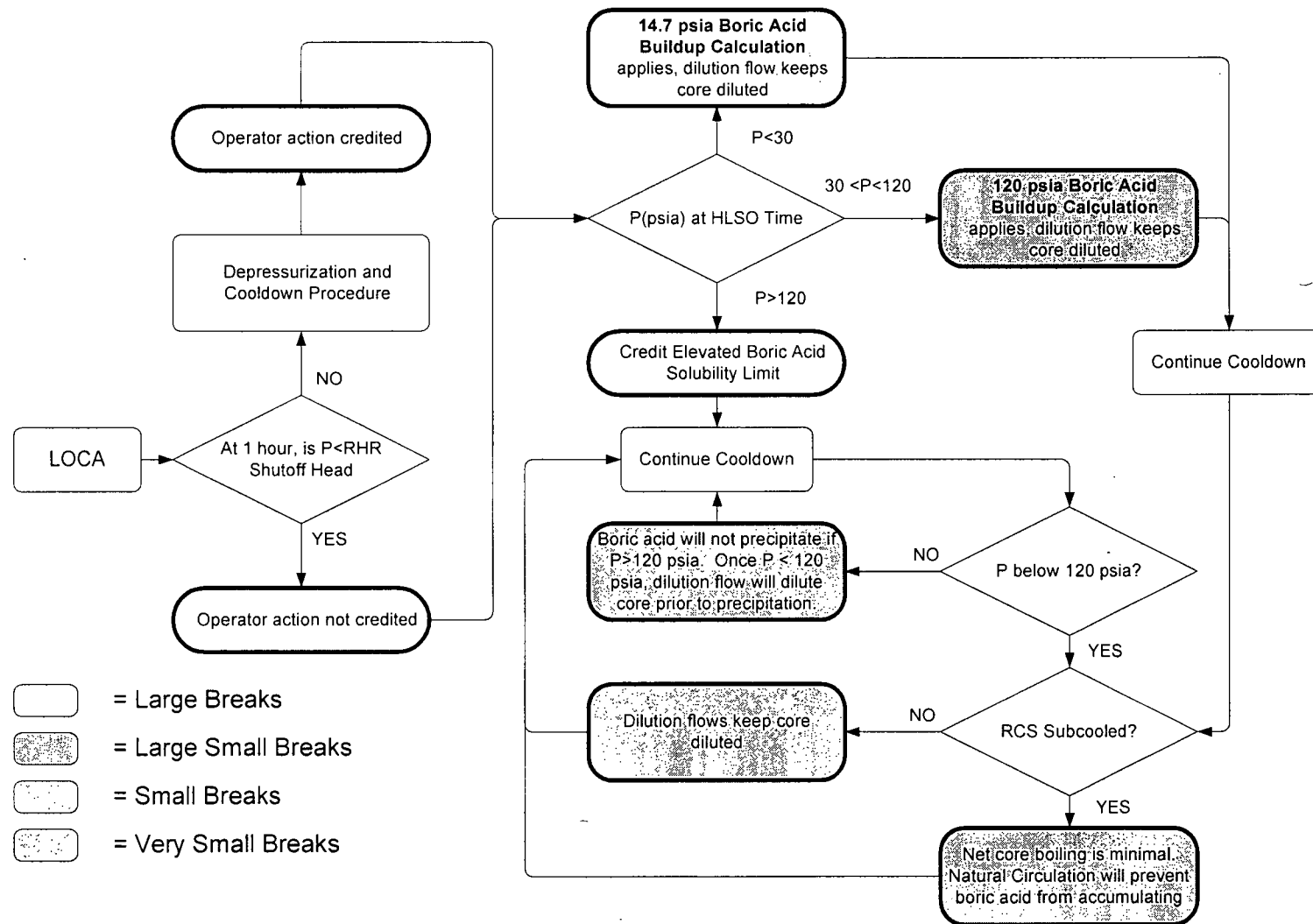
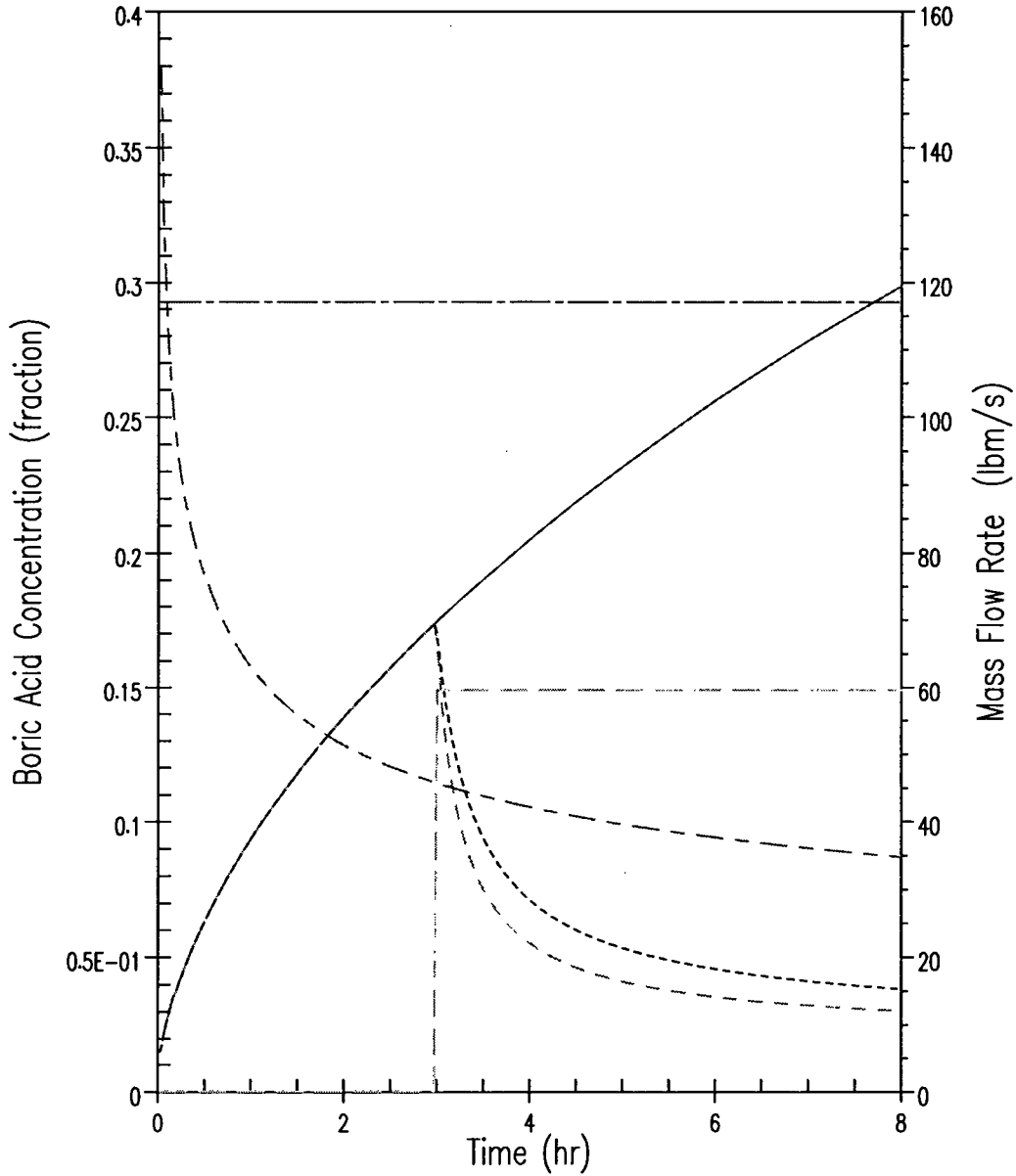


Figure 2.8.5.6.3.5-2 Post-LOCA Boric Acid Precipitation Analysis Logic

COMANCHE PEAK UNITS 1 & 2 UPRATE - 14.7 PSIA

Boric Acid Concentration (fraction)
 - - - - - NO HL DILUTION FLOW
 - - - - - WITH HL DILUTION FLOW
 - - - - - BOILOFF + 10% HL DILUTION FLOW
 - - - - - BORIC ACID SOL LIMIT @ T_{sat}=218 degF (14.7 PSIA)
 Mass Flow Rate (lbm/s)
 - - - - - CORE BOILOFF
 - - - - - HL SI FLOW

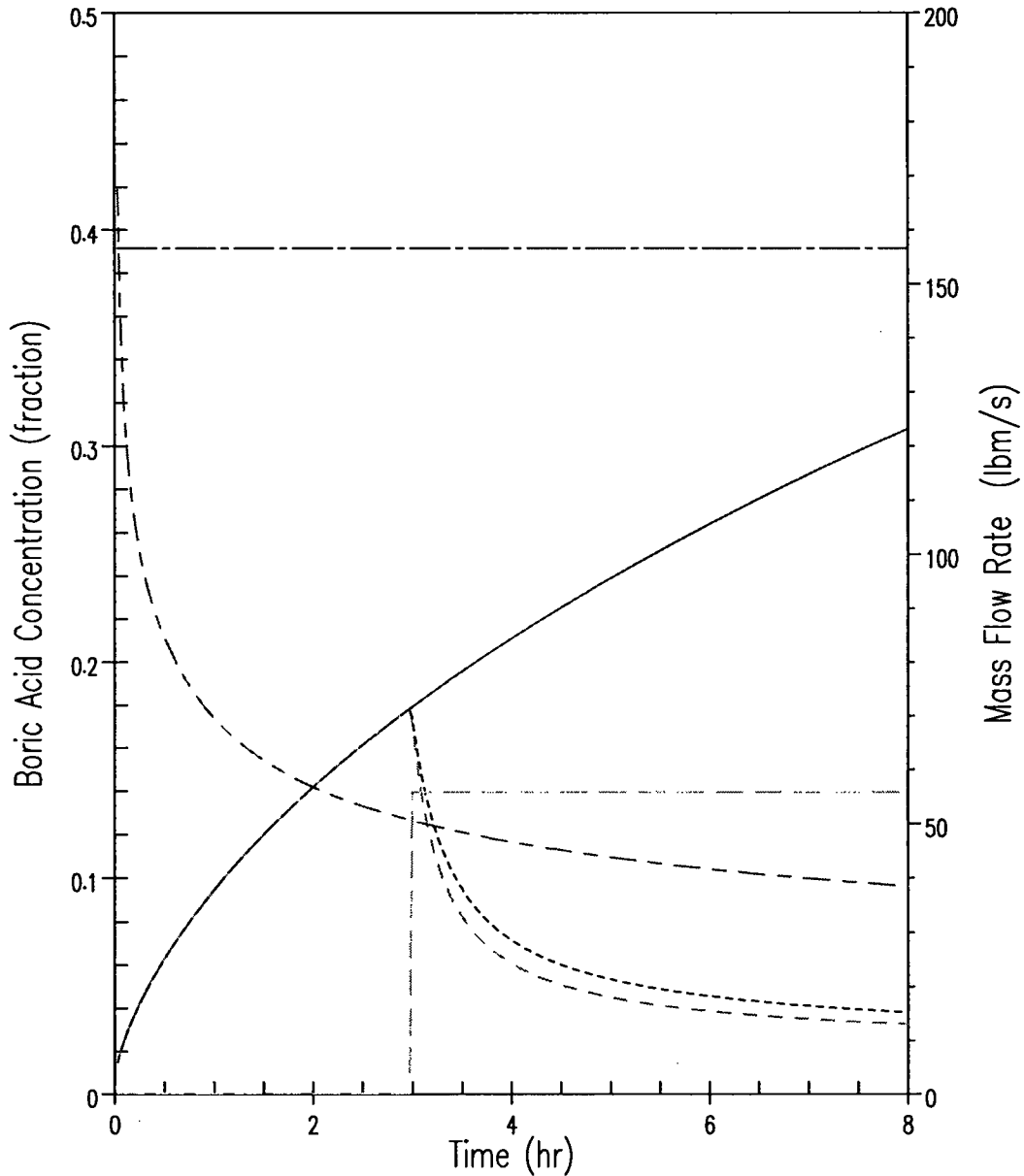


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Figure 2.8.5.6.3.5-3 Boiloff, SI, and Core Dilution Rate at a 3-Hour HLSO Time at 14.7 psia

COMANCHE PEAK UNITS 1 & 2 UPRATE - 120 PSIA

Boric Acid Concentration (fraction)
 - - - - NO HL DILUTION FLOW
 - - - - WITH HL DILUTION FLOW
 - - - - BOILOFF + 10% HL DILUTION FLOW
 - - - - BORIC ACID SOL LIMIT @ $T_{sat}=250.3$ degF (30 PSIA)
 Mass Flow Rate (lbm/s)
 - - - - CORE BOILOFF
 - - - - HL SI FLOW

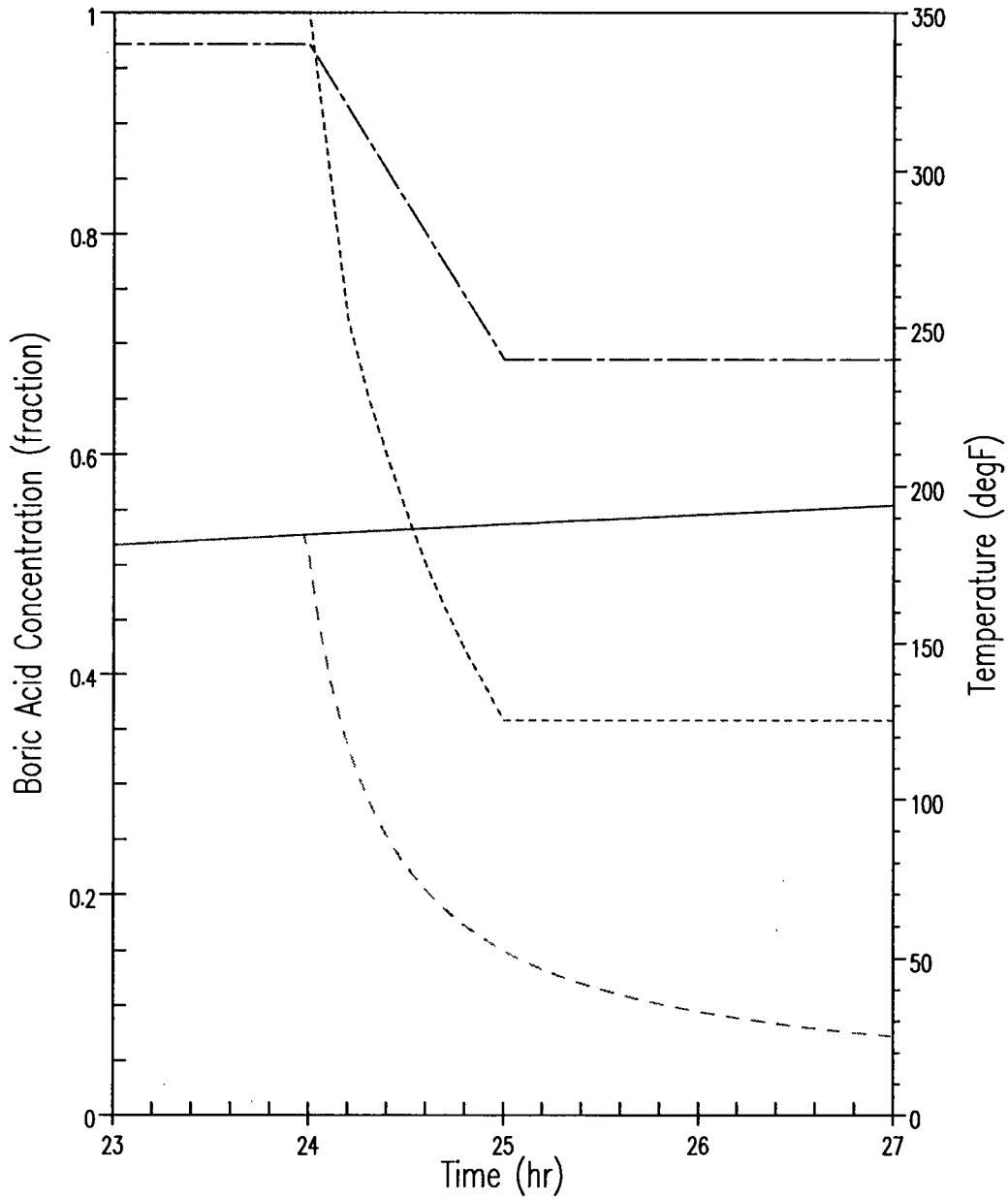


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Figure 2.8.5.6.3.5-4 Boiloff, SI, and Core Dilution Rate at a 3-Hour HLSO Time at 120 psia

COMANCHE PEAK UNITS 1 & 2 - 120 PSIA

Boric Acid Concentration (fraction)
--- NO HL DILUTION FLOW
- - - WITH HL DILUTION FLOW
- - - - BORIC ACID SOL LIMIT W/ 100F/HR COOLDOWN
Temperature (degF)
- - - - TEMP W/ 100F/HR COOLDOWN



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Figure 2.8.5.6.3.5-5 Demonstration of Core Dilution at 24 hours

2.8.5.6.3.6 LOCA Forces

2.8.5.6.3.6.1 Regulatory Evaluation

- GDC-4, insofar as it allows exclusion of dynamic effects of postulated pipe ruptures from the design basis.

The LOCA hydraulic forces analysis generates the hydraulic forcing functions that act on RCS components as a result of the postulated LOCA. The most recent qualification of the vessel internals and fuel was performed using an advanced beam model version of MULTIFLEX 3.0 (Reference 1), in accordance with methodology approved by the NRC in WCAP-15029 (Reference 2). This same version of the MULTIFLEX code was used in the hydraulic forces analysis for the CPNPP Units 1 and 2 SPU Program.

The leak-before-break methodology demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated pipe ruptures in the primary coolant loop piping and 10-inch and larger reactor coolant loop branch lines, as discussed in FSAR Section 3.6B.2.5. Implementation of this technology eliminates the need for pipe whip restraints and jet impingement barriers, respectively. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by this modification.

FSAR Section 3.9N.1.4.2 describes the modeling and analytical methods for evaluating the structural stress analysis for the reactor coolant loop and supports for faulted loading conditions. This section identifies that the faulted loading condition of the RCS loop and supports considers loading due to: 1) internal pressure; 2) weight; 3) safe shutdown earthquake; 4) LOCA (pipe break); and 5) transients. For LOCAs, mechanical loads are developed in the broken and unbroken reactor coolant loops and in the reactor vessel as a result of transient flow and pressure fluctuations following a postulated pipe break in one of the reactor coolant loops. Time-history dynamic analysis is performed for a number of postulated break cases. Hydraulic models are used to generate time-dependent hydraulic forcing functions used in the analysis of the RCS for each break case. The transient applied forces are described in FSAR Section 3.6B.2. Also, FSAR Section 3.9N.1.4.3 provides additional information.

FSAR Section 3.9N5.2 identifies the loading conditions for normal, upset, emergency, and faulted conditions that form the basis for the design of the reactor internals. The design bases for the mechanical design of the reactor vessel internals relevant to LOCAs are: 1) the core internals are designed to withstand mechanical loads arising from operating basis earthquake, safe shutdown earthquake, and pipe ruptures and meet the requirements of Item 2; 2) the reactor shall have mechanical provisions that are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions; and 3) following the design basis accident, the plant shall be capable of being shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in FSAR Table 3.9N-13. Details of the dynamic analyses, input forcing functions, and response loadings are presented in FSAR Section 3.9N.2.

2.8.5.6.3.6.2 Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

To conservatively calculate LOCA hydraulic forces for CPNPP Units 1 and 2, the following operating conditions were considered in establishing the limiting temperature and pressures:

- Initial RCS conditions associated with a minimum thermal design flow of 95,700 gpm per loop
- Uprated core power of 3,612 MWt (analyzed nuclear steam supply system (NSSS) power of 3,628 MWt)
- A nominal RCS hot full power (HFP) vessel T_{avg} range of 574.2° to 589.2°F (This provides an RCS T_{cold} range of 542.2° to 558.0°F.)
- An RCS temperature uncertainty of $\pm 6.0^\circ\text{F}$
- A feedwater temperature range of 390.0° to 450.3°F
- A nominal RCS pressure of 2,250 psia
- A pressurizer pressure uncertainty of ± 30 psi

Based on these conditions, the LOCA hydraulic forces on the vessel and steam generator were generated at a minimum T_{cold} of 536.2°F, including uncertainty, and a pressurizer pressure of 2,280 psia, including uncertainty. In order to accommodate the NSSS RCS loop piping structural analyses for the CPNPP Units 1 and 2 SPU Program, the LOCA hydraulic forces on the RCS piping were generated at the minimum T_{cold} of 536.2°F; along with two higher T_{cold} values of 546.7° and 552.0°F, including uncertainty; and a pressurizer pressure of 2,280 psia, including uncertainty.

The LOCA hydraulic forcing functions (HFF) and loads that occur as a result of a postulated LOCA are calculated assuming a limiting break location and break area. The NRC's revision to GDC-4 allowed the main coolant piping breaks to be "excluded from the design basis when analyses reviewed and approved by the commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for piping." This exemption is generally referred to as "leak before break." The analysis presented in WCAP-10527 (Reference 3) is technical justification for eliminating primary loop pipe ruptures from the design basis for CPNPP Units 1 and 2. The analyses presented in References 4 through 9 provide technical justification for eliminating accumulator, pressurizer surge, and residual heat removal branch line ruptures from the design basis for CPNPP Units 1 and 2.

Thus, the primary loop piping breaks and larger branch lines did not need to be considered when generating CPNPP Unit 1 and 2 LOCA hydraulic forces. The breaks that were considered were the 6-inch safety injection (SI) line connection to the cold leg and the 4-inch pressurizer spray line connection on the hot leg.

Description of Analyses and Evaluations

LOCA hydraulic forces were generated with a focus on the component of interest (such as loop, vessel, or steam generator) using the advanced beam model version of MULTIFLEX 3.0 (Reference 1), and assuming a conservative break opening time of 1 millisecond.

Generally, this improved modeling results in lower, more realistic, but still conservative hydraulic forces on the core barrel.

The MULTIFLEX computer code calculated the thermal-hydraulic transient within the RCS and considered subcooled, transition, and early two-phase (saturated) blowdown regimes. The code used the method of characteristics to solve the conservation laws, assuming one-dimensional (1-D) flow and a homogeneous liquid-vapor mixture. The RCS was divided into subregions, in which each subregion was regarded as an equivalent pipe. A complex network of these equivalent pipes was used to represent the entire primary RCS.

For the reactor pressure vessel (RPV) and specific vessel internal components, the MULTIFLEX code generated the LOCA thermal-hydraulic transient that was input to the LATFORC and FORCE2 post-processing codes (Reference 10). These codes, in turn, were used to calculate the actual forces on the various components.

These forcing functions for horizontal and vertical LOCA hydraulic forces, combined with seismic, thermal, and flow-induced vibration loads as applicable, were used in the structural analyses to determine the resultant mechanical loads on the vessel and vessel internal components. The vessel forces results are provided for use in the analyses described in Section 2.2.3, Reactor Pressure Vessel Internals and Core Supports – Mechanical System Evaluations.

The loop forces analysis used the THRUST post-processing code to generate X, Y, and Z directional component forces during a LOCA blowdown. RCS pressure, density, and mass flux were calculated by the MULTIFLEX code and were used as inputs to the THRUST code. The THRUST code is described in WCAP-8252 (Reference 11). The loop forces results are provided for use in the analyses described in LR subsection 2.2.2.1, NSSS Piping, Components, and Supports; and in LR subsection 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects.

The steam generator forces analysis utilizes the hydraulic transient time-history data, which is extracted directly from the MULTIFLEX computer code output. This analysis is performed to qualify the steam generators for duty using loads associated with the uprated power conditions.

Results

All LOCA hydraulic forces analyses for the CPNPP Units 1 and 2 SPU Program were performed directly at the analyzed NSSS power level of 3,628 MWt, using models specific to the CPNPP Units 1 and 2 NSSS design. The analyses of the forces acting on the RPV and vessel internals, fuel, loop piping, and steam generator were performed. The results of the LOCA hydraulic forces analyses were then used as input to the calculations for component qualification.

Discussion of Margin Change

As previously mentioned, the LOCA forces are used as input to the various structural analyses, so margin quantification would be appropriately derived from the calculations for the specific component. Qualitatively speaking, margin in the forces analyses is realized by analyzing smaller diameter lines, because larger diameter lines would yield higher forces.

2.8.5.6.3.6.3 Conclusions

Luminant Power has reviewed the analyses of the LOCA events and the ECCS, and has concluded that the analyses have adequately accounted for plant operation at the proposed power level and that the analyses were performed using acceptable analytical models. Luminant Power further concluded that the evaluation has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, Luminant Power concluded that the plant will continue to meet the CPNPP current licensing basis requirements with respect to GDC-4, -27, -35, and 10 CFR 50.46 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the LOCA.

2.8.5.6.3.6.4 References

1. WCAP-9735, Rev. 2, and WCAP-9736, Rev. 1, "MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic- Structural System Dynamics Advanced Beam Model," February 1998.
2. WCAP-15029 and WCAP-15030, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," January 1999.
3. WCAP-10527 and WCAP-10528, "Technical Basis for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Comanche Peak Units 1 and 2," April 1984.
4. WCAP-12267 and WCAP-12268, "Technical Bases for Eliminating Rupture of the Accumulator Injection Nozzles as a Structural Design Basis for Comanche Peak Unit 1," May 1989.

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5. WCAP-12247 and WCAP-12248, "Evaluation of Thermal Stratification for the Comanche Peak Unit 1 Pressurizer Surge Line," April 1989.
 6. WCAP-12258 and WCAP-12259, "Evaluation of Thermal Stratification for the Comanche Peak Unit 1 Residual Heat Removal Lines," April 1989.
 7. WCAP-13167 and WCAP-13168, "Technical Justification for Eliminating 10 Inch Accumulator Lines Rupture as the Structural Design Basis for the Comanche Peak Nuclear Plant Unit 2," January 1992.
 8. WCAP-13100 and WCAP-13101, "Technical Justification for Eliminating Pressurizer Surge Line Rupture from the Structural Design Basis for Comanche Peak Unit 2," December 1991.
 9. WCAP-13165 and WCAP-13166, "Technical Justification for Eliminating Residual Heat Removal Lines Rupture as the Structural Design Basis for Comanche Peak Nuclear Power Plant – Unit 2," December 1991.
 10. WCAP-8708 and WCAP-8709, "MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," September 1977.
 11. WCAP-8252, Rev. 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," May 1977.

2.8.5.7 Anticipated Transients Without Scram

2.8.5.7.1 Regulatory Evaluation

An anticipated transient without scram (ATWS) is defined as an anticipated operational occurrence followed by the failure of the reactor trip portion of the protection system. Nuclear Regulatory Commission (NRC) regulation 10 CFR 50.62 requires that:

- Each pressurized water reactor (PWR) must have equipment that is diverse from the reactor trip system to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must perform its function in a reliable manner and be independent from the existing reactor trip system.
- Each PWR manufactured by Combustion Engineering or Babcock and Wilcox must have a diverse scram system (DSS). This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

The review was conducted to ensure that:

- The above requirements were met.
- The setpoints for the ATWS mitigating system actuation circuitry (AMSAC) remain valid for the SPU Program.

Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 are Westinghouse plants and are not required to have a DSS. In addition, for plants where a DSS is not specifically required by 10 CFR 50.62, verification is required to ensure that the consequences of an ATWS are acceptable. Luminant Power has satisfied the applicable requirements by having an AMSAC system installed at CPNPP Units 1 and 2, and by verifying that the consequences of an ATWS are acceptable for CPNPP Units 1 and 2 with the installed AMSAC system. The review included:

- The limiting event determination
- The analytical model and its applicability
- The values of parameters used in the analytical model
- The analyses results

The justification of the applicability of the analyses to CPNPP Units 1 and 2 and the operating conditions for the SPU Program were reviewed.

Current Licensing Basis

The final ATWS rule (10 CFR 50.62(c)(1)(Reference 1)) requires the incorporation of a system to provide diverse (from the reactor trip system) actuation of the auxiliary feedwater (AFW) system and turbine trip for Westinghouse-designed plants. The installation of the NRC-approved AMSAC system, described in Final Safety Analysis Report (FSAR) Section 7.8.1.1, satisfies the final ATWS Rule. The bases for this rule and the AMSAC design are supported by Westinghouse analyses documented in Westinghouse letter NS-TMA-2182 (Reference 2). For consistency with the basis of the final ATWS Rule and the supporting analyses documented in NS-TMA-2182, the peak RCS pressure should not exceed the limiting pressure at which the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code Service Level C stress limit (3,200 psig) is exceeded. This value corresponds to the maximum allowable pressure for the weakest component in the reactor pressure vessel (the nozzle safe end). The analysis of the ATWS event is described in FSAR Section 15.8.

2.8.5.7.2 Technical Evaluation

2.8.5.7.2.1 Introduction

As noted above, the final ATWS Rule, 10 CFR 50.62(c)(1) (Reference 1), requires the incorporation of a diverse (from the reactor trip system) actuation of the AFW system and turbine trip for Westinghouse-designed plants. The installation of the NRC-approved AMSAC satisfies this final ATWS Rule. However, it must also be demonstrated that the deterministic ATWS analyses that form the basis for this rule and the AMSAC design remain valid for the plant. This is typically done by confirming that the analyses documented in NS-TMA-2182 (Reference 2) remain valid or by performing new deterministic analyses for the proposed plant state.

To address the uprate program for CPNPP, the loss of load (LOL) and loss of normal feedwater (LONF) ATWS events were re-analyzed to ensure that the analytical basis for the final ATWS rule continues to be met. The LOL and LONF ATWS events are the two most limiting RCS overpressure transients reported in NS-TMA-2182 (Reference 2). The approach taken was to demonstrate that the ATWS unfavorable exposure time (UET) is less than 5 percent of an operating cycle. UET is the duration of a given cycle for which the core reactivity feedback is insufficient to preclude the RCS pressure from exceeding the Service Level C pressure limit of 3,200 psig following an ATWS event. The objective is to show that the ATWS pressure limit of 3,200 psig is met for at least 95 percent of the cycle, and therefore the analytical basis for the final ATWS rule continues to be met.

The UET approach has been previously approved by the NRC per Reference 3. The analysis must show that the UET, given the cycle design (including moderator temperature coefficient (MTC)), will be less than 5 percent. This 5-percent requirement for the UET is equivalent to the probability level in the reference analyses for the ATWS rule analytical basis (Reference 2). In those analyses, the NRC required that all parameters be best-estimate values with the exception of the MTC initial condition, which is to be at a full-power value that is bounding for at least 95 percent of a given cycle. The UET approach provides a similar level of assurance for the effectiveness of the reactivity feedback.

To determine UET, the reactivity conditions of the core and plant conditions under consideration must be compared to the ATWS analysis conditions that lead to a peak RCS pressure at the ATWS pressure limit of 3,200 psig. The variable conditions of significance to the resulting peak RCS pressure following the LONF and LOL ATWS events are total reactivity feedback (primarily MTC), primary-side pressure relief capacity, and AFW capacity. For a given primary-side pressure relief configuration and AFW capacity, reactivity feedback (MTC) can be adjusted in the ATWS analysis until the peak RCS pressure during the specific ATWS event equals 3,200 psig. At these specific reactivity feedback conditions, the change in power with increasing temperature represents what is defined as the critical power trajectory (CPT) (or heatup/shutdown characteristics) for the specific plant configuration. The heatup/shutdown characteristics of a given core at various times in the cycle can then be compared to the CPT to establish UET for the given core at the specific plant configuration conditions.

2.8.5.7.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The ATWS analyses performed for the SPU Program showed that the results obtained for CPNPP Unit 1 with Westinghouse $\Delta 76$ steam generators are more limiting than those obtained for Unit 2 with D-5 steam generators and, therefore, may be conservatively applied to CPNPP Unit 2. As such, only the Unit 1 inputs, assumptions, and results are reported.

The primary input to the LOL and LONF ATWS analysis for the CPNPP Units 1 and 2 SPU Program was the four-loop reference LOL and LONF ATWS models from the analyses supporting NS-TMA-2182. The following analysis assumptions were used:

- The nominal and initial conditions were updated to the nuclear steam supply system (NSSS) design parameters for 3,628 MWt.
- The steam generator data was revised to reflect the Westinghouse $\Delta 76$ steam generator for the Unit 1 analyses.
- Consistent with the analysis basis for the Final ATWS Rule (NS-TMA-2182):
 - Thermal design flow (TDF) is assumed, no uncertainties are applied to the initial power, RCS average temperature or RCS pressure.
 - Zero-percent steam generator tube plugging (SGTP) is assumed. Zero-percent SGTP is more limiting (that is, results in a higher peak RCS pressure) for ATWS events.
 - Control rod insertion was not assumed.
 - 100-percent pressurizer power-operated relief valve capacity was assumed.
 - The AMSAC actuation setpoint is not directly assumed in the ATWS analyses. Turbine trip and AFW actuation are modeled to occur at generic times after event initiation, consistent with NS-TMA-2182.
- A CPNPP best-estimate AFW flow of 2,148 gpm was assumed.
- The reactivity feedback (MTC) was adjusted until the peak RCS pressure during the specific ATWS event equaled 3,200 psig.

To remain compliant with the basis of the final ATWS rule (10 CFR 50.62), the UET calculated for the ATWS reference conditions (no control rod insertion, nominal AFW flow, and unblocked pressurizer power-relief valves) must be less than 5 percent for a given cycle.

2.8.5.7.2.3 Description of Analyses and Evaluations

ATWS CPTs were generated for the two pressure-limiting ATWS events. The ATWS CPTs were generated based on the four-loop reference LOL and LONF ATWS models from the analyses supporting NS-TMA-2182. The models were revised to incorporate the uprated power conditions reflecting an NSSS power level of 3,628 MWt, the Unit 1 Westinghouse Δ 76 steam generators (the Unit 1 Model Δ 76 steam generators were determined to be limiting compared to the Unit 2 Model D-5 steam generators), and plant-specific, best-estimate AFW flow. The CPTs were then used to determine the ATWS UET.

2.8.5.7.2.4 Results

CPT curves were calculated for CPNPP Unit 1 with Westinghouse Δ 76 steam generators at an uprated NSSS power level of 3,628 MWt. These critical power trajectory curves for the LOL and LONF ATWS transients are shown in Figures 2.8.5.7-1 and 2.8.5.7-2, respectively.

The results of this analysis may be conservatively applied to CPNPP Unit 2 with Model D-5 steam generators since the results obtained for the Model Δ 76 SGs are more limiting than those obtained for the Model D-5 steam generators.

To remain compliant with the basis of the final ATWS rule (10 CFR 50.62), the UET must be less than 5 percent for a given cycle, or equivalently, the ATWS pressure limit of 3,200 psig must be met for 95 percent of the cycle. The UET is met for the anticipated operating conditions with a representative core design and will be checked on a cycle-specific basis. Therefore, the basis of the final ATWS rule (10 CFR 50.62) is met for the CPNPP Units 1 and 2 SPU.

2.8.5.7.3 Conclusion

The information related to ATWS has been reviewed and it was concluded that it has adequately accounted for the proposed CPNPP Units 1 and 2 SPU Program effects on ATWS. The evaluation has demonstrated that the AMSAC continues to meet the requirements of 10 CFR 50.62. The evaluation has shown that the plant is not required by 10 CFR 50.62 to have a diverse scram system. Additionally, the evaluation has shown that the UET, for the anticipated operating conditions with a representative core design, will be less than five percent, or equivalently, that the ATWS pressure limit of 3,200 psig will be met for at least 95 percent of the cycle. The UET will continue to be checked on a cycle-specific basis. Therefore, the proposed uprate is acceptable with respect to ATWS.

2.8.5.7.4 References

1. 10 CFR 50.62 and Supplementary Information Package, "Requirements for Reduction of Risk from ATWS Events for Light Water-Cooled Nuclear Power Plants."
2. NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979.
3. NRC letter to D. L. Farrar (ComEd), "Issuance of Amendments (TAC NOs. M89092, M89093, M89072, and M89091)," July 27, 1995.

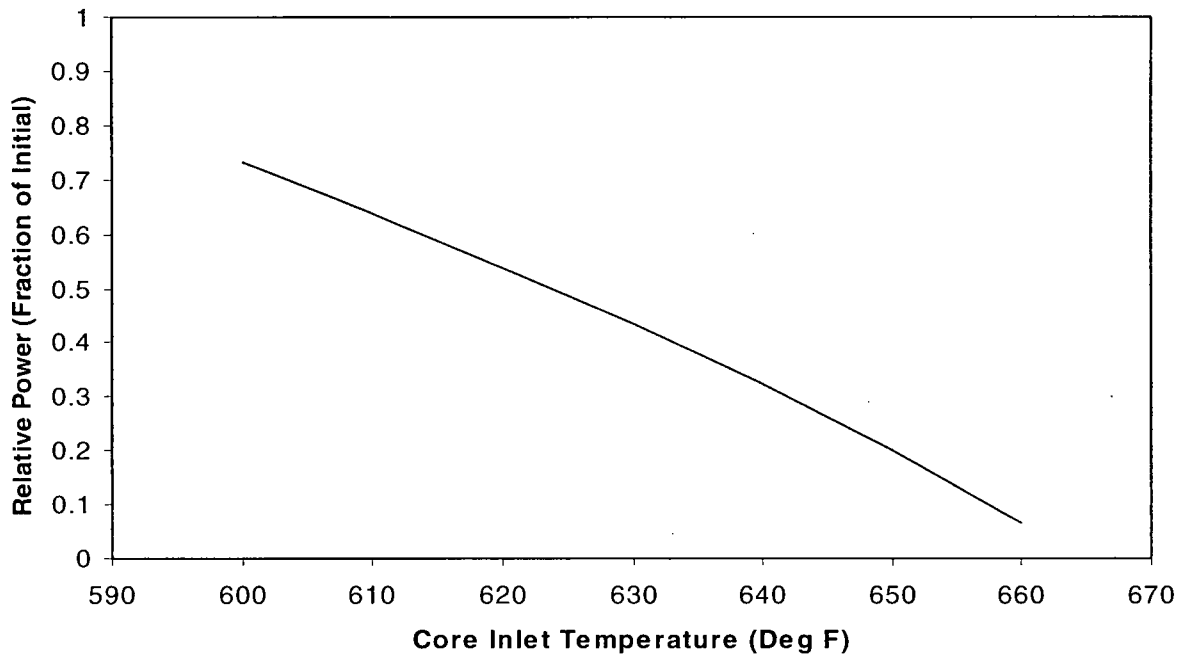


Figure 2.8.5.7-1 Critical Power Trajectory for Loss of Load ATWS at Up-rated NSSS Power Conditions (3,628 MWt)

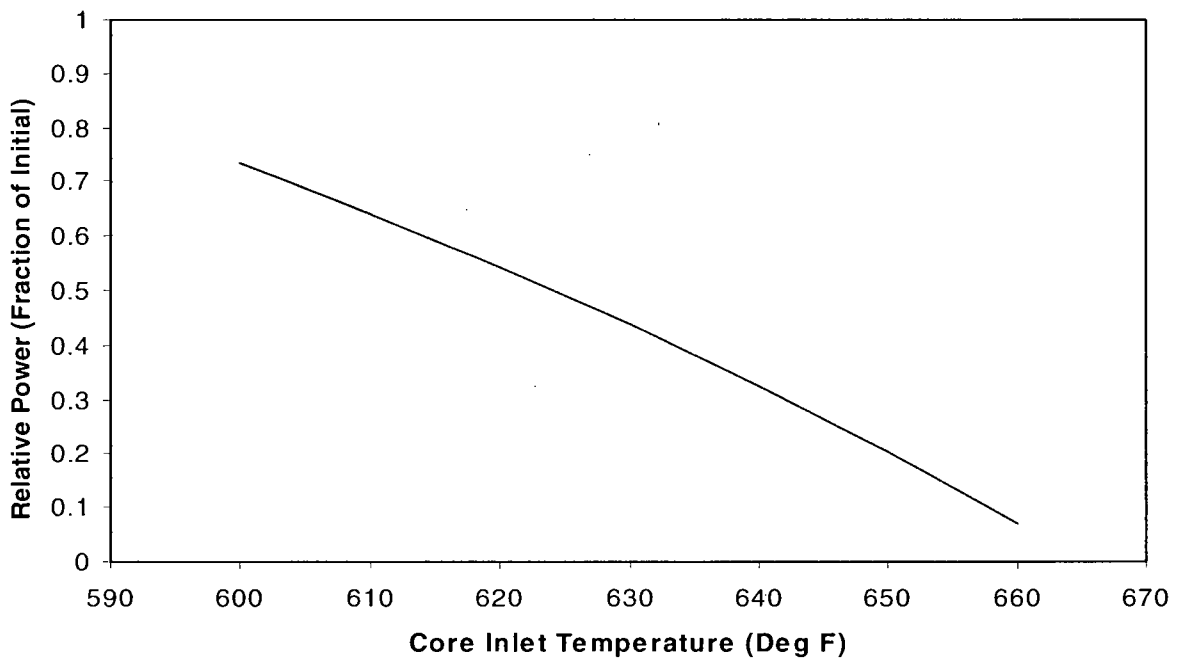


Figure 2.8.5.7-2 Critical Power Trajectory for Loss of Normal Feedwater ATWS at Up-rated NSSS Power Conditions (3,628 MWt)

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

2.8.6.1.1 Regulatory Evaluation

Nuclear power plants include facilities for the storage of new fuel. The quantity of new fuel stored varies from plant to plant, depending on the specific design of the plant and individual refueling needs. The Luminant Power review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions.

The acceptance criteria are based on:

- General Design Criterion (GDC)-62, insofar as it requires that criticality in the fuel storage and handling system be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the new fuel storage regarding conformance to:

- GDC-62, Prevention of Criticality in Fuel Storage and Handling, is described in FSAR Section 3.1.6.3.

Criticality in new fuel and spent fuel storage areas is prevented by both physical separation of fuel assemblies and the presence of boric acid in the spent fuel storage pool. For the new fuel storage racks no changes are needed for the stretch power uprate (SPU) program. For the high density Region I racks in Spent Fuel Pool 1 (SFP1) and Spent Fuel Pool 2 (SFP2), the space between storage positions within the rack module and between individual rack modules is blocked to prevent insertion of fuel assemblies. With respect to criticality, there are no administrative restrictions on placement of spent fuel within the high density Region I racks in SFP1 and SFP2. For the high density Region II racks in SFP1 and SFP2, there are no spaces where assemblies could be inserted between storage positions within the rack module, and the spaces between individual rack modules are blocked to prevent insertion of fuel assemblies. Placement of fuel in the high density Region II racks in SFP1 and SFP2 is administratively controlled based on initial enrichment and minimum burnup requirements. Placement of fuel in the spent fuel pool outside of a rack module, such as a gap between the rack module and pool wall or within the oversized inspection cells, is administratively controlled. A discussion of the SFP Region I and Region II oversized inspection cells is contained in the Technical Specification Bases 3.7.17.

For high density Region I racks (any storage configuration) and high density Region II racks (1-out-of-4 storage configuration) fuel assembly spacing is such that subcriticality is ensured ($k_{\text{eff}} < 0.95$) even if assemblies are immersed in unborated water.

For high density Region II racks (2-out-of-4 storage configuration, 3-out-of-4 storage configuration, and 4-out-of-4 storage configuration) fuel assembly spacing is such that criticality is ensured ($k_{\text{eff}} < 0.95$ taking credit for soluble boron and $k_{\text{eff}} < 1.0$ assuming unborated water).

Criticality prevention and criticality considerations are discussed in FSAR Section 9.1 and 4.3, respectively. CPNPP complies with 10 CFR 50.68(b) in lieu of maintaining a system capable of detecting a criticality event as described in 10 CFR 70.24.

The containment refueling cavity of each unit has additional interim storage space for one low density rack. For the low density rack in the refueling cavity, the space between storage positions within the rack module is blocked to prevent insertion of fuel assemblies. There are no administrative restrictions on placement of spent fuel within the low density rack. Fuel assembly spacing is such that subcriticality is ensured ($k_{\text{eff}} < 0.95$) even if assemblies are immersed in unborated water.

2.8.6.1.2 Technical Evaluation

Luminant Power has reviewed the potential effects of the SPU for CPNPP, and this review has identified the following: (1) There are no pertinent fuel design changes implemented in support of the SPU, (2) there is no increase in the Technical Specification maximum allowed fuel enrichment (5.0 w/o U-235) for the SPU, and (3) there are no modifications to the new fuel storage vault for the SPU. Therefore, Luminant Power concludes that the criticality analysis of record for the new fuel storage vault remains valid for the SPU for CPNPP. No additional analysis is required.

2.8.6.1.3 Conclusion

Luminant Power has reviewed the analyses related to the effect of the new fuel on the analyses for the new fuel storage facilities and concludes that the new fuel storage facilities will continue to meet the requirements of GDC-62 following implementation of the proposed SPU.

2.8.6.2 Spent Fuel Storage

2.8.6.2.1 Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The Luminant Power review covered the effect of the proposed SPU on the criticality analysis (such as reactivity of the spent fuel storage array).

The acceptance criteria are based on:

- General Design Criterion (GDC)-4, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC-62, insofar as it requires that criticality in the fuel storage and handling system be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of the CPNPP spent fuel storage regarding conformance to:

- GDC-4, Environmental and Dynamic Effects Design Bases, is described in FSAR Section 3.1.1.4.

The station's SSCs important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant-accident (LOCA). Environmental conditions are described in FSAR Section 3.11.

These SSCs are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

Details of the design, environmental testing, and construction of these SSCs are included in FSAR Chapters 3, 5, 6, 7, 8, 9, and 10. Evaluation of the performance of safety features is contained in FSAR Chapter 15.

- GDC-62, Prevention of Criticality in Fuel Storage and Handling, is described in FSAR Section 3.1.6.3.

Criticality in new fuel and spent fuel storage areas is prevented by both physical separation of fuel assemblies and the presence of borated water in the spent fuel storage pool. For the high density Region I racks in Spent Fuel Pool 1 (SFP1) and Spent Fuel Pool 2 (SFP2), the space between storage positions within the rack module and between individual rack modules is blocked to prevent insertion of fuel assemblies. With respect to criticality, there are no administrative restrictions on placement of spent fuel within the high density Region I racks in SFP1 and SFP2. For the high density Region II

racks in SFP1 and SFP2, there are no spaces where assemblies could be inserted between storage positions within the rack module, and the spaces between individual rack modules are blocked to prevent insertion of fuel assemblies. Placement of fuel in the high density Region II racks in SFP1 and SFP2 is administratively controlled based on initial enrichment and minimum burnup requirements. Placement of fuel in the spent fuel pool outside of a rack module, such as a gap between the rack module and pool wall or within the oversized inspection cells, is administratively controlled. A discussion of the SFP Region I and Region II oversized inspection cells is contained in the Technical Specification Bases 3.7.17.

For high density Region I racks (any storage configuration) and high density Region II racks (1-out-of-4 storage configuration) fuel assembly spacing is such that subcriticality is ensured ($k_{\text{eff}} < 0.95$) even if assemblies are immersed in unborated water.

For high density Region II racks (2-out-of-4 storage configuration, 3-out-of-4 storage configuration, and 4-out-of-4 storage configuration) fuel assembly spacing is such that subcriticality is ensured ($k_{\text{eff}} < 0.95$ taking credit for soluble boron and $k_{\text{eff}} < 1.0$ assuming unborated water).

Criticality prevention and criticality considerations are discussed in FSAR Sections 9.1 and 4.3, respectively. CPNPP complies with 10 CFR 50.68(b) in lieu of maintaining a system capable of detecting a criticality event as described in 10 CFR 70.24.

The containment refueling cavity of each unit has additional interim storage space for one (1) low density rack. For the low density rack in the refueling cavity, the space between storage positions within the rack module is blocked to prevent insertion of fuel assemblies. There are no administrative restrictions on placement of spent fuel within the low density rack. Fuel assembly spacing is such that subcriticality is ensured ($k_{\text{eff}} < 0.95$) even if assemblies are immersed in unborated water.

2.8.6.2.2 Technical Evaluation

2.8.6.2.2.1 Introduction

The purpose of this section is to describe the results of Region II spent fuel pool criticality calculations due to the proposed SPU. All of the spent fuel storage racks in the CPNPP spent fuel pools were re-analyzed for the proposed SPU. This re-analysis was necessary because the reactivity of spent fuel stored in the spent fuel racks will be different, depending on the reactor core operating conditions present when the fuel was depleted in the reactor. There are no physical changes being made to the spent fuel pool or storage racks due to the SPU. The effects of the SPU on the spent fuel pool cooling system are evaluated in Licensing Report (LR) subsection 2.5.4.1.

2.8.6.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The Region II spent fuel pool criticality safety analysis utilizes input parameters that are appropriate for the proposed SPU of CPNPP. Many of these input parameters are consistent with the Region II spent fuel pool current licensing basis. However, inputs parameters affected by the CPNPP SPU, such as core operating conditions, are updated appropriately. A complete list of the input parameters and assumptions utilized in the spent fuel pool criticality safety analysis is included in WCAP-16827 (Reference 1).

The acceptance criteria for the Region II spent fuel pool criticality safety analysis require that there is a 95-percent probability at a 95-percent confidence level that the effective multiplication factor (k_{eff}) of the spent fuel pool will be less than 1.0 at no soluble boron conditions, and will be less than or equal to 0.95 with soluble boron present.

2.8.6.2.2.3 Description of Analyses

The spent fuel pool criticality safety analysis determined the loading requirements for safe storage of fuel assemblies in the Region II storage racks of the CPNPP spent fuel pool. Reactivity credit for assembly burnup, Pu-241 decay, axial blankets and soluble boron is considered in the analysis. The presence of soluble boron is credited for both normal and postulated accident conditions. The analysis is described in detail in Reference 1.

2.8.6.2.2.4 Results

As shown in Reference 1, the following are the results of the criticality analysis of all the fuel storage configurations in Region II of the CPNPP spent fuel pool:

- Region II 2-out-of-4 fuel storage – A curve of allowable fuel enrichment versus burnup for the SPU for this configuration is specified in Reference 1. The analysis for this storage configuration confirms that there is a 95-percent probability at a 95-percent confidence level that the effective multiplication factor (k_{eff}) will remain less than 1.0 at no soluble boron conditions, and will remain less than 0.95 with soluble boron present. The existing required Technical Specification curve of allowable enrichment versus burnup (Technical Specification Figure 3.7.17-3) for storage of fuel in this configuration will be replaced with the proposed curve for the SPU.
- Region II 3-out-of-4 fuel storage – A curve of allowable fuel enrichment versus burnup and decay times for the SPU for this configuration is specified in Reference 1. An additional curve of allowable fuel enrichment versus burnup and decay times for the SPU for fuel with axial blankets in this configuration is also specified. The analyses for these storage configurations confirms that there is a 95-percent probability at a 95-percent confidence level that the effective multiplication factor (k_{eff}) will remain less than 1.0 at no soluble boron conditions, and will remain less than 0.95 with soluble boron present. The existing required Technical Specification curve of allowable enrichment versus burnup and decay time (Technical Specification Figure 3.7.17-2) for storage of fuel in this configuration will be replaced with the proposed curves for the SPU.

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- Region II 4-out-of-4 fuel storage – A curve of allowable fuel enrichment versus burnup and decay times for the SPU for this configuration is specified in Reference 1. An additional curve of allowable fuel enrichment versus burnup and decay times for the SPU for fuel with axial blankets in this configuration is also specified. The analyses for these storage configurations confirms that there is a 95-percent probability at a 95-percent confidence level that the effective multiplication factor (k_{eff}) will remain less than 1.0 at no soluble boron conditions, and will remain less than 0.95 with soluble boron present. The existing required Technical Specification curve of allowable enrichment versus burnup and decay time (Technical Specification Figure 3.7.17-1) for storage of fuel in this configuration will be replaced with the two proposed curves for the SPU.
 - Accident Conditions – As described in Reference 1, various accidents were analyzed for their effect on the spent fuel pool k_{eff} . Soluble boron is needed to mitigate certain accident conditions. The limiting accident condition, consistent with the current design and licensing basis, is the inadvertent placement or drop of a 5 weight-percent fresh optimized fuel assembly in a vacant storage location, surrounded by other storage locations filled with fuel of Technical Specification limiting allowed reactivity. The analysis confirms that there is a 95-percent probability at a 95-percent confidence level that the effective multiplication factor (k_{eff}) will remain less than 0.95. This analysis credits soluble boron for the limiting accident condition in addition to that required for normal operation. The amount of soluble boron needed per Reference 1 is 964 ppm to mitigate the limiting accident event. When combined with the 643 ppm of soluble boron necessary to maintain k_{eff} less than 0.95 for steady-state operations, the total soluble boron concentration required is 1,607 ppm. This is far less than the current Technical Specification requirement of 2,000 ppm. Therefore, no Technical Specification changes are needed.

In summary, the analysis provided in Reference 1 outlines the storage configurations, limits on soluble boron concentration, and features, enrichment, burnup and decay time of the fuel that ensure that there is a 95-percent probability at a 95-percent confidence level that the effective multiplication factor (k_{eff}) of Region II of the CPNPP spent fuel pool will remain less than 1.0 at no soluble boron conditions, and will be less than or equal to 0.95 with soluble boron present.

The SPU analyses and the associated Technical Specification changes discussed above result in an increased capability of the SFP due to reactivity credits of plutonium decay and axial blankets.

2.8.6.2.3 Conclusions

Luminant Power concludes that the effects of the proposed SPU on the spent fuel rack criticality analyses have been accounted for by a complete criticality reanalysis of Region II fuel storage racks in the spent fuel pool and the results of these analyses are acceptable. Luminant Power concludes that the spent fuel pool design will continue to ensure an acceptable degree of subcriticality following implementation of the proposed SPU. Based on this, Luminant Power concludes that the spent fuel storage facilities will continue to meet the requirements of GDCs-4 and -62 following implementation of the proposed SPU.

2.8.6.2.4 Reference

1. WCAP-16827, "Comanche Peak Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis," July 2007.

2.8.7 Additional Review Areas (Reactor Systems)

2.8.7.1 Auxiliary Systems Pumps, Heat Exchangers, Valves, and Tanks

2.8.7.1.1 Regulatory Evaluation

Evaluations were performed for the essential components of the auxiliary systems at the Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 for impact by the thermal transients and maximum operating temperatures, pressures, and flow rates resulting from the stretch power uprate (SPU) conditions. The evaluation consists of a structural and flow capacity review of the component pressure boundaries. In addition to the systems evaluations discussed in other Licensing Report (LR) sections, the following supplemental evaluations were performed.

The plant auxiliary systems components consist of the pumps, heat exchangers, valves, and tanks listed in Tables 2.8.7.1-1 through 2.8.7.1-5, that are the essential components of the various safety-related and non-safety-related plant auxiliary systems. These components include those installed in the reactor auxiliary cooling water systems required for safe shutdown of the plant during all conditions and for accident prevention and/or mitigation.

The review of reactor auxiliary cooling water system components focused on the effects of the proposed SPU on the various systems' components continued functionality, including the capability to provide heat sink capacity, and withstand any adverse dynamic loads.

The acceptance criteria for this review are:

- General Design Criterion (GDC)-2, insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches without loss of capability to perform their safety functions.
- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.
- GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specific acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

-
- GDC-44, insofar as it requires that a system with the capability to transfer heat from SSCs important-to-safety to an ultimate heat sink be provided. The system safety function shall be to transfer the combined heat load of these SSCs under both normal operating and accident conditions. Redundancy in components and features, isolation capabilities shall be provided the system safety function can be accomplished, assuming a single failure.
 - GDC-61, insofar as it requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity, be designed to assure adequate safety under normal and postulated accident conditions.

The CPNPP design was reviewed in accordance with the guidance documents cited in RS-001.

Current Licensing Basis

Specifically, the adequacy of the CPNPP design relative to:

- GDC-2 is described in Final Safety Analysis Report (FSAR) Section 3.1.1.2, General Design Criterion 2 – Design Bases for Protection Against Natural Phenomena.

Those features of plant facilities that are essential to the prevention of accidents that could affect the public health and safety or to the mitigation of accident consequences are designed to:

1. Quality standards that reflect the importance of the function to be performed. Approved design codes are used when appropriate to the nuclear application.
2. Performance standards that enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquake, flooding condition, wind, ice, or other natural phenomena for the site, and credible combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

All piping, components, and supporting structures of the safety-related systems are designed to withstand a specified seismic disturbance and credible combinations of effects of normal and accident conditions coincident with the effects of natural phenomena. Plant design criteria specify that there is to be no loss of function of such equipment in the event of the safe shutdown earthquake (SSE) ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of Seismic Category I structures to ground acceleration, based on an envelope of characteristics of the site foundation soils and on the critical damping of the foundation and structures, is included in the design analysis.

Design of structures for protection against natural phenomena is described in FSAR Section 3.8. Safety-related structures have sufficient capacity to accept a combination of normal operating loads, functional loads due to the design basis accident, and the

loadings imposed by the maximum wind velocity, or those due to the SSE, whichever is the larger.

- GDC-4 is described in the FSAR Section 3.1.1.4, Environmental and Missile Design Bases.

Structures, systems, and components important to safety are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operating, maintenance, testing, and postulated accidents including LOCAs. These items are either protected from accident conditions or designed to withstand, without failure, exposure to the combination of temperature, pressure, humidity, radiation, and dynamic effects expected during the required operational period.

Physical separation, physical protection, pipe restraints, and redundancy are included in the design of safety-related systems to ensure that each such system performs its intended safety function.

In the Federal Register, Volume 51, No. 70, dated April 11, 1986, the Nuclear Regulatory Commission (NRC) published a final rule modifying GDC-4 to allow use of leak-before-break technology for excluding from the design basis the dynamic effects of postulated ruptures in primary coolant loop piping in pressurized water reactors. This applies to CPNPP as discussed in FSAR Sections 3.6B2.5.1 and 3.6B2.5.

SSCs important to safety are classified as and designed in accordance with the codes and classifications indicated in the FSAR, Section 3.2. FSAR Section 3.11 provides information to demonstrate that the safety-related electrical equipment is capable of performing designated safety-related functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. Protection against the dynamic effects associated with the postulated rupture of pipes is described in the FSAR Section 3.6.

- GDC-5 is described in the FSAR Section 3.1.1.5, Sharing of Structures, Systems and Components.

The sharing of this equipment does not impair its ability to perform its safety, cooldown, or orderly shutdown function.

- GDC-34 is described in the FSAR Section 3.1.4.5, Residual Heat Removal.

The residual heat removal system, in conjunction with the steam and power conversion system, is designed to transfer the fission product decay heat and other residual heat from the reactor core within acceptable limits. The transfer of the heat removal function from the steam and power conversion system to the residual heat removal system occurs when the reactor coolant system is at approximately 350°F and 425 psig.

Suitable redundancy at temperatures below approximately 350°F is accomplished with the two residual heat removal pumps, the two heat exchangers, and the associated piping, cabling, and electric power sources. The residual heat removal system is designed to operate on either the onsite or the offsite electrical power system. Suitable redundancy at temperatures above approximately 350°F is provided by the steam generators and associated piping system.

FSAR Section 5.4.7 contains additional details on how the residual heat removal system is used to transfer fission product decay heat and other residual heat from the reactor core.

FSAR Section 15.2 discusses a number of transients and accidents that could result in a reduction of the capacity of the secondary system to remove heat from the reactor coolant system. Fuel design limits and the design conditions of the reactor coolant pressure boundary (RCPB) are preserved by reactor protection system operation, actuation of pressurizer and steam generator pressure relieving devices, and auxiliary feedwater.

- GDC-44 is described in the FSAR Section 3.1.4.15, Cooling Water

GDC-44, Cooling Water, which states that a system to transfer heat from SSCs important to safety to the ultimate heat sink shall be provided. The safety function is to transfer the combined heat load of these SSCs under normal operating and accident conditions. Redundancy in components and features, and suitable interconnections, Leak detection, and isolation capabilities shall be provided to ensure the safety function can be accomplished, assuming a single failure.

- GDC-61 is described in FSAR Section 3.1.6.2, Fuel Storage and Handling and Radioactive Control

The spent fuel pool and cooling system, fuel handling system, radioactive waste processing systems, and other systems that contain radioactivity are designed to ensure adequate safety under normal and postulated accident conditions.

2.8.7.1.2 Technical Evaluation

2.8.7.1.2.1 Input Parameters and Assumptions

The auxiliary system pumps and requirements are listed in Table 2.8.7.1-1. The heat exchangers and data sheets are listed in Table 2.8.7.1-2. Tables 2.8.7.1-3 and 2.8.7.1-4 list the valves and specification data sources evaluated. Table 2.8.7.1-5 lists the auxiliary tanks. The components design information is contained in the design documents listed in the tables and references. The Performance Capabilities Working Group (PCWG) parameters provided in LR Section 1.1 document the impact of the SPU program on the nuclear steam supply system (NSSS) operating temperatures and pressures. This information was used as input for evaluation of the auxiliary equipment maximum operating temperatures and pressures. In

addition, the impact on the NSSS auxiliary system transients based on the SPU program conditions have been assessed. The impact of the uprate parameters and design transients on the NSSS system performance has been assessed in other LR sections.

The components listed in Tables 2.8.7.1-1 through 2.8.7.1-5 represent the as-shipped hardware provided by Westinghouse in the plant. The impact of the SPU parameters or auxiliary system performance, including current component performance characteristics, was assessed. These system assessments conclude that the auxiliary systems performance characteristics are acceptable at SPU conditions.

2.8.7.1.2.2 Description of Analysis and Evaluations

The design parameters were reviewed for the auxiliary equipment. The specified criteria included design temperature, pressure and flow rates. These parameters were compared to those used in the SPU program to determine if the design parameters continue to bound those parameters of the SPU program.

Auxiliary System Pumps

The NSSS auxiliary pumps reviewed are listed in Table 2.8.7.1-1. The table includes the documents that list the design data that were used in the manufacture of the equipment. Based on LR Section 1.1, there is no impact on the auxiliary system pumps as a result of the SPU program. The operating temperature and pressure ranges for the pumps remain bounded by the current design parameters. The current design transients for the auxiliary equipment bound the transients associated with the SPU program. Consequently, the auxiliary pumps are not impacted by the SPU program.

Auxiliary System Heat Exchangers

The original design parameters for the auxiliary heat exchangers were defined by the equipment specifications and/or purchase order documents as identified in Table 2.8.7.1-2. Based on LR Section 1.1, there is no impact on the auxiliary system heat exchangers as a result of the SPU program. The original design transients for the auxiliary equipment bound the transients associated with the SPU program. Consequently, the auxiliary heat exchangers are not impacted by the SPU program.

Auxiliary System Valves

Six-inch and larger American Society of Mechanical Engineers (ASME) Class 1 and 2 valves originally supplied by Westinghouse are included in this evaluation. The Class 2 valves have been included when fatigue analysis has been performed even though the ASME Code only requires 6-inch and larger Class 1 valves be evaluated for fatigue.

The list of valves for both units was derived from Reference 1. The original design parameters for CPNPP Units 1 and 2 were defined by the equipment specifications identified in Tables 2.8.7.1-3 and 2.8.7.1-4.

As a result of this review and applicability of Reference 2 to the SPU program, the evaluation for the 6-inch and larger valves ASME Class 1 and 2 valves for CPNPP Units 1 and 2 shows the SPU program parameters remain bounded by the previous qualification of the auxiliary valves.

Auxiliary Tanks

A list of auxiliary tanks that were originally supplied by Westinghouse are shown in Table 2.8.7.1-5. The table includes the tank name and data sheets. The original design transients for the auxiliary equipment bound the transients associated with the SPU program. Consequently, the auxiliary tanks are not impacted by the SPU program.

2.8.7.1.3 Conclusion

The revised design conditions have been evaluated with respect to the impact on auxiliary pumps, heat exchangers, valves, and tanks. Based on the results of this review, the auxiliary equipment continues to meet the design pressure and temperature requirements, as well as the fatigue usage factors and allowable limits, for which the equipment is designed.

2.8.7.1.4 References

1. Westinghouse Valve Index "Comanche Peak Units 1 and 2," July 25, 1988.
2. V-EC-1734, Revision 0, "Fatigue Evaluation Westinghouse Supplied Valves for the Comanche Peak Units 1 and 2 Uprating," October 28, 1998.

**Table 2.8.7.1-1
CPNPP Units 1 and 2 (TBX/TCX) Auxiliary Pumps**

Plant	Spin	Description	P.O. #	Ship Date	No. of Pumps	Vendor
TBX	CSAPBA	Boric Acid Transfer Pump	237016	02/02/77	2	CHEMPUMP
TBX	CSAPCH	Centrifugal Charging Pump	236901	05/20/77	2	IDP (PACIFIC)
TBX	WPAPCD	Chemical Drain Tank Pump	237016	02/02/77	1	CHEMPUMP
TBX	TRAPCI	Chiller Pump	178686	09/02/77	2	CRANE DEMING
TBX	WPAPFD	Floor Drain Tank Pump	178686	09/02/77	2	CRANE DEMING
TBX	GHAPGD	Gas Decay Tank Drain Pump	204204	09/13/76	1	CHEMPUMP
TBX	WPAPLT	Laundry & Hot Shower Tank Pump	178686	09/02/77	1	CRANE DEMING
TBX	CSAPPD	Positive Displacement Charging Pump	231321	09/29/78	1	UNION
TBX	WPAPRD	RCS Drain Tank Pump	237016	02/02/77	2	CHEMPUMP
TBX	BRAPRE	Recycle Evap Feed Pump	237016	12/03/76	2	CHEMPUMP
TBX	RHAPRH	Residual Heat Removal Pump	MB-12197-D; 181971	12/30/77	2	IDP (I-R)
TBX	SIAPSI	Safety Injection Pump	236954	10/24/77	2	IDP (PACIFIC)
TBX	WPAPRS	Spent Resin Sluice Pump	237016	03/15/77	1	CHEMPUMP
TBX	WPAPWC	Waste Evap Cond Pump	237016	02/02/77	1	CHEMPUMP
TBX	WPAPWE	Waste Evap Feed Pump	237016	06/29/77	1	CHEMPUMP
TBX	GHAPCP	Waste Gas Compressor Package	226081	07/29/77	2	NASH ENGINEERING
TBX	WPAPMT	Waste Monitor Tank Pump	237016	02/02/77	2	CHEMPUMP
TCX	CSAPBA	Boric Acid Transfer Pump	237017	02/02/77	2	CHEMPUMP
TCX	CSAPCH	Centrifugal Charging Pump	236907	06/24/77	2	IDP (PACIFIC)
TCX	TRAPCI	Chiller Pump	178687	09/02/77	2	CRANE DEMING
TCX	CSAPPD	Positive Displacement Charging Pump	231322	07/26/78	1	UNION
TCX	WPAPRD	RCS Drain Tank Pump	237017	02/20/77	2	CHEMPUMP
TCX	RHAPRH	Residual Heat Removal Pump	MB-12197-D; 181972	12/30/77	2	IDP (I-R)
TCX	SIAPSI	Safety Injection Pump	236959	06/15/77	2	IDP (PACIFIC)
TBX	CSAPBA	Boric Acid Transfer Pump	237016	02/02/77	2	CHEMPUMP

Table 2.8.7.1-2

CPNPP Units 1 and 2 (TBX/TCX) Auxiliary Heat Exchangers

Component Manufacturer	Spin/Drawing No.	Equipment Data Sheet	Equipment Specification	Purchase Order No.
Excess Letdown Hx Atlas Industrial	CSAHEL D-4651	AH-EL645	G-679150 Rev. 1 G-679153 Rev. 1 409A92 Rev. 0	TBX 546-CAZ-241661 TCX 546-CAZ-241662
Seal Water Hx Atlas Industrial	CSAHSW D-4641	AH-SW643	G-679150 Rev. 1 G-679155 Rev. 1 409A89 Rev. 0	TBX 546-CAZ-241661 TCX 546-CAZ-241662
Horizontal Letdown Hx Joseph Oat	CSAHL D 5668	LD-650	G-679150 Rev. 1 G-679152 Rev. 1 409A91 Rev. 0	546-AAZ-215365 546-AAZ-215366
Letdown Chiller Hx Atlas Industrial	TRAHLC D-4628	AH-LC631	G-679150 Rev. 1 955994 Rev. 0	TBX 546-CAZ-241661 TCX 546-CAZ-241662
Letdown Reheat Hx Atlas Industrial	TRAHLR D-4646	AH-LR630	G-679150 Rev. 1	TBX 546-CAZ-241661 TCX 546-CAZ-241662
Moderating Hx Atlas Industrial	TRAHMH D-4623	AH-M629	G-679150 Rev. 1 955995 Rev. 0	TBX 546-CAZ-241661 TCX 546-CAZ-241662
Regenerative Hx Atlas Industrial	CSAHRG D-4313	AH-RG649	G-679150 Rev. 1	TBX 546-CAZ-241661 TCX 546-CAZ-241662
Residual Hx Joseph Oat	RHAHRS 5773	RS-681	G-679150 Rev. 1 G-679154 Rev. 1 409A88 Rev. 0	546-AAZ-241671 546-AAZ-241672
Reactor Coolant Drain Tank Hx Atlas Industrial	WPAHRD 4654	AH-RC708	G-679150 Rev. 1 G-679159 Rev. 1 955996 Rev. 0 409A90 Rev. 0	TBX 546-CAZ-241661 TCX 546-CAZ-241662

**Table 2.8.7.1-3
CPNPP Unit 1 (TBX) Valves E-Spec. List**

E-Spec. No.	Rev.	General or Project Spec.	Title
678837	2	G	Auxiliary Relief Valves ASME III Class 2 and 3
678838	2	G	Pressurizer Safety Valves ASME III Class 1
678839	5	G	Feedwater Control Valves & Feedwater By-Pass Valves ASME III Class 1,2 and 3
678844	2	G	Control Valves ASME III Class 1,2 and 3
678845	1	G	Diaphragm Type Valves ASME III Class 2 and 3
678846	1	G	Butterfly Valves ASME III Class 2 and 3
678847	1	G	Gas Process Control Valves ASME III Class 2 and 3
678852	2	G	Motor Operated Valves ASME III Class 1,2 and 3
678853	2	G	2 in. and larger Manual Globe, Gate and Self Actuated Check Valves ASME III Class 1,2 and 3
952845	1	G	Auxiliary Relief Valves ASME III Class 2 and 3
952852	0	G	Control Valves ASME III Class 1,2 and 3
952855	0	G	1/8 thru 3/8 in. Manually Operated Globe, Throttle and Check Valves ASME III Class 2 and 3
679227	3	P	Three (3") Inch and Larger Manual Globe, Gate and Self-Actuated Check Valves ASME Boiler and Pressure Vessel Code Section III Class 1, 2 and 3
679230	7	P	Motor Operated Valves ASME Boiler and Pressure Vessel Code Section III Class 1, 2 and 3
952216	3	P	Auxiliary Relief Valves ASME Boiler and Pressure Vessel Code Section III Class 2 and 3
952534	1	P	Pressurizer Safety Valves ASME Boiler and Pressure Vessel Code Section III Class 1
952573	4	P	Diaphragm-Type Valves Class 2 and 3 of the ASME Boiler and Pressure Vessel Code, Section III
952581	2	P	Gas Process Control Valves (Class 3 of the ASME Boiler and Pressure Vessel Code, Section III)
952645	8	P	Control Valves ASME Boiler and Pressure Vessel Code, Section III Class 1, 2 and 3
952912	6	P	Feedwater Control Valves and Feedwater Bypass Valves ASME Boiler and Pressure Vessel Code Section III Class 3
953097	3	P	Butterfly Valves ASME Boiler & Pressure Vessel Code, Section III Class 2 and 3
955186	1	G	Solenoid Operated Globe and Throttle Valves ASME III Class 1,2 and 3
955458	0	P	2 Inch and Below Manual Valves (Class 1, 2, & 3 of ASME Boiler and Pressure Vessel Code, Section III)
955563	3	P	Auxiliary Relief Valves ASME Boiler & Pressure Vessel Code Section III Class 2 and 3
955620	1	P	Control Valves (Outside containment-Base Specification G-952852) ASME Boiler and Pressure Vessel Code Section III
952954	0	P	Pressurizer Spray Valve, Class 1 of ASME Boiler and Pressure Vessel Code, Section III

**Table 2.8.7.1-4
CPNPP Unit 2 (TCX) Valves E-Spec. List**

E-Spec. No.	Rev.	General or Project Spec.	Title
678837	2	G	Auxiliary Relief Valves ASME III Class 2 and 3
678838	2	G	Pressurizer Safety Valves ASME III Class 1
678839	5	G	Feedwater Control Valves & Feedwater By-Pass Valves ASME III Class 1,2 and 3
678844	2	G	Control Valves ASME III Class 1,2 and 3
678845	1	G	Diagram Type Valves ASME III Class 2 and 3
678846	1	G	Butterfly Valves ASME III Class 2 and 3
678847	1	G	Gas Process Control Valves ASME III Class 2 and 3
678852	2	G	Motor Operated Valves ASME III Class 1,2 and 3
678853	2	G	3 in. and larger Manual Globe, Gate and Self Actuated Check Valves ASME III Class 1,2 and 3
952845	1	G	Auxiliary Relief Valves ASME III Class 2 and 3
952852	0	G	Control Valves ASME III Class 1,2 and 3
952855	0	G	1/8 thru 3/8 in. Manually Operated Globe, Throttle and Check Valves SSME III Class 2 and 3
679227	3	P	Three (3") Inch and Larger Manual Globe, Gate and Self-Actuated Check Valves ASME Boiler and Pressure Vessel Code Section III Class 1, 2 and 3
679230	7	P	Motor Operated Valves ASME Boiler and Pressure Vessel Code Section III Class 1, 2 and 3
952216	3	P	Auxiliary Relief Valves ASME Boiler and Pressure Vessel Code Section III Class 2 and 3
952534	1	P	Pressurizer Safety Valves ASME Boiler and Pressure Vessel Code Section III Class 1
952573	4	P	Diaphragm-Type Valves Class 2 and 3 of the ASME Boiler and Pressure Vessel Code, Section III
952581	2	P	Gas Process Control Valves (Class 3 of the ASME Boiler and Pressure Vessel Code, Section III)
952645	8	P	Control Valves ASME Boiler and Pressure Vessel Code, Section III Class 1, 2 and 3
952912	6	P	Feedwater Control Valves and Feedwater Bypass Valves ASME Boiler and Pressure Vessel Code Section III Class 3
953097	3	P	Butterfly Valves ASME Boiler & Pressure Vessel Code, Section III Class 2 and 3
955186	1	G	Solenoid Operated Globe and Throttle Valves ASME III Class 1,2 and 3
955458	0	P	2 Inch and Below Manual Valves (Class 1, 2, & 3 of ASME Boiler and Pressure Vessel Code, Section III)
955563	3	P	Auxiliary Relief Valves ASME Boiler & Pressure Vessel Code Section III Class 2 and 3
955620	1	P	Control Valves (Outside containment-Base Specification G-952852) ASME Boiler and Pressure Vessel Code Section III
952954	0	P	Pressurizer Spray Valve, Class 1 of ASME Boiler and Pressure Vessel Code, Section III

**Table 2.8.7.1-5
CPNPP Units 1 and 2 (TBX/TCX) – Tanks**

Component	Data Sheet	
Reagent Tank	EDS-AT-004	Rev. 1
Boric Acid Batching Tank	EDS-AT-025	Rev. 3
Chemical Mixing Tank	EDS-AT-022	Rev. 2
RCP Seal Standpipe	EDS-AT-106	Rev. 2
Volume Control Tank	EDS-AT-033	Rev. 2
Pressurizer Relief Tank	EDS-AT-002	Rev. 4
Accumulator Tank	EDS-AT-038	Rev. 4
Chiller Surge Tank	EDS-AT-019	Rev. 1
Waste Gas Decay Tank	EDS-AT-021	Rev. 5
Chemical Drain Tank	EDS-AT-041	Rev. 4
Waste Evap. Cond. Tank	EDS-AT-040	Rev. 2
Floor Drain Tank	EDS-AT-026	Rev. 2
Laundry and Hot Shower Tank	EDS-AT-024	Rev. 3
Waste Monitor Tank	EDS-AT-023	Rev. 2
Spent Resin Storage Tank	EDS-AT-017	Rev. 3
Reactor Coolant Drain Tank	EDS-AT-016	Rev. 5
Waste Evap. Reagent Tank	EDS-AT-007	Rev. 1
Waste Holdup Tank	EDS-AT-003	Rev. 3

2.8.7.2 Natural Circulation Cooldown

2.8.7.2.1 Regulatory Evaluation

Nuclear Regulatory Commission (NRC) review standard RS-001 Rev. 0 does not explicitly call out guidance documentation for current or post-uprate license basis reviews for natural circulation cooldown.

However, NRC Branch Technical Position (BTP) RSB 5-1, "Design Requirements of the Residual Heat Removal (RHR) System," requires that test programs for pressurized water reactors (PWRs) include tests with supporting analyses to: (1) confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (2) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the Emergency Operating Procedures (EOPs). In addition, the plant is to be designed so that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems. A comparison of performance to that of previously tested plants of similar design may be substituted for these tests.

Natural circulation cooldown became an explicit issue for all plants following Three Mile Island (TMI). For the Comanche Peak Nuclear Power Plant (CPNPP) SPU to 3,612 MWt, an evaluation is performed to show that the natural circulation cooldown exhibits performance consistent with that of a previously tested plant of similar design.

While there are no explicit criteria for this evaluation, the following are reasonable criteria to show acceptable natural circulation cooldown behavior:

- The natural circulation ΔT s and temperatures should be reasonable (that is, bounded by full-power conditions). This helps to avoid any concerns with thermal stresses and also helps to ensure adequate reactor coolant system (RCS) subcooling.
- The steam generator atmospheric relief valves (ARVs) should be capable of cooling down the plant to residual heat removal (RHR) initiation conditions (less than 350°F in the RCS hot legs and less than 365 psia in the RCS) within a reasonable time. Allowing for 4 hours at hot standby and a cooldown rate limited by EOPs to a maximum of 50°F/hour, the time frame for RHR initiation should be on the order of 5 hours – assuming all four ARVs are available and used. For this scenario, a 50°F/hr cooldown rate should remain achievable throughout the cooldown to RHR initiation conditions.
- Compare the hydraulic flow resistance coefficients with those of Diablo Canyon.

Current Licensing Basis

In Section 5.4.3 of NUREG-0797, "Safety Evaluation Report Related to the Operation of Comanche Peak Nuclear Power Plant, Units 1 and 2" (Reference 1) the NRC indicated that verification of adequate mixing of borated water added to the RCS under natural circulation

conditions and confirmation of natural circulation cooldown ability could be accomplished either by reference to the results of the tests from a plant similar in design or actual testing to be conducted at CPNPP. It should be noted that the safe shutdown design basis of CPNPP is hot standby.

A natural circulation/boron mixing/cooldown test was performed at Diablo Canyon Unit 1 on March 28-29, 1985. In a letter dated March 3, 1987, the staff informed Pacific Gas and Electric that it had concluded, on the basis of the Diablo Canyon Unit 1 tests and submittals (WCAP-11086, March 1986) and the Brookhaven National Laboratory technical evaluation report, that the Diablo Canyon Unit 1 systems meet the intent of BTP RSB 5-1 for a Class 2 plant.

The evaluation provided in Final Safety Analysis Report (FSAR) Section Appendix 5A demonstrates that CPNPP natural circulation capabilities are comparable to those of Diablo Canyon Unit 1.

In summary, CPNPP is a Class 2 plant (as defined by the implementation section of BTP RSB 5-1) and is, thus, subject to the technical requirements of RSB 5-1 only as they apply to Class 2 plants. The natural circulation/boron mixing/cooldown test performed at Diablo Canyon on March 28-29, 1985 meets the intent of BTP RSB 5-1 for a Class 2 plant. In order to verify natural circulation cooldown and boron mixing capability per the requirements of BTP RSB 5-1, CPNPP referenced test results from Diablo Canyon Unit 1 that were found to be acceptable.

2.8.7.2.2 Technical Evaluation

Introduction and Description of Evaluations

This section summarizes the CPNPP comparison to the Diablo Canyon natural circulation test results, which has been performed to support the CPNPP SPU to 3,612 MWt. It also summarizes the results of an evaluation performed to verify the plant's ability to cool down on natural circulation to RHR initiation conditions.

To demonstrate capability for natural circulation decay heat removal, many utilities reference the Diablo Canyon Unit 1 natural circulation cooldown test performed in 1985 and provide justification that this test reflects the capability of their plant by comparing relevant parameters. These parameters include hydraulic resistances, power levels, pressures, temperatures, and natural circulation driving heads. For this evaluation, the effects of the SPU on the hydraulic resistances, flow ratio per loop, and thermal driving head are compared to Diablo Canyon.

Natural Circulation

The Diablo Canyon natural circulation test evaluation verified that RCS natural circulation flow could be established, thereby permitting boron mixing and RCS cooldown/depressurization to RHR system initiation conditions. This phase of the test has no specific acceptance criteria and it was evaluated based on the results of boron mixing and cooldown/depressurization phases of the natural circulation cool down test.

The Diablo Canyon test results indicated that natural circulation flow rates were adequate to ensure that core decay heat removal, boron mixing, and plant cooldown/depressurization were maintained throughout the test. The response of the RCS temperatures indicated stable natural circulation conditions throughout the test.

CPNPP and Diablo Canyon Unit 1 were compared in FSAR Appendix 5A to ascertain any differences between the two plants that could potentially affect natural circulation flow. This comparison is still valid. The general configuration of the piping and components in each reactor coolant loop is the same in both CPNPP and Diablo Canyon Unit 1. The elevation head represented by these components and the system piping is similar in both plants. There are no significant differences in the design of the steam generators in the two plants that would adversely affect the natural circulation characteristics.

To compare the natural circulation capabilities of CPNPP and Diablo Canyon, the hydraulic resistance coefficients were also compared. The coefficients were generated on a per-loop basis. The hydraulic resistance coefficients tabulated in Table 2.8.7.2-1 are applicable to normal flow conditions. Although the hydraulic resistance coefficients would increase slightly for natural circulation conditions, the ratio of the total hydraulic flow coefficients remains applicable for natural circulation conditions since the individual hydraulic resistance coefficients for the two comparable plants would be affected in a similar manner. Therefore, the flow ratio per loop as reported in Table 2.8.7.2-1 is expected to be valid for both normal flow and natural circulation conditions.

The general arrangement of the reactor core and internals is the same for CPNPP and Diablo Canyon. The slight variation in the hydraulic resistance coefficient is primarily due to the specific design details of the vessel and internals (such as flow area, upper/lower support plate designs, thermal design flow, or elevations).

The reactor vessel outlet nozzle configuration for both plants is the same. The radius in curvature between the vessel inlet nozzle and downcomer section of the vessel in the two plants is different. Based on 1/7 scale model testing performed by Westinghouse, the radius on the vessel nozzle/vessel downcomer juncture influences the hydraulic resistance of the flow turning from the nozzle to the downcomer. The Diablo Canyon vessel inlet nozzle radius is significantly smaller than that of CPNPP as reflected by the higher coefficient for Diablo Canyon.

The resistance coefficient for the RCS piping for both plants is similar, as reflected in the resistance coefficients reported in Table 2.8.7.2-1.

Steam generator (SG) units were also compared to ascertain any variation that could affect natural circulation capability by changing the effective elevation of the heat sink or the hydraulic resistance seen by the primary coolant. The Diablo Canyon design utilized a Model 51 SG, whereas the CPNPP Unit 1 design utilizes a Model $\Delta 76$ SG and the Unit 2 design utilizes a Model D-5 SG. With respect to the $\Delta 76$ SG, Reference 2 determined that the hydraulic resistance and the natural circulation flow driving head for the CPNPP Unit 1 SGs are similar to those of the original Model D-4 SGs. Reference 2 concluded that "it is anticipated that the $\Delta 76$ SGs will have no adverse impact on the natural circulation cooling capability of the plant."

Design differences between the CPNPP and Diablo Canyon SGs include the following. The tube bundle in the CPNPP SGs is shorter than that in the Model 51, which has an effect on the natural circulation driving head established by the system. The longer tube bundle in the Model 51 SG for Diablo Canyon Unit 1 would result in approximately a 6.3 ± 2.5 percent (3.8 to 8.8-percent) higher driving head when compared to the Model D-5 SGs installed at CPNPP Unit 2. This variance in net driving head is relatively small and should not significantly affect the natural circulation flow rate. The $\Delta 76$ SGs installed in CPNPP Unit 1 have a slightly higher driving head than the Model 51 SG at Diablo Canyon. (Note that the natural circulation mass flow rates and differential temperatures reported in Tables 2.8.7.2-2A, -3A, -2B, and -3B are calculated based on a thermal center in the SGs that is conservatively assumed to be located at the top of the tubesheet.)

Another design difference is that the Model D-5 (CPNPP Unit 2) incorporates a preheater in the lower tube bundle region whereas the Model 51 incorporates a feedwater ring (as does the CPNPP Unit 1 $\Delta 76$ SG, per Reference 3). For natural circulation conditions, the presence of the preheater unit has negligible effects on flow conditions because the auxiliary feedwater flow does not pass directly through the preheater.

The coefficients reported in Table 2.8.7.2-1 represent the resistance seen by the flow in one loop, excluding the resistance through the reactor coolant pump (RCP). The RCP flow resistances for the two plants are on the same order of magnitude as the total hydraulic flow coefficients reported in Table 2.8.7.2-1. The RCP resistances are comparable because the RCP impeller designs for the Diablo Canyon and CPNPP pumps are similar. Accordingly, the flow ratio per loop as reported in Table 2.8.7.2-1 would remain very close to unity when considering RCP flow resistance. Note that the flow ratios reported for CPNPP are greater than unity.

As indicated, the overall hydraulic loss coefficient (excluding RCPs) for CPNPP is lower than that for Diablo Canyon. It is expected that the relative effect of the coefficients seen would be the same under natural circulation conditions. Therefore, based on the flow ratio correlation utilizing the total hydraulic loss coefficient for each plant (and excluding the differences in driving heads and RCP loss coefficients), the CPNPP units will have a natural circulation flow rate approximately 2- to 7-percent higher than that of Diablo Canyon. If the slight differences in the thermal driving head and RCP flow resistance (as detailed previously) are accounted for, the natural circulation flow rate for CPNPP will be approximately 7-percent lower than that for Diablo Canyon. CPNPP natural circulation flow of this magnitude will be sufficiently similar to that obtained for Diablo Canyon such that the Diablo Canyon natural circulation test results are applicable to CPNPP. Slight differences in reactor power and decay heat levels between the two plants would not alter this conclusion.

Boron Mixing

The evaluation of the Diablo Canyon boron-mixing test evaluation demonstrated adequate boron mixing under natural circulation conditions when highly borated water at low temperatures and low flow rates (relative to RCS temperature and flow rates) was injected into the RCS. The acceptance criterion for this phase of the Diablo Canyon test was that RCS hot legs

(Loops 1 and 4) indicate that the active portions of the RCS were borated such that the boron concentration had increased by 250 ppm or more.

Boron injection was conducted at the Diablo Canyon test using the 20,000 ppm boron solution contained in the boron injection tank (BIT). The BIT's contents were flushed into the RCS and, within 12 minutes, natural circulation had provided adequate mixing to increase the boron concentration in the RCS by 340 ppm. Following injection, makeup to the volume control tank (VCT) was set to provide 2,000 ppm boron. This simulated the alignment of the charging pump suction to the refueling water storage tank (RWST). The charging pump discharge was aligned to provide seal injection flow to each RCP and charging flow to one RCS loop. This alignment was continued throughout the remainder of the test causing the boron concentration to further increase.

At CPNPP, safety-grade sources of boron are available for injection into the RCS to provide the required shutdown margin. The primary sources of boron are the two safety-grade boric acid tanks (minimum 7,000 ppm). An alternate water supply for boration and inventory control is the RWST. These sources have a boron concentration significantly less than that available for the Diablo Canyon test. Therefore, to reach the desired change in concentration, a larger quantity of borated water will have to be added over a longer time. As reported in Reference 4, CPNPP has calculated that a change in boron concentration of 300 ppm would take approximately 1 hour without letdown. The calculation assumes that the normal charging line plus the RCP seals provide the boron injection paths at a total boration rate of 75 gpm. This change is reasonably comparable with that demonstrated in the Diablo Canyon test. On this basis, there is reasonable assurance that sufficient time exists for boron injection and mixing to achieve the required shutdown margin. For CPNPP Unit 1, the $\Delta 76$ SG has a slightly larger primary-side volume and would take slightly longer to achieve the desired boron concentration.

During normal operation at CPNPP, the boric acid solution is injected into the RCS via the charging and RCP seal injection lines. On loss of instrument air, the normal charging path may be lost. In this event, a boration flow path is available via the safety injection flow path.

A concern during the boron-mixing period of natural circulation would be inventory control without letdown. Charging with borated water can be accomplished without letdown because the RCS cooldown will result in the contraction of the RCS inventory. The reduction in volume will be sufficient to allow for boration control and maintenance of the required shutdown margin during the cooldown process. Should letdown be desirable, the RCS inventory can be partially released directly to the containment via the reactor head vent system.

In this regard Luminant Power cited, in Reference 4, a calculation that showed that boration and depressurization can be accomplished without letdown and without taking full credit for the available volume created by the cooldown contraction. Should boration without letdown prove impractical due to any combination of plant conditions or equipment failure, the operator could initiate alternate methods of boration and/or depressurization, that is, boration could be accomplished via letdown through the reactor vessel head vent line and depressurization could be accomplished by discharging RCS inventory via the pressurizer power-operated relief valves (PORVs) to the pressurizer relief tank (PRT).

RCS Cooldown

The plant's ability to cool the RCS at a specified cooldown rate, assuming a sufficient supply of auxiliary feedwater and a subcooled RCS, is determined by the capacity of the steam generator ARVs. Steam flow through these valves removes the sensible heat and decay heat throughout the cooldown period. The end of the cooldown period, when the steam generator pressure is low, provides the most limiting conditions for valve capacity. The energy to be removed is determined by the water inventory and the amount of structural material in the RCS, the level of decay heat, and the cooldown rate.

The atmospheric steam dump valves for CPNPP are safety grade. Each of the four air-operated valves has a safety-related accumulator and hand wheel for local operation. Should available air be insufficient, hot standby conditions can be maintained while the air supply is restored or manual action can be taken to permit cooldown. In the event of a single failure, one steam generator would be unavailable for cooldown. Cooling can be provided with the remaining three steam generators.

A potential exists for void formation in the upper head of the reactor vessel during the cooldown/depressurization under natural circulation conditions since the upper head is relatively isolated from the rest of the RCS and the temperature of the upper head fluid remains higher than the temperature of the coolant in the main flow paths of the RCS. Upper head cooling under natural circulation conditions is influenced by core bypass flow and mixing in the upper head.

Westinghouse plants may be divided into two groups according to the magnitude of the bypass flow: T_{hot} and T_{cold} plants. For T_{cold} plants, such as CPNPP, sufficient bypass flow exists to make the temperature of the upper head fluid essentially equal to the cold leg temperature. On the other hand, for T_{hot} plants, which include Diablo Canyon, the bypass flow is much smaller. As a result, the upper head temperature ranges between the cold leg and hot leg temperatures and a possibility of void formation in the upper head region is raised.

Contributing to the T_{cold} design at CPNPP are spray nozzles between the downcomer and the upper head area. These spray nozzles allow better flow communication and mixing in the upper head during natural circulation. The upper head volume for CPNPP is larger than that for Diablo Canyon. The effect of upper head volume on cooling of the upper head is small compared with the contribution of flow through the spray nozzles. Therefore, the head cooling time for a T_{cold} plant is expected to be shorter than that for a T_{hot} plant of similar thermal rating.

To ascertain any difference between the upper head cooling capabilities between CPNPP and Diablo Canyon, a qualitative comparison was made in Reference 4. Since CPNPP is a T_{cold} plant, the upper head region is expected to cool at a rate comparable to or exceeding that of Diablo Canyon. Due primarily to differences in the upper support plate design between the two plants, the upper head region for CPNPP is approximately 87 percent larger than that of Diablo Canyon. The reactor vessel spray nozzles between the downcomer and upper head region have a flow area more than 10 times larger for CPNPP (versus Diablo Canyon), providing the enhanced flow mixing capability that maintains the upper head region at a temperature near the

cold leg temperature. Therefore, adequate upper head cooling for CPNPP is expected during a natural circulation cooldown scenario.

Loop circulation flow is dependent on reactor core decay heat, which is a function of time based on core rating and its power operating history. Under natural circulation flow conditions, flow into the upper head area will constitute only a small percentage of the total core natural circulation flow and, therefore, will not result in an unacceptable reduction in the natural circulation flow required to cool the core.

RCS Depressurization

The depressurization portion of the Diablo Canyon test demonstrated the capability to control pressure in the RCS under natural circulation conditions. Pressure control capability included the ability to maintain adequate RCS pressure without operating the pressurizer heaters and the ability to significantly reduce RCS pressure when needed to initiate RHR system operation. Three methods of reducing pressure were demonstrated. During the RCS cooldown, pressurizer pressure exhibited a downward trend due to ambient heat losses from the pressurizer. This was followed by operator initiated RCS depressurization using auxiliary spray. For auxiliary spray to be effective, the charging lines to the RCS loops must be isolated. Finally, depressurization was completed using a pressurizer PORV. Each method was determined to be effective in reducing RCS pressure.

For CPNPP, pressure control and depressurization capability is similar to Diablo Canyon due to similarities in the design of the RCS and CVCS. Ambient heat losses gradually reduce RCS pressure. Pressurizer PORVs or auxiliary spray are effective in depressurizing the RCS when needed to permit RHR system initiation.

2.8.7.2.2.1 RCS Cooldown Evaluation

This section describes and presents the results of an evaluation performed to demonstrate the ability of the plant to cool down on natural circulation to RHR initiation conditions (~350 psig, and less than 350°F in the RCS hot legs) within a reasonable period of time.

2.8.7.2.2.2 Input Parameters, Assumptions, and Acceptance Criteria

The input parameters for the evaluation are given in Table 2.8.7.2-4. The assumptions are summarized in Table 2.8.7.2-5. While there are no formal acceptance criteria for this evaluation, the guidelines for reasonableness are listed in Licensing Report (LR) subsection 2.8.7.2.1.

2.8.7.2.2.3 Description of Evaluation Methodology

To evaluate the natural circulation capability for the CPNPP Units 1 and 2 SPU Program, the Pressurized Water Reactor Owners Group (PWROG) Emergency Response Guideline (ERG) methodology (Reference 5) is used to estimate RCS flow rates and core ΔT s using core hydraulic resistance coefficients. The equations of Reference 5 are evaluated for several decay heat rates (1, 2, 3, and 4 percent) over a range of T_{avg} values. The calculated flow rates and

loop ΔT s are presented in Tables 2.8.7.2-2A and 2.8.7.2-3A (for Unit 1) and in Tables 2.8.7.2-2B and 2.8.7.2-3B (for Unit 2).

In addition, the ARV capacities are estimated as function of steam generator secondary pressure, which is correlated with primary system T_{cold} . After 4 hours at hot standby conditions, the plant is assumed to cool down to the RHR initiation conditions at the maximum EOP rate (50°F/hour).

The establishment of natural circulation cooldown conditions is expected to take several loop transits after the RCPs have tripped. Thus, the decay heat will be 3 percent at ~3.5 minutes after trip. Therefore, the conditions for the evaluation at hot standby period will be bounded by the 3-percent decay heat values of natural circulation cooldown flow rates and natural circulation cooldown flow ΔT s. The values in Tables 2.8.7.2-2A and 2.8.7.2-3A (for Unit 1) and in Tables 2.8.7.2-2B and 2.8.7.2-3B (for Unit 2) are presented to illustrate the parametric behavior at 1, 2, 3, and 4 percent of decay heat.

The CPNPP SPU to 3,612 MWt will not adversely impact the natural circulation cooldown capability of the plant for the following reasons:

- Acceptable results were found for natural circulation cooling during the hot standby period for residual heat rates as high as 4 percent of 3,634 MWt. The temperatures calculated for this case are bounded by those specified for full-power operation for the uprate high T_{avg} cases.
- More realistically, the decay heat will be ~3 percent of full power by the time the RCPs coast down and the core/hot leg side heats up to quasi-steady-state conditions. Results expected for this situation are the following:
 - Hot leg/core exit temperature = 610.55°F for Unit 1; 611°F for Unit 2⁽¹⁾
 - Hot leg to cold leg ΔT = 43.55°F for Unit 1; 44°F for Unit 2⁽¹⁾
 - Cold leg temperature = 557°F
(controlled to no load per the EOPs)
 - Core flow rate $\cong 5.972 \times 10^6$ lb_m/hr (~4.19% of nominal), for Unit 1
 $\cong 5.906 \times 10^6$ lb_m/hr (~4.15% of nominal), for Unit 2
- The steam generator ARV capacity at the uprated conditions is sufficient to achieve cooldown to the RHR entry point within 5 hours of commencing cooldown at a rate of 50°F/hr, following 4 hours at hot standby – assuming all four ARVs are available and used. The 50°F/hr cooldown rate can be maintained throughout the cooldown to RHR initiation conditions.

1. Four generators steaming through main steam safety valves.

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- If only two SG ARVs are available and used, the plant can be cooled down to a SG pressure of 100 psia within 5 hours of commencing cooldown at a rate of 50°F/hr, following 4 hours at hot standby. On only two SG ARVs, RHR initiation conditions can be achieved in ~6.2 hours, following 4 hours at hot standby, although the cooldown rate can no longer be maintained at 50°F after ~8.25 hours post-trip. These results bound the single-failure case, which is a plant cooldown on three SGs to RHR initiation conditions. Therefore, in the event of a single-failure, the plant can still be cooled down to RHR conditions in a reasonable time.

2.8.7.2.2.4 Results

The Diablo Canyon Unit 1 natural circulation/boron mixing/cooldown test demonstrated that the plant can safely be taken to cold shutdown under natural circulation conditions.

In order to apply the test results to CPNPP, a general comparison of the plant systems and equipment that affect natural circulation, boron mixing, cooldown, and depressurization capabilities has been made between the CPNPP and Diablo Canyon Unit 1 plants in Reference 4. This comparison is still valid. The evaluation performed in support of the CPNPP 3,612 MWt SPU verifies that the CPNPP capabilities remain comparable to those of Diablo Canyon Unit 1. Therefore, it is concluded that CPNPP at the SPU conditions meets the testing comparison requirements of BTP RSB 5-1, Design Requirements for Decay Heat Removal Systems.

2.8.7.2.3 Conclusion

Luminant Power has reviewed the assessment of the effects of the proposed SPU on natural circulation cooldown and concludes that the evaluation has adequately accounted for the effects of changes in plant conditions. Luminant Power concludes that CPNPP Units 1 and 2 maintain the ability to perform a natural circulation cooldown following a trip from full power to RHR initiation conditions. Therefore, Luminant Power finds the proposed SPU acceptable with respect to the systems used for natural circulation cooldown.

2.8.7.2.4 References

1. NUREG-0797, "Safety Evaluation Report Related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2," July 1981 and Supplement No. 22, January 1990.
2. WCAP-16469, Revision 1, "Comanche Peak Unit 1 Replacement Steam Generator Program NSSS Engineering Report," June 2006.
3. WCAP-16480, Revision 1, "Delta 76 Replacement Steam Generator Thermal and Hydraulic Design Analysis Report for Comanche Peak Unit 1," April 2007.

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4. Comanche Peak Nuclear Power Plant (CPNPP), Final Safety Analysis Report (FSAR), Appendix 5A, Amendment 75, November 18, 1988.
 5. Pressurized Water Reactor Owners Group (PWROG), "Emergency Response Guidelines (ERGs), Revision 2, Executive Volume," High-Pressure/Low-Pressure Version, Generic Issue – Natural Circulation, April 30, 2005.

Table 2.8.7.2-1			
Diablo Canyon Versus Comanche Peak Hydraulic Resistance Coefficients for Normal Flow Conditions			
Hydraulic Resistance Coefficients for Normal Flow Conditions			
	Diablo Canyon (ft/(gpm) ²)	CPNPP Unit 1 (ft/(gpm) ²)	CPNPP Unit 2 (ft/(gpm) ²)
Reactor Core and Internals	129.0E-10	125.4E-10	125.4E-10
Reactor Nozzles	36.1E-10	26.6E-10	26.6E-10
Reactor Coolant Loop Piping	20.9E-10	24.4E-10	25.1E-10
Steam Generator	112.0E-10	94.9E-10	114.1E-10
Total Hydraulic Flow Resistance Coefficient (without RCPs)	298.0E-10	271.3E-10	291.3E-10
$\text{Flow Ratio per loop} = \left(\frac{\text{HFC}_{\text{TOT}} \text{ For Diablo Canyon}}{\text{HFC}_{\text{TOT}} \text{ For CPSES U1}} \right)^{1/2} = 1.048$			
$\text{Flow Ratio per loop} = \left(\frac{\text{HFC}_{\text{TOT}} \text{ For Diablo Canyon}}{\text{HFC}_{\text{TOT}} \text{ For CPSES U2}} \right)^{1/2} = 1.011$			

Table 2.8.7.2-2A

**Unit-1 Summary of Natural Circulation Flow Rates
For Various Values of T_{avg} and Decay Heat⁽¹⁾**

T_{avg} °F	Flow Rate (for 1% decay heat) lb _m /hr	Flow Rate (for 2% decay heat) lb _m /hr	Flow Rate (for 3% decay heat) lb _m /hr	Flow Rate (for 4% decay heat) lb _m /hr
300	3.721E+06	4.688E+06	5.366E+06	5.906E+06
350	3.816E+06	4.807E+06	5.503E+06	6.057E+06
400	3.898E+06	4.911E+06	5.622E+06	6.188E+06
450	3.975E+06	5.008E+06	5.733E+06	6.310E+06
500	4.053E+06	5.106E+06	5.845E+06	6.433E+06
550	4.141E+06	5.217E+06	5.972E+06	6.574E+06
600	4.267E+06	5.377E+06	6.155E+06	6.774E+06

Note:

1. Decay heat expressed as percentage of 3,634 MWt.

Table 2.8.7.2-3A

**Unit-1 Summary of Natural Circulation Temperature Differentials (ΔT s)
For Various Values of T_{avg} and Decay Heat⁽¹⁾**

T_{avg} °F	ΔT (for 1% decay heat) °F	ΔT (for 2% decay heat) °F	ΔT (for 3% decay heat) °F	ΔT (for 4% decay heat) °F
300	32.73	51.95	68.07	82.47
350	31.35	49.76	65.21	78.99
400	29.93	47.50	62.25	75.41
450	28.33	44.98	58.94	71.40
500	26.42	41.93	54.95	66.57
550	23.90	37.93	49.71	60.21
600	20.08	31.88	41.77	50.60

Note:

1. Decay heat expressed as percentage of 3,634 MWt.

Table 2.8.7.2-2B

**Unit-2 Summary of Natural Circulation Flow Rates
For Various Values of T_{avg} and Decay Heat⁽¹⁾**

T_{avg} °F	Flow Rate (for 1% decay heat) lb _m /hr	Flow Rate (for 2% decay heat) lb _m /hr	Flow Rate (for 3% decay heat) lb _m /hr	Flow Rate (for 4% decay heat) lb _m /hr
300	3.679E+06	4.635E+06	5.306E+06	5.840E+06
350	3.773E+06	4.754E+06	5.442E+06	5.989E+06
400	3.855E+06	4.856E+06	5.559E+06	6.119E+06
450	3.930E+06	4.952E+06	5.669E+06	6.239E+06
500	4.007E+06	5.049E+06	5.779E+06	6.361E+06
550	4.095E+06	5.159E+06	5.906E+06	6.500E+06
600	4.220E+06	5.316E+06	6.086E+06	6.698E+06

Note:

1. Decay heat expressed as percentage of 3,634 MWt.

Table 2.8.7.2-3B

**Unit-2 Summary of Natural Circulation Temperature Differentials (ΔT s)
For Various Values of T_{avg} and Decay Heat⁽¹⁾**

T_{avg} °F	ΔT (for 1% decay heat) °F	ΔT (for 2% decay heat) °F	ΔT (for 3% decay heat) °F	ΔT (for 4% decay heat) °F
300	33.10	52.54	68.85	83.40
350	31.70	50.33	65.95	79.89
400	30.27	48.04	62.95	76.26
450	28.66	45.49	59.61	72.21
500	26.72	42.41	55.57	67.32
550	24.17	38.36	50.27	60.90
600	20.31	32.24	42.24	51.17

Note:

1. Decay heat expressed as percentage of 3,634 MWt.

Table 2.8.7.2-4			
Input Parameters for Natural Circulation Cooldown Evaluation			
Name	Units	Value	Comment
Power Level	MWt	3,634	Includes uncertainty of 0.6%
Tube Plugging	percent	10	
Inlet Temperature	°F	558	Highest uprate case
Core Flow Rate	10E+6 lbm/hr	142.4	
Maximum Cooldown Rate	°F/hour	50	EOP EOS-0.2A (B), Natural Circulation Cooldown

Table 2.8.7.2-5	
Assumptions for Natural Circulation Cooldown Evaluation	
Number	Assumption
1	Decay heat rates are based on ANSI/ANS-5.1-1979, including 2-sigma uncertainty.
2	Nominal ARV capacities are assumed for the cooldown portion of the transient.
3	The hydraulic resistance of the RCP is calculated using the forward-flow locked-rotor "k" for RCP.

2.8.7.3 Loss of Residual Heat Removal at Mid-Loop

2.8.7.3.1 Regulatory Evaluation

Nuclear Regulatory (NRC) Generic Letter (GL) 88-17, Loss of Decay Heat Removal, identified actions to be taken to preclude loss of decay heat removal during nonpower operations. These actions included operator training and the development of procedures and hardware modifications as necessary to prevent the loss of decay heat removal during reduced reactor coolant inventory operations, to mitigate accidents before they progress to core damage, and to control radioactive material if a core damage accident should occur. Procedures and administrative controls were required that cover reduced inventory operations and ensure that all hot legs are not blocked by nozzle dams unless a vent path is provided that is large enough to prevent pressurization and loss of water from the reactor vessel. Instrumentation was required to provide continuous core exit temperature and reactor water level indication. Sufficient equipment was required to be maintained in an operable or available status so as to mitigate the loss of the residual heat removal (RHR) cooling or loss of reactor coolant system (RCS) inventory should such an event occur during mid-loop or reduced inventory conditions.

There are no specific NRC acceptance criteria within NRC regulations for operations at mid-loop or reduced inventory conditions. However, the NRC requested all holders of operating licenses to respond to the following recommended actions identified in GL 88-17:

- Provide training prior to operating in a reduced inventory condition.
- Implement procedures and administrative controls that reasonably ensure that containment closure will be achieved prior to the time at which core uncover could result from a loss of decay heat removal coupled with an inability to initiate alternate cooling or addition of water to the RCS inventory.
- Provide at least two independent, continuous temperature indications that are representative of the core exit conditions whenever the RCS is in a mid-loop condition and the reactor vessel head is located on top of the reactor vessel.
- Provide at least two independent, continuous RCS water level indications whenever the RCS is in a reduced inventory condition.
- Implement procedures and administrative controls that generally avoid operations that deliberately or knowingly lead to perturbations to the RCS and/or systems that are necessary to maintain the RCS in a stable and controlled conditions while the RCS is in a reduced inventory condition.
- Provide at least two available or operable means of adding inventory to the RCS that are in addition to pumps that are a part of the normal decay heat removal systems.

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- Implement procedures and administrative controls that reasonably ensure that all hot legs are not blocked simultaneously by nozzle dams unless a vent path is provided that is large enough to prevent pressurization of the upper plenum of the reactor vessel.

Current Licensing Basis

The adequacy of the Comanche Peak Nuclear Power Plant (CPNPP) design as it pertains to GL 88-17, is addressed as part of the outage risk management for Units 1 and 2. There have also been actions taken at CPNPP Units 1 and 2 per the NRC recommendations described in GL 88-17 and listed above.

2.8.7.3.2 Technical Evaluation

2.8.7.3.2.1 Introduction

As a result of the SPU, the decay heat at a given time after shutdown increases, roughly in proportion to the SPU. This, in turn, reduces the time to boiling and the time to core uncover following a postulated loss of RHR cooling.

An evaluation is performed for CPNPP Units 1 and 2 as part of the outage risk management evaluation for each plant outage. This evaluation addresses the issues raised by the NRC for the loss of RHR cooling during nonpower operations.

2.8.7.3.2.2 Description of Analyses and Evaluations

A “time to boil” calculation is performed for CPNPP at the nuclear steam supply system (NSSS) design parameters approved for the SPU (Licensing Report (LR) Section 1.1, NSSS Parameters). This calculation is used as an input to the evaluation of shutdown risk.

As part of the analysis of the uprate conditions for CPNPP Units 1 and 2, an evaluation is performed to address the risks associated with an outage, including the loss of RHR cooling at reduced inventory conditions. Included in this analysis are the inputs required from the “time to boil” calculation.

In addition to the shutdown risk that is considered for each outage, plant modifications, training (initial and continuing), and procedure changes have been implemented to address the NRC recommendations that cover the other methods of mitigating the loss of RHR cooling or loss of RCS inventory should such an event occur during mid-loop or reduced inventory conditions.

2.8.7.3.2.3 Results

The results from the shutdown risk assessment are used for information purposes at CPNPP. This information will be used as part of the overall outage risk management assessment for CPNPP Units 1 and 2.

There will be continued training and adherence to the procedural changes implemented as a response to GL 88-17.

2.8.7.3.3 Conclusion

The licensing basis for the loss of RHR at mid-loop conditions, as it pertains to GL 88-17, is addressed for Units 1 and 2 in the shutdown risk assessment which is part of the overall outage risk management consideration. This assessment carries out the recommendations to ensure that CPNPP mitigates the loss of RHR cooling or loss of RCS inventory should such an event occur during mid-loop or reduced inventory conditions.

Additionally, there have been plant modifications, training (initial and continuing), and procedure changes that have been implemented to address the NRC recommendations that cover the other methods of mitigating the loss of RHR cooling or loss of RCS inventory should such an event occur during mid-loop or reduced inventory conditions. These actions were not required as a result of GL 88-17, but adhere to the recommendations given by the NRC. The actions will be continuously implemented for the SPU.

2.9 SOURCE TERMS AND RADIOLOGICAL CONSEQUENCES

2.9.1 Source Terms for Radwaste Systems Analyses

2.9.1.1 Regulatory Evaluation

Luminant Power reviewed the radioactive source term associated with the stretch power uprate (SPU) to ensure the adequacy of the sources of radioactivity used as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The review included the parameters used to determine:

- Concentration of each radionuclide in the reactor coolant
- The fraction of fission product activity released to the reactor coolant
- Concentrations of all radionuclides other than fission products in the reactor coolant
- Leakage rates and associated fluid activity of all potentially radioactive water and steam systems
- Potential sources of radioactive materials in effluents that are not considered in the Final Safety Analysis Report (FSAR) related to liquid waste management systems and gaseous waste management systems

The Nuclear Regulatory Commission (NRC) acceptance criteria for source terms are based on:

- 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas
- 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the “as low as is reasonably achievable” criterion
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to

- GDC-60, Control of Releases of Radioactive Materials to the Environment, is described in FSAR Section 3.1.6.1.

Waste handling systems are incorporated in the facility design for processing and/or retention of radioactive wastes for normal operation and anticipated operational occurrences. Controls and monitoring are provided to ensure that releases during normal operation do not exceed a few percent to the limits of 10 CFR Part 20 and yield offsite doses within the numerical guides for design objectives and limiting conditions of operation set forth in 10 CFR Part 50, Appendix I.

Section 9.4 of the FSAR describes the primary plant ventilation system and the non-engineered safety feature (ESF) exhaust units that satisfy GDC-60.

Chapter 11 of the FSAR describes the radioactive waste processing systems' design criteria, holdup capacities, and estimated releases of radioactive effluents to the environment. Compliance with 10 CFR Part 50, Appendix I is described in FSAR Appendix 11A.

2.9.1.2 Technical Evaluation

2.9.1.2.1 Radiation Sources for Uprate

2.9.1.2.1.1 Input Parameters, Assumptions, and Acceptance Criteria

Radiation source terms are used as input to several radiological consequence analyses that are required in support of the CPNPP Units 1 and 2 Uprate Program. In this section, an evaluation is provided that updates the required radiation sources based on operation at a nominal core power level of 3,612 MWt with additional margins added to account for core power uncertainty and for fuel cycle design variability from cycle to cycle. The results of this evaluation are summarized in LR subsection 2.9.1.2.1.3.

2.9.1.2.1.2 Description of the Evaluation of the Radiation Sources

The source term information for use in the radiological consequence analyses described later in this section was generated using the ORIGEN-S and FIPCO-V (References 1 and 2) computer codes. The ORIGEN-S code is a versatile point depletion and radioactive decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein. The ORIGEN-S code is an industry standard code based on the latest industry experimental data. The FIPCO-V computer coded calculates the buildup of fission product activities in plant systems and components including the RCS, chemical and volume control system (CVCS) demineralizer resins, and volume control tank liquid and vapor phases. The time-dependent inventory of the core fission products calculated by ORIGEN-S is used as input to the FIPCO-V evaluations.

In the analysis for the CPNPP Units 1 and 2 uprate, the radiological sources were updated to account for the projected increase in core power level and also to account for the longer fuel cycles and higher fuel enrichments and burnup that are characteristic of more recent fuel cycle designs. In particular, the current analysis is based on a conservative core power level of 3,684 MWt with additional margins added to account for core power uncertainty and potential future fuel cycle design differences. The core operating scenario consisted of irradiation times of 495, 990, and 1,485 effective full-power days (EFPDs) resulting in a final discharge region burnup of 66,000 MWD/MTU.

2.9.1.2.1.3 Results of the Evaluation of Radiation Sources

Results of the radiation source calculations for the CPNPP Units 1 and 2 SPU are provided in Tables 2.9.1.2.1-1 and 2.9.1.2.1-2. In Table 2.9.1.2.1-1, the total core inventory (193 fuel assemblies) of noble gas and halogen isotopes at the end of an equilibrium fuel cycle consisting of a three region core with the regions operating for 495, 990, and 1,485 EFPDs, respectively is given. The isotopic inventories are provided at reactor shutdown as well as for a series of decay times ranging from 50 hours to 60 days. In Table 2.9.1.2.1-2, a summary of the calculated RCS inventory of radioactive isotopes is provided.

The radiation source terms generated with the current analysis are suitable for application to the CPNPP SPU. The inclusion of the 4-percent additional margin allows for reasonable variability in the design of future fuel cycles.

2.9.1.3 Conclusion

The SPU radiation source term calculation parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. In addition, the projected radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I and GDC-60. Therefore, Luminant Power finds the proposed SPU acceptable with respect to source terms.

The radiological consequences resulting from incidents addressed in Licensing Report (LR) subsections 2.9.2 through 2.9.9, and 2.9.11 and 2.9.12 are based on the projected source term.

2.9.1.4 References

1. ORNL/TM-2005/39, Version 5, Volume II, Book 1, Section F7, "ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," April 2005.
2. WCAP-7949, "FIPCO-V, A Computer Code for Calculating the Distribution of Fission Products in Reactor Systems," August 1972.

Table 2.9.1.2.1-1

Total Core Inventory of Noble Gases and Halogens as a Function of Time After Shutdown

Total Core Inventory [Curies]										
	Discharge	Decay Time [days]								
		50 hr	75 hr	100 hr	25 d	25.5 d	26 d	26.5 d	27 d	27.5 d
br82	4.42E+05	1.66E+05	1.02E+05	6.22E+04	3.39E+00	2.68E+00	2.12E+00	1.67E+00	1.32E+00	1.04E+00
br83	1.15E+07	6.71E+00	4.90E-03	3.57E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br84	2.04E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr83m	1.15E+07	2.80E+01	2.06E-02	1.50E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85m	2.43E+07	1.08E+04	2.25E+02	4.70E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06
kr87	4.75E+07	7.08E-05	8.55E-11	1.03E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr88	6.36E+07	3.19E+02	7.14E-01	1.60E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr89	7.85E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i130	2.97E+06	1.81E+05	4.46E+04	1.10E+04	7.29E-09	3.72E-09	1.90E-09	9.68E-10	4.94E-10	2.52E-10
i131	1.02E+08	8.71E+07	8.00E+07	7.34E+07	1.22E+07	1.17E+07	1.12E+07	1.08E+07	1.03E+07	9.86E+06
i132	1.47E+08	9.49E+07	7.61E+07	6.10E+07	7.24E+05	6.51E+05	5.85E+05	5.26E+05	4.73E+05	4.25E+05
i133	2.06E+08	4.00E+07	1.74E+07	7.55E+06	4.38E-01	2.94E-01	1.97E-01	1.32E-01	8.85E-02	5.93E-02
i134	2.30E+08	6.34E-09	1.64E-17	4.25E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	1.96E+08	1.01E+06	7.19E+04	5.14E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe131m	1.09E+06	1.08E+06	1.07E+06	1.06E+06	5.34E+05	5.22E+05	5.11E+05	4.99E+05	4.88E+05	4.77E+05
xe133m	6.44E+06	4.64E+06	3.55E+06	2.65E+06	3.82E+03	3.26E+03	2.78E+03	2.37E+03	2.03E+03	1.73E+03
xe133	2.06E+08	1.81E+08	1.61E+08	1.42E+08	9.28E+06	8.69E+06	8.13E+06	7.61E+06	7.13E+06	6.67E+06
xe135m	4.39E+07	1.64E+05	1.18E+04	8.40E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	3.65E+07	9.62E+06	1.65E+06	2.62E+05	9.37E-12	3.77E-12	1.52E-12	6.11E-13	2.46E-13	9.90E-14
xe138	1.74E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 2.9.1.2.1-1 (cont.)

Total Core Inventory of Noble Gases and Halogens as a Function of Time After Shutdown

	Total Core Inventory [Curies]								
	Decay Time [days]								
	28.0 d	28.5 d	29.0 d	29.5 d	30.0 d	30.5 d	31.0 d	31.5 d	32.0 d
br82	8.25E-01	6.52E-01	5.15E-01	4.07E-01	3.22E-01	2.54E-01	2.01E-01	1.59E-01	1.25E-01
br83	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br84	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr83m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06
kr87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i130	1.29E-10	6.56E-11	3.35E-11	1.71E-11	8.71E-12	4.44E-12	2.27E-12	1.16E-12	5.90E-13
i131	9.45E+06	9.05E+06	8.67E+06	8.30E+06	7.95E+06	7.62E+06	7.30E+06	6.99E+06	6.69E+06
i132	3.83E+05	3.44E+05	3.09E+05	2.78E+05	2.50E+05	2.25E+05	2.02E+05	1.82E+05	1.63E+05
i133	3.98E-02	2.67E-02	1.79E-02	1.20E-02	8.04E-03	5.39E-03	3.61E-03	2.42E-03	1.62E-03
i134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe131m	4.66E+05	4.56E+05	4.45E+05	4.35E+05	4.25E+05	4.16E+05	4.06E+05	3.97E+05	3.87E+05
xe133m	1.48E+03	1.26E+03	1.08E+03	9.19E+02	7.84E+02	6.70E+02	5.72E+02	4.88E+02	4.17E+02
xe133	6.24E+06	5.84E+06	5.47E+06	5.12E+06	4.79E+06	4.49E+06	4.20E+06	3.93E+06	3.68E+06
xe135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 2.9.1.2.1-1 (cont.)

Total Core Inventory of Noble Gases and Halogens as a Function of Time After Shutdown

	Total Core Inventory [Curies]								
	Decay Time [days]								
	32.5 d	33.0 d	33.5 d	34.0 d	34.5 d	35.0 d	40.0 d	50.0 d	60.0 d
br82	9.90E-02	7.82E-02	6.18E-02	4.88E-02	3.86E-02	3.05E-02	2.89E-03	2.60E-05	2.33E-07
br83	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
br84	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr83m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.13E+06	1.12E+06	1.12E+06
kr87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i130	3.01E-13	1.54E-13	7.84E-14	4.00E-14	2.04E-14	1.04E-14	1.24E-17	1.78E-23	2.54E-29
i131	6.41E+06	6.14E+06	5.88E+06	5.63E+06	5.40E+06	5.17E+06	3.36E+06	1.42E+06	5.99E+05
i132	1.47E+05	1.32E+05	1.19E+05	1.07E+05	9.60E+04	8.63E+04	2.98E+04	3.55E+03	4.23E+02
i133	1.09E-03	7.29E-04	4.89E-04	3.28E-04	2.20E-04	1.47E-04	2.70E-06	9.08E-10	3.05E-13
i134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe131m	3.78E+05	3.69E+05	3.60E+05	3.52E+05	3.44E+05	3.35E+05	2.62E+05	1.57E+05	9.19E+04
xe133m	3.55E+02	3.03E+02	2.59E+02	2.21E+02	1.89E+02	1.61E+02	3.31E+01	1.40E+00	5.90E-02
xe133	3.44E+06	3.22E+06	3.02E+06	2.82E+06	2.64E+06	2.48E+06	1.28E+06	3.41E+05	9.08E+04
xe135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe138	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 2.9.1.2.1-2

**Reactor Coolant System Specific Activities
(No Volume Control Tank Purge)**

Nuclide	Activity ($\mu\text{Ci/g}$)	Nuclide	Activity ($\mu\text{Ci/g}$)	Nuclide	Activity ($\mu\text{Ci/g}$)
Kr-83m	4.766E-01	Rb-88	4.536E+00	H-3	3.5
Kr-85m	1.953E+00	Rb-89	2.068E-01	Cr-51	5.00E-3
Kr-85	8.729E+00	Sr-89	4.458E-03	Mn-56	1.40E-2
Kr-87	1.294E+00	Sr-90	1.774E-04	Fe-55	2.10E-3
Kr-88	3.606E+00	Sr-91	8.132E-03	Fe-59	5.10E-4
Kr-89	1.039E-01	Sr-92	1.365E-03	Co-58	1.20E-2
Xe-131m	3.196E+00	Y-90	4.933E-05	Co-60	1.40E-3
Xe-133m	4.808E+00	Y-91m	4.337E-03	Rb-86	2.277E-02
Xe-133	2.801E+02	Y-91	6.122E-04	Ba-137m ⁽²⁾	8.632E-01
Xe-135m	5.355E-01	Y-92	1.177E-03	Ba-140	4.426E-03
Xe-135	8.180E+00	Y-93	3.861E-04	La-140	1.467E-03
Xe-137	1.961E-01	Zr-95	6.941E-04	Ce-141	6.691E-04
Xe-138	7.247E-01	Nb-95	6.861E-04	Ce-143	5.279E-04
Br-83	1.017E-01	Mo-99	8.142E-01	Pr-143	6.570E-04
Br-84	5.236E-02	Tc-99m	7.509E-01	Ce-144	4.762E-04
Br-85	6.168E-03	Ru-103	5.773E-04	Pr-144 ⁽²⁾	3.887E-04
I-127 ⁽¹⁾	7.454E-11	Rh-103m	5.778E-04		
I-129	4.427E-08	Ru-106	1.609E-04		
I-130	2.306E-02	Rh-106 ⁽²⁾	1.150E-04		
I-131	2.871E+00	Ag-110m	9.890E-04		
I-132	3.065E+00	Te-125m	3.200E-04		
I-133	4.596E+00	Te-127m	3.355E-03		
I-134	6.654E-01	Te-127	1.189E-02		
I-135	2.533E+00	Te-129m	1.162E-02		
Cs-134	1.779E+00	Te-129	1.316E-02		
Cs-136	2.014E+00	Te-131m	3.494E-02		
Cs-137	1.101E+00	Te-131	1.792E-02		
Cs-138	1.082E+00	Te-132	3.070E-01		
Mn-54	3.900E-04	Te-134	3.357E-02		
Notes:					
1. Grams of I-127 per gram of coolant.					
2. Ba-137m, Rh-106, and Pr-144 are in equilibrium with parent isotopes and are not calculated separately.					

2.9.2 Radiological Consequences of Main Steam Line Failures Outside Containment

2.9.2.1 Regulatory Evaluation

The analysis of the radiological consequences of a main steam line break (MSLB) outside containment included:

- The sequence of events, models, and assumptions used for the calculation of the radiological doses
- Evaluation of the Technical Specifications on the primary and secondary coolant iodine activities
- Determination for reactor coolant iodine concentration corresponding to a pre-accident iodine spike and a concurrent iodine spike

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of a MSLB outside containment are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA), as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provision are made for Control Room air to be recirculated through

high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

Assumptions used for the MSLB analysis are described in FSAR Section 15.1.5.3. NRC approval of the CPNPP licensing basis MSLB analysis is documented in Reference 1.

2.9.2.2 Technical Evaluation

Description of Event

The complete severance of a main steam line outside containment is assumed to occur. The affected steam generator will rapidly depressurize and release radioiodines initially contained in the secondary coolant to the outside atmosphere. Primary coolant activity transferred to the affected steam generator via tube leaks will be released to the outside atmosphere as well. Releases from the intact steam generators will be terminated when the residual heat removal system (RHRS) is placed in service. Releases from reactor coolant system (RCS) leakage into the faulted steam generator will be terminated when the RCS cools to below the boiling temperature of water at atmospheric pressure. The steam line break outside containment will bound any break inside containment since the outside break provides a means for direct release into the environment.

Analysis Assumptions and Parameters

The MSLB analysis was performed using the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect the stretch power uprate (SPU) conditions. Specific changes include:

- Revised source terms were used that reflect the core power uprate to 3,612 MWt.
- The steam generator and RCS masses have been updated.
- Steam releases are recalculated as described in Licensing Report (LR) subsection 2.9.10.
- The time required to cool the RCS below 212°F, terminating releases from the faulted steam generator has been increased to 25.75 hours. This was confirmed to be a bounding assumption for the SPU.

Results

The radiological consequences of the MSLB accident are provided in Table 2.9.2-1 and meet all regulatory limits.

2.9.2.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of a MSLB outside containment to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of a postulated MSLB outside containment since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary meet the exposure guideline values specified in 10 CFR 100.11 (assuming a pre-accident iodine spike) and are a small fraction of the exposure guideline values in 10 CFR 100.11 for the concurrent iodine spike. It is also demonstrated that the Control Room doses meet the dose requirements of GDC-19 for DBAs.

2.9.2.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

**Table 2.9.2-1
MSLB Dose Results**

Location	Dose Type	Pre-Accident Iodine Spike		Accident-Initiated Iodine Spike	
		Dose (rem)	Acceptance Limit (rem)	Dose (rem)	Acceptance Limit (rem)
EAB	Thyroid	1.2E+00	300	1.5E+00	30
	Whole-Body	2.5E-03	25	4.9E-03	2.5
LPZ	Thyroid	6.7E-01	300	3.1E+00	30
	Whole-Body	1.1E-03	25	5.4E-03	2.5
CR	Thyroid	1.1E+00	50	3.1E+00	50
	Whole-Body	1.3E-03	5	1.5E-03	5
	β -Skin	3.6E-02	50	3.7E-02	50

2.9.3 Radiological Consequences of a Reactor Coolant Pump Locked-Rotor Accident

2.9.3.1 Regulatory Evaluation

The analysis of the radiological consequences of a reactor coolant pump locked-rotor accident included:

- Determination of a need for a radiological consequences analysis
- The sequence of events, models and assumptions used for the calculation of the radiological doses

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of a reactor coolant pump locked-rotor accident are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to:

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA) as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

Assumptions used for the RCP locked rotor analysis are described in FSAR Section 15.3.3.3. NRC approval of the CPNPP licensing basis RCP locked rotor analysis is document in Reference 1.

2.9.3.2 Technical Evaluation

Description of Event

An instantaneous seizure of a reactor coolant pump rotor is assumed to occur, which rapidly reduces flow through the affected reactor coolant loop. Fuel cladding damage may be predicted to occur as a result of this accident, releasing fission product gap inventory into the primary coolant. Due to the pressure differential between the primary and secondary systems and the assumption of steam generator tube leakage, fission products pass into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the atmospheric relief valves and/or safety valves. In addition, iodine activity is contained in the secondary coolant before the accident and some of this activity is also released to the atmosphere as a result of steaming from the steam generators following the accident to remove decay heat. Releases from the secondary side are terminated when the residual heat removal system is placed in service and is removing all decay heat.

Analysis Assumptions and Parameters

The locked rotor radiological analysis was performed using the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect the stretch power uprate (SPU) conditions. Specific changes include:

- Revised source terms were used that reflect the core power uprate to 3,612 MWt.
- The steam generator and reactor coolant system masses have been updated.
- 15 percent of fuel rods are assumed to violate the departure from nucleate boiling limit. This was confirmed to be a bounding assumption for the SPU.
- Steam releases are recalculated as described in Licensing Report (LR) subsection 2.9.10.

It is noted that the radial peaking factor of 1.65 applied in determining the activity released from the failed fuel was confirmed to remain bounding for the uprate. It was also confirmed that the Regulatory Guide (RG) 1.195, Table 2, Footnote 7 criteria for applying the gap fractions are met for failed fuel.

Results

The radiological consequences of the locked rotor accident are provided in Table 2.9.3-1 and meet all regulatory limits.

2.9.3.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of a reactor coolant pump locked-rotor accident to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of a postulated locked-rotor accident since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary are a small fraction of the exposure guideline values in 10 CFR 100.11. It is also demonstrated that the Control Room doses meet the dose requirements of GDC-19 for DBAs.

2.9.3.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

Table 2.9.3-1 Locked Rotor Dose Results			
Location	Dose Type	Dose (rem)	Acceptance Limit (rem)
EAB	Thyroid	2.7E+00	30
	Whole-Body	1.7E-01	2.5
LPZ	Thyroid	5.1E+00	30
	Whole-Body	5.8E-02	2.5
CR	Thyroid	3.8E+00	50
	Whole-Body	2.0E-01	5
	β -Skin	2.6E+00	50



2.9.4 Radiological Consequences of a Control Rod Ejection Accident

2.9.4.1 Regulatory Evaluation

Luminant Power performed an analysis of the radiological consequences of a control rod ejection accident. The analysis included the plant response to a control rod ejection accident and the calculation of radiological doses at the exclusion area boundary (EAB) and low population zone (LPZ) outer boundary, and in the Control Room due to the releases resulting from a rod ejection accident. The purpose of the analysis was to:

- Ensure the plant procedures for recovery from a rod ejection accident and the plant Technical Specifications are properly taken into account in computing the doses
- Compare the calculated doses against the appropriate guidelines

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of a control rod ejection accident outside containment are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA), as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

Assumptions used for the control rod ejection analysis are described in FSAR Section 15.4.8.3. NRC approval of the CPNPP licensing basis control rod ejection accident analysis is documented in Reference 1.

2.9.4.2 Technical Evaluation

Description of Event

It is assumed that a mechanical failure of a control rod drive mechanism pressure housing has occurred, resulting in the ejection of a rod control cluster assembly and drive shaft. As a result of the accident, some fuel cladding damage and a small amount of fuel melt are assumed to occur.

Separate calculations are performed to calculate the doses resulting from the release of activity to containment and subsequent leakage to the environment and the doses resulting from the leakage of activity to the secondary system and subsequent release to the environment.

Analysis Assumptions and Parameters

The rod ejection radiological consequences analysis was performed using the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect the stretch power uprate (SPU) conditions. Specific changes include:

- Revised source terms were used that reflect the core power uprate to 3,612 MWt.
- The steam generator and reactor coolant system masses have been updated.
- 15 percent of fuel rods are assumed to violate the departure from nucleate boiling (DNB) limit.
- Steam releases are recalculated as described in Licensing Report (LR) subsection 2.9.10.
- Consistent with the rods-in-DNB assumption, 0.375 percent of the fuel in the core is assumed to melt.

It is noted that the core peaking factor of 1.65 applied in determining the activity released from the failed fuel was confirmed to remain bounding for the uprate.

Results

The radiological consequences of the rod ejection accident are provided in Table 2.9.4-1 and meet all regulatory limits..

2.9.4.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of a control rod ejection accident to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of a postulated control rod ejection accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values in 10 CFR 100.11. It is also demonstrated that the Control Room doses meet the dose requirements of GDC-19 for DBAs.

2.9.4.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

Table 2.9.4-1 Control Rod Ejection Accident Dose Results				
Location	Dose Type	Dose (rem)		Acceptance Limit (rem)
		Containment Release	Secondary Release	
EAB	Thyroid	2.0E+01	4.2E+00	75
	Whole-Body	8.2E-02	4.2E-01	6.3
LPZ	Thyroid	2.8E+01	7.9E+00	75
	Whole-Body	4.3E-02	1.5E-01	6.3
CR	Thyroid	2.3E+01	5.8E+00	50
	Whole-Body	2.6E-02	5.0E-01	5
	β -Skin	4.1E-01	6.3E+00	50

2.9.5 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

2.9.5.1 Regulatory Evaluation

Luminant Power performed an analysis of the radiological consequences of failures of small lines outside the containment which are connected to the primary coolant pressure boundary (such as instrument lines and sample lines). The analysis included:

- The identification of small lines postulated to fail and the isolation provisions for these lines
- The failure scenario
- The models and assumptions for the calculation of the radiological doses for the postulated failure
- An evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the Technical Specifications for the reactor coolant iodine activity

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of the failure of small lines carrying primary coolant outside containment are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident
- GDC-55, insofar as it establishes isolation requirements for small-diameter lines connected to the primary system that form the basis of meeting 10 CFR 100.11

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA) as detailed

in FSAR Sections 12.1 and 15.6. The Control Room air-conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5 and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

- GDC-55, Reactor Coolant Pressure Boundary Penetrating Containment, is described in FSAR Section 3.1.5.6.

Each line that is part of the reactor coolant pressure boundary (RCPB) and which penetrates the containment is provided with isolation valves meeting this criterion. For a detailed description of the containment isolation system, see FSAR Section 6.2.4. Instrument lines are designed in accordance with the requirements of NRC Regulatory Guide 1.11, Instrument Lines Penetrating Primary Reactor Coolant, as described in FSAR Section 6.2.4.1.4.

Assumptions used for the letdown line break analysis are described in FSAR Section 15.6.2. NRC approval of the CPNPP licensing basis analysis is document in Reference 1.

2.9.5.2 Technical Evaluation

Description of Event

This event assumes a complete severance of a chemical and volume control system (CVCS) letdown line just outside containment, between the outboard letdown isolation valve and letdown heat exchanger, at full rated power conditions. The severance of the letdown line results in a loss of reactor coolant. The loss of coolant is within the makeup capacity of any two of the three charging pumps, as in the CPNPP current licensing basis. Due to the elevated temperature of the primary coolant being spilled, a portion of the coolant flashes to steam and the iodine in the flashed water is assumed to become airborne and is released to the atmosphere. Also, all noble gases contained in the spilling primary coolant are released to the atmosphere. A low pressure signal will activate an alarm, alerting the operator of the rupture, resulting in manual isolation of the system. Releases from the broken CVCS letdown line are terminated when the isolation valve is closed.

Analysis Assumptions and Parameters

The radiological consequences analysis for the letdown line break was performed in a manner consistent with the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect the stretch power uprate (SPU) conditions. Specific changes include:

- Revised source terms were used that reflect the core power uprate to 3,612 MWt.
- The reactor coolant system mass has been updated.

Results

The radiological consequences of the CVCS letdown line break outside containment are provided in Table 2.9.5-1 and meet all regulatory limits.

2.9.5.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of a failure in a line carrying primary coolant outside the containment (the CVCS letdown line) to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of a postulated failure of a small line carrying primary coolant outside the containment since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary are a small fraction of the exposure guideline values in 10 CFR 100.11. It is also demonstrated that the Control Room doses meet the dose requirements of GDC-19 for DBAs.

2.9.5.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS),' August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

Table 2.9.5-1			
Letdown Line Break Dose Results			
Location	Dose Type	Dose (rem)	Acceptance Limit (rem)
EAB	Thyroid	6.2E+00	30
	Whole-Body	5.0E-02	2.5
LPZ	Thyroid	1.0E+00	30
	Whole-Body	7.0E-03	2.5
CR	Thyroid	7.0E-01	50
	Whole-Body	2.0E-02	5
	β -Skin	5.0E-01	50

2.9.6 Radiological Consequences of Steam Generator Tube Rupture

2.9.6.1 Regulatory Evaluation

The analysis of the radiological consequences of a postulated steam generator tube rupture (SGTR) includes:

- A review of the sequence of events and plant procedures for recovery from the accident to ensure that the most severe case of radioactive releases has been considered
- A review of the models and assumptions for the calculation of the radiological doses for the postulated accident
- An evaluation of the Technical Specifications on the primary and secondary coolant iodine activity concentration
- An evaluation of the radiological consequences of an SGTR concurrent with a loss of offsite power and the most limiting single failure

The analysis included two cases for the reactor coolant iodine concentration, one corresponding to a preaccident iodine spike and the other to a concurrent spike.

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of an SGTR are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to:

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is

provided for the Control Room in the event of a design basis accident (DBA) as detailed in FSAR sections 12.1 and 15.6. The Control Room air-conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provision are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

Assumptions used for the main SGTR analysis are described in FSAR Section 15.6.3.2. NRC approval of the CPNPP licensing basis analysis is document in Reference 1

2.9.6.2 Technical Evaluation

Description of Event

The complete severance of a single steam generator tube is assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system as break flow to the ruptured steam generator and as tube leakage to the intact steam generators. A portion of this radioactivity is released to the outside atmosphere through the main steam condenser, the atmospheric relief valves, or the main steam safety valves. In addition, iodine activity contained in the secondary coolant prior to the accident is released to the atmosphere as a result of steaming from the steam generators following the accident.

Analysis Assumptions and Parameters

The SGTR analysis was performed using the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect the stretch power uprate (SPU) conditions. Specific changes include:

- Revised source terms were used that reflect the core power uprate to 3,612 MWt.
- The steam generator and reactor coolant system masses have been updated.
- Break flow and steam releases are recalculated as described in Licensing Report (LR) subsection 2.8.5.6.2.

Results

The radiological consequences of the SGTR accident are provided in Table 2.9.6-1 and meet all regulatory requirements.

2.9.6.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of an SGTR to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the plant site and the dose-mitigating engineered safety features remain acceptable with respect to the radiological consequences of a postulated SGTR since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary meet the exposure guideline values specified in 10 CFR 100.11 (assuming a pre-accident iodine spike) and are a small fraction of the exposure guideline values in 10 CFR 100.11 for the concurrent iodine spike. It is also demonstrated that the Control Room doses meet the dose requirements of GDC-19 for DBAs.

2.9.6.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS),' August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

**Table 2.9.6-1
SGTR Dose Results**

Location	Dose Type	Pre-Accident Iodine Spike		Accident-Initiated Iodine Spike	
		Dose (rem)	Acceptance Limit (rem)	Dose (rem)	Acceptance Limit (rem)
EAB	Thyroid	4.0E+01	300	2.6E+01	30
	Whole-Body	1.4E-01	25	1.8E-01	2.5
LPZ	Thyroid	6.0E+00	300	4.0E+00	30
	Whole-Body	3.0E-02	25	3.0E-02	2.5
CR	Thyroid	1.7E+01	50	3.6E+00	50
	Whole-Body	1.0E-01	5	1.0E-01	5
	β -Skin	2.4E+00	50	2.4E+00	50

2.9.7 Radiological Consequences of a Design Basis Loss-of-Coolant Accident

2.9.7.1 Regulatory Evaluation

Luminant Power performed an analysis of the radiological consequences of a design basis loss-of-accident (LOCA). The analysis considered a summary review of the doses from the hypothetical design basis LOCA and a specific review of the doses from containment leakage and leakage from engineered safety feature (ESF) components outside containment that contribute to the total LOCA doses. The analysis also included:

- The methodology and results of calculations of the radiological consequences resulting from containment and engineered safety feature (ESF) component leakage following a hypothetical LOCA
- An assessment of the containment with respect to the assumptions and the values of input parameters for the dose calculations

The calculations are based on the pertinent information in the Final Safety Analysis Report (FSAR) and considered the evaluation of dose-mitigating ESFs.

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of a design basis LOCA are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to:

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA) as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features

redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

Assumptions used for design basis LOCA analysis are described in FSAR Section 15.6.5.4. NRC approval of the CPNPP licensing basis analysis is document in Reference 1.

2.9.7.2 Technical Evaluation

Description of Event

An abrupt failure of the main reactor coolant pipe is assumed to occur. Activity from the reactor coolant system (RCS) is released to containment and a portion of this activity is released to the atmosphere via the pressure relief line prior to containment isolation. It is assumed that the emergency core cooling features fail to prevent the core from experiencing significant degradation (melting). This sequence cannot occur unless there are multiple failures and thus goes beyond the typical design basis accident that considers a single active failure. Activity from the core is released to the containment and from there is released to the environment by means of containment leakage. In addition, once recirculation of the emergency core cooling system is established, iodine activity in the sump solution may be released to the environment by means of leakage from ESF equipment outside containment in the Auxiliary Building.

Analysis Assumptions and Parameters

The LOCA radiological consequences analysis was performed using the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect the stretch power uprate (SPU) conditions. Specific changes include:

- Revised source terms were used that reflect the core power uprate to 3,612 MWt.
- The RCS mass has been updated.
- Updated modeling was used to calculate the whole body dose to Control Room operators from external sources which considers the activity that remains in containment, the cloud of activity outside the Control Room, streaming of the cloud of activity outside containment, and the activity accumulated on the Control Room filters.

Results

The radiological consequences of the LOCA are provided in Table 2.9.7-1 and meet all regulatory requirements.

2.9.7.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of a design basis LOCA to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated design-basis LOCA since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary meet the exposure guideline values specified in 10 CFR 100.11 and the calculated doses in the Control Room meet the requirements of GDC-19.

2.9.7.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

**Table 2.9.7-1
LOCA Dose Results**

Location	Dose Type	Dose (rem)	Acceptance Limit (rem)
EAB	Thyroid	5.9E+01	300
	Whole-Body	7.0E-01	25
LPZ	Thyroid	4.4E+01	300
	Whole-Body	2.5E-01	25
CR	Thyroid	4.0E+01	50
	Whole-Body	1.2E+00	5
	β -Skin	1.3E+01	50

2.9.8 Radiological Consequences of Fuel Handling Accident

2.9.8.1 Regulatory Evaluation

Luminant Power performed an analysis of the radiological consequences of a postulated fuel handling accident (FHA). The analysis evaluated the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The analysis included review of:

- Sequence of events, models, and assumption used for the calculation of radiological doses
- The adequacy of the engineered safety features (ESFs) provided for the purpose of mitigating potential accident doses
- The containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of the FHA are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident
- GDC 61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to:

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA) as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

- GDC-61, Fuel Storage and Handling and Radioactivity Control, is described in FSAR Section 3.1.6.2.

The spent fuel pool, spent fuel pool cooling and cleanup system, fuel storage and handling system, radioactive waste processing systems, and other systems that contain radioactivity are designed as follows to ensure adequate safety under normal and postulated accident conditions:

1. Components are designed and located so that appropriate periodic inspection and testing can be performed.
2. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in FSAR Section 12.3
3. Individual components that contain significant radioactivity are located in confined areas that are adequately ventilated through appropriate filtering systems. Radioactive waste management is discussed in detail in FSAR Chapter 11.
4. The spent fuel pool cooling and cleanup system provides cooling to remove residual decay heat from the fuel stored in the spent fuel pool and is designed with redundancy and testability to ensure contained heat removal. A purification loop is provided to remove fission product activity. The spent fuel pool cooling and cleanup system is described in FSAR Section 9.1.
5. The spent fuel pool is designed so that no postulated accident can cause excessive loss-of-coolant inventory.
6. The primary plant ventilation system is designed to filter the exhaust from the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity. The non-ESF exhaust units that satisfy GDC-61 are described in FSAR Section 9.4.

The piping connected to the fuel pool is designed so that a significant loss of fuel pool water does not occur because of a pipe rupture. Level instrumentation indicates a reduction in fuel pool water level, and redundant sources of fuel pool water are available.

Assumptions used for the FHA analysis are described in FSAR Section 15.7.4. NRC approval of the CPNPP licensing basis analysis is document in Reference 1.

2.9.8.2 Technical Evaluation

Description of Event

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the Auxiliary Building. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the fuel pool ventilation system.

Analysis Assumptions and Parameters

The FHA radiological consequences analysis was performed using the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect the stretch power uprate (SPU) conditions. Specific changes include:

- Revised source terms were used that reflect the core power uprate to 3,612 MWt.
- A fuel decay time of 50 hours was used for the analysis.
- The analysis conservatively assumes 10% of the rods in the damaged assembly exceed the Regulatory Guide (RG) 1.195, Table 2, Footnote 7 criteria. The gap fractions from RG 1.25 (as modified by the direction of NUREG/CR-5009) are used for the fraction of rods in the fuel assembly that are assumed to exceed the footnote 7 criteria. In the pre-SPU analysis, all damaged rods were assumed to meet the RG 1.195, Table 2, Footnote 7 criteria.

It is noted that the radial peaking factor of 1.65 applied in determining the activity released from the damaged fuel was confirmed to remain bounding for the uprate.

Results

The radiological consequences of the FHA are provided in Table 2.9.8-1 and meet all regulatory requirements.

2.9.8.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of an FHA to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the

plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated FHA since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary are well within the exposure guideline values in 10 CFR 100.11. It is also demonstrated that the Control Room doses meet the dose requirements of GDC-19 for DBAs.

2.9.8.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

Table 2.9.8-1			
Fuel Handling Accident Dose Results			
Location	Dose Type	Dose (rem)	Acceptance Limit (rem)
EAB	Thyroid	2.6E+1	75
	Whole-Body	1.4E-1	6.3
LPZ	Thyroid	3.9E+0	75
	Whole-Body	2.1E-2	6.3
CR	Thyroid	4.2E+0	50
	Whole-Body	1.8E-1	5
	β -Skin	5.2E+0	50

2.9.9 Radiological Consequences of Spent Fuel Cask Drop Accidents

2.9.9.1 Regulatory Evaluation

Luminant Power evaluated the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations.

The evaluation included:

- Determining a need for a design basis radiological analysis, and if necessary:

Sequence of events, models, and assumptions used for the calculation of radiological doses

Comparing the calculated doses to exposure guidelines to determine the acceptability of the exclusion area boundary (EAB) and low population zone (LPZ) outer boundary distances and to confirm the adequacy of the engineered safety features (ESFs) provided for the purpose of mitigating potential accident doses from spent fuel cask drop accidents, including the effects on Control Room habitability

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of the spent fuel cask drop accident are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.
- GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems.
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to:

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is

provided for the Control Room in the event of a design basis accident (DBA) as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

- GDC-61, Fuel Storage and Handling and Radioactivity Control, is described in FSAR Section 3.1.6.2.

The spent fuel, spent fuel pool cooling and cleanup system, fuel storage and handling system, radioactive waste processing systems, and other systems that contain radioactivity are designed as follows to ensure adequate safety under normal and postulated accident conditions:

1. Components are designed and located so that appropriate periodic inspection and testing can be performed.
2. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in FSAR Section 12.3
3. Individual components that contain significant radioactivity are located in confined areas that are adequately ventilated through appropriate filtering systems. Radioactive waste management is discussed in detail in FSAR Chapter 11.
4. The spent fuel pool cooling and cleanup system provides cooling to remove residual decay heat from the fuel stored in the spent fuel pool and is designed with redundancy and testability to ensure contained heat removal. A purification loop is provided to remove fission product activity. The spent fuel pool cooling and cleanup system is described in FSAR Section 9.1.
5. The spent fuel pool is designed so that no postulated accident can cause excessive loss-of-coolant inventory.
6. The primary plant ventilation system is designed to filter the exhaust from the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity. The non-ESF exhaust units that satisfy GDC-61 are described in FSAR Section 9.4.

The piping connected to the fuel pool is designed so that a significant loss of fuel pool water does not occur because of a pipe rupture. Level instrumentation indicates a reduction in fuel pool water level, and redundant sources of fuel pool water are available.

2.9.9.2 Technical Evaluation

The CPNPP Fuel Handling Building crane satisfies NUREG-0554 single-failure proof requirements and is designed to the requirements of seismic Category I. As such, it can retain the maximum design load during a safe shutdown earthquake and remain in place under all postulated seismic loadings. The crane is also provided with interlocks that prevent a fuel cask from being lifted more than 29.25 ft above floor elevation or from passing over the new fuel storage area during the spent fuel cask mode of operation. The crane loads do not pass over the spent fuel pool in any mode of operation. Based on this design approach, the radiological consequences of a spent fuel cask drop accident need not be evaluated. (See FSAR Section 15.7.5.)

2.9.9.3 Conclusion

Luminant Power reviewed the current design and licensing basis for a Cask Drop Accident as described in FSAR Section 15.7.5. SPU does not impact the design approach outlined in FSAR Section 15.7.5, therefore the radiological consequences of a spent fuel cask drop accident need not be further evaluated.

2.9.10 Steam Releases from Intact Steam Generators for Locked Rotor and MSLB Radiological Dose Analyses

2.9.10.1 Regulatory Evaluation

Radioactive steam releases to the environment are postulated to occur in support of radiological dose analyses. It is assumed that an activity level exists in the reactor coolant system (RCS). The activity level in the RCS may be low, resulting from activated corrosion products or from the potential minute release of fission material from defective fuel assemblies. The activity level may also be moderate to high, resulting from potential fuel cladding failures and the subsequent fission product release. A primary-to-secondary leakage rate is also assumed, based on the Technical Specifications leakage limit. It is further postulated that the condenser is not available for steam dump so that steam and radioactivity will be released through the atmospheric relief valves while the plant is being brought to a cold shutdown condition after reactor trip.

Vented steam releases have been calculated for the locked rotor and steam line break events to support the uprated nuclear steam supply system (NSSS) power of 3,628 MWt.

The locked rotor and steam line break events are Condition IV events as defined by Reference 1, and the offsite dose calculations for Conditions IV events must meet the requirements of 10 CFR Part 100 (Reference 2). As previously stated, the steam releases for dose calculated for the locked rotor and steam line break events are used as input to the radiological dose consequences analysis. The steam line break event steam release for dose analysis assumptions for the current Comanche Peak Nuclear Power Plant (CPNPP) dose licensing basis are provided in Reference 3.

2.9.10.2 Technical Evaluation

Input Parameters, Assumptions, and Acceptance Criteria

The following general assumptions have been used in the calculation of the steam releases. They are applicable to CPNPP Unit 1 (with Model Δ 76 steam generators) and Unit 2 (with Model D-5 steam generators).

- NSSS power of 3,628 MWt plus 0.6-percent uncertainty
- RCS average temperature of 589.2°F plus 5.96°F uncertainty
- Nominal RCS pressure of 2,250 psia
- Initial full-power steam temperature of 548.4°F for Unit 1 and 546.6°F for Unit 2
- No steam generator tube plugging or fouling
- Residual heat removal initiation conditions of 350°F and 350 psig (which is conservative with respect to the expected higher initiation pressure)

-
- Steam releases from the unaffected/intact steam generators are determined for the intervals 0-to-2 hours and 2-to-11 hours. By the end of the 2-hour time frame (and extending until the time of the residual heat removal (RHR) initiation), the RCS and intact steam generators are assumed to be in a steady-state condition.

Steam release will be required until the RHR system (RHRS) is placed in-service and removing all decay and sensible heat. It has been conservatively assumed that 11 hours of steam release could occur prior to placing the plant in the RHR mode of operation. After the first 2 hours, it is assumed the plant will have cooled down and stabilized at no-load conditions. The additional 9 hours are conservatively longer than the actual time required to cool down and depressurize the plant from no-load conditions to the RHR operating conditions.

Acceptance Criteria

There are no specific acceptance criteria associated with the calculation of the steam releases used as input to the radiological dose analysis. Tables of vented steam releases for the two cooldown intervals, for each of these two transients, are used as input to the radiological dose analysis in support of the SPU project.

Description of Analyses and Evaluation

The amount of steam released to the atmosphere depends on the sensible heat and decay heat generated while reducing the RCS temperature from the full-power value to the shutdown conditions. No computer program is used for this analysis. A hand calculation is performed to determine the amount of the vented steam release to the atmosphere.

The calculation is an energy balance that determines the amount of heat that would be dissipated via steam release through the steam generator safety valves or atmospheric relief valves. The energy balance considers the heat generated in the core, heat released or absorbed by thick metal in the RCS and the intact steam generators, and heat released or absorbed within the fluids in the RCS and the intact steam generators. The energy that cannot be stored within the defined boundary of the RCS and intact steam generators is removed via steaming, and the purpose of the calculation is to determine the steam mass released. The decay heat model predates the American Nuclear Society (ANS) standard. It includes decay heat and residual fissions based on one-percent shutdown margin. The decay heat values used in this calculation are more conservative than the 1979 ANS + 2σ model. For the steam line break event, only 3 steam generators are considered to be intact, while the steam releases for the locked rotor event consider all 4 steam generators as intact.

Results

Table 2.9.10-1 summarizes the vented steam releases from the intact-loop steam generators for the 0-to-2-hour time period and the 2-to-11-hour time period. These time intervals are evaluated for the locked rotor and steam line break events in support of the CPNPP stretch power uprate (SPU). For the steam line break event, the steam release from the faulted steam generator is separately addressed in the dose analysis and is not included in the data presented in Table 2.9.10-1 (See Licensing Report (LR) Section 2.9). The steam releases provided in Table 2.9.10-1 for the steam line break and locked rotor events are used as input to the radiological dose analysis.

2.9.10.3 Conclusions

The vented steam releases from the intact steam generators have been calculated at the uprated NSSS power level of 3,628 MWt for the locked rotor and steam line break events. The assumptions delineated in LR subsection 2.9.10.2 have been included in the steam release calculations for each transient, such that the results are consistent with and continue to comply with the current licensing basis and acceptance requirements associated with the radiological analysis. The steam releases discussed in this section have been provided for use in the radiological dose analysis (see LR Section 2.9) to support the CPNPP Units 1 and 2 SPU Program.

2.9.10.4 References

1. American Nuclear Society ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants."
2. U.S. Nuclear Regulatory Commission, Rules and Regulations, Title 10, Chapter 1, Code of Federal Regulations – Energy, Part 100, "Reactor Site Criteria," Section 100.11.
3. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS),'" August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Steam Electric Station (CPSES), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

Table 2.9.10-1

Steam Releases from Intact Steam Generators for Radiological Dose Results

Event	Unit 1 Mass of Steam Vented to the Environment (lbm)		Unit 2 Mass of Steam Vented to the Environment (lbm)	
	0-2 hours	2-11 hours	0-2 hours	2-11 hours
Steam Line Break	434,000	1,235,000	405,000	1,228,000
Locked Rotor	440,000	1,277,000	402,000	1,275,000

2.9.11 Radiological Consequences of Gas Decay Tank Rupture

2.9.11.1 Regulatory Evaluation

Luminant Power performed an analysis of the radiological consequences of the rupture of a gas decay tank. The analysis was conducted to verify various design and operations aspects of the system.

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of the gas decay tank rupture are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to:

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA) as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

Assumptions used for the gas decay tank rupture analysis are described in FSAR Section 15.7.1. NRC approval of the CPNPP licensing basis analysis is document in Reference 1.

2.9.11.2 Technical Evaluation

Description of Event

This event assumes a failure that results in the release of the contents of one gas decay tank.

Analysis Assumptions and Parameters

The gas decay tank rupture radiological consequences analysis was performed using the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect an increase in core power. Specific changes include revised source terms that reflect the core stretch power uprate (SPU) to 3,612 MWt.

Results

The radiological consequences of the gas decay tank rupture are provided in Table 2.9.11-1 and meet all regulatory requirements.

2.9.11.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of the rupture of a single gas decay tank to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated gas decay tank rupture since the calculated whole-body doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary are substantially below the exposure guideline values in 10 CFR 100.11. It is also demonstrated that the Control Room doses meet the dose requirements of GDC-19 for DBAs.

2.9.11.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

Table 2.9.11-1			
Gas Decay Tank Rupture Doses			
Location	Dose Type	Dose (rem)	Acceptance Limit (rem)
EAB	Whole-Body	1.9E-01	0.5
LPZ	Whole-Body	2.8E-02	0.5
CR	Whole-Body	3.0E-01	5
	β -Skin	3.2E+01	50

2.9.12 Radiological Consequences of Liquid Waste Tank Rupture

2.9.12.1 Regulatory Evaluation

Luminant Power performed an analysis of the radiological consequences of a rupture of a liquid waste tank. The analysis was conducted to verify various design and operations aspects of the system.

The Nuclear Regulatory Commission (NRC) acceptance criteria for the radiological consequences of the liquid waste tank rupture are based on:

- General Design Criterion (GDC)-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.
- 10 CFR Part 100, insofar as it establishes requirements for assuring that offsite radiological doses from postulated accidents will be acceptably low.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to:

- GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA) as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

Assumptions used for the rupture of a liquid waste tank analysis are described in FSAR Section 15.7.2. NRC approval of the CPNPP licensing basis analysis is document in Reference 1.

2.9.12.2 Technical Evaluation

Description of Event

This event assumes an uncontrolled atmospheric release from the 30,000 gallon floor drain tank due to the postulated rupture of the tank. This tank has the highest potential atmospheric release source term because of its large volume and the fact that it is assumed to be 80-percent full of reactor coolant.

Analysis Assumptions and Parameters

The liquid waste tank rupture radiological consequences analysis was performed using the analytical methods and assumptions presented in the current licensing basis analysis, Reference 1, with appropriate changes to reflect the stretch power uprate (SPU) conditions. Specific changes include revised source terms that reflect the core power uprate to 3,612 MWt.

Results

The radiological consequences of the liquid waste tank rupture are provided in Table 2.9.12-1 and meet all regulatory requirements.

2.9.12.3 Conclusion

Luminant Power has reanalyzed the radiological consequences of the rupture of a liquid waste tank to account for the effects of the proposed SPU on the analysis. The results of the analysis demonstrate that the plant site and the dose-mitigating engineered safety features remain acceptable with respect to the radiological consequences of a postulated liquid waste tank rupture since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary are substantially below the exposure guideline values in 10 CFR 100.11. It is also demonstrated that the Control Room doses meet the dose requirements of GDC-19 for DBAs.

2.9.12.4 Reference

1. TXX-05127, "License Amendment Request (LAR) 05-005 Revision to Technical Specification 3.7.10, 'Control Room Emergency Filtration/Pressurization System (CREFS)'," August 22, 2005, approved by NRC letter dated February 20, 2007, "Comanche Peak Steam Electric Station (CPSES), Units 1 and 2 – Issuance of Amendments RE: New Methods and Assumptions for Radiological Consequences Calculations (TAC Nos. MC8163 and MC8164)," (Available in NRC ADAMS database: Accession No. ML070310476).

Table 2.9.12-1

Liquid Waste Tank Rupture Dose Results

Location	Dose Type	Dose (rem)	Acceptance Limit (rem)
EAB	Thyroid	2.1E+00	6
	Whole-Body	3.8E-03	0.5
LPZ	Thyroid	3.2E-01	6
	Whole-Body	5.6E-04	0.5
CR	Thyroid	3.8E+01	50
	Whole-Body	3.5E-03	5
	β -Skin	2.2E-02	50

2.10 HEALTH PHYSICS

2.10.1 Occupational and Public Radiation Doses

2.10.1.1 Regulatory Evaluation

Luminant Power conducted its review in this area to ascertain the overall effects the proposed stretch power uprate (SPU) will have on both occupational and public radiation doses, and to determine that the Comanche Peak Nuclear Power Plant (CPNPP) has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable (ALARA). The Luminant Power review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones and plant area accessibility. Luminant Power evaluated how personnel doses needed to access plant vital areas following an accident are affected. Luminant Power considered the effects of the proposed SPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary.

The acceptance criteria for occupational and public radiation doses are based on 10 CFR 20 and General Design Criterion (GDC)-19.

Current Licensing Basis

As noted in Final Safety Analysis Report (FSAR) Section 3.1, the GDC used during the licensing of the Comanche Peak Nuclear Power Plant (CPNPP) Units are compared against Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, GDC for Nuclear Power Plants. The adequacy of the CPNPP design relative to the GDC is discussed in FSAR Section 3.1.

Specifically, the adequacy of CPNPP design relative to conformance to GDC-19, Control Room, is described in FSAR Section 3.1.2.10.

- Safe occupancy of the Control Room during abnormal conditions is provided for in the design.
- The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.
- In the unlikely event that access to the Control Room is restricted, the hot shutdown panel, local control stations, or manual operation of critical components can be used to effect hot shutdown from outside the Control Room for an extended period.
- Before evacuation takes place, the reactor can be manually tripped and neutron level and control rod position can be verified. By use of appropriate procedures and equipment, the unit can also be brought to cold shutdown conditions (see FSAR Section 7.4).

Licensing Report (LR) Section 2.9, Source Terms and Radiological Consequence Analyses, summarizes the SPU assessment of impact on post-accident dose consequences at the site boundary and at locations onsite that require continuous occupancy, such as the Control Room.

Additional details that define the CPNPP licensing basis with respect to radiation protection of plant personnel and the public are described in various FSAR sections as described below:

- FSAR Section 12.3.2, Shielding, discusses the radiation shielding design and CPNPP commitment to 10 CFR 20.
- FSAR Section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems; and Section 12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation, discuss the radiation monitoring system (RMS) design and describe how the RMS assists in the protection of the general public and plant personnel from exposure to radiation or radioactive materials in excess of those allowed by 10 CFR 20, and in limiting the radiation levels to as low as reasonably achievable in accordance with 10 CFR 50.
- FSAR Section TMI - II.B.2, Plant Shielding to provide Access to Vital Areas and Protect Safety Equipment for Post accident Operation, discusses post-accident access to vital areas, and CPNPP compliance with NUREG-0737.
- FSAR Section 11.2, Liquid Waste Management System, Section 11.3, Gaseous Waste Management Systems; and Section 11.4, Solid Waste Management System, discuss radioactivity in effluents resulting from operation of the liquid, gaseous, and solid waste management systems, respectively, and CPNPP compliance with 10 CFR 20 and 10 CFR 50, Appendix I.
- FSAR Section 12.1, Ensuring that Occupational Radiation Exposures are As-Low-As Reasonably Achievable (ALARA), and Section 12.5, Radiation Protection, discuss CPNPP policy to implement a program that meets the requirements of 10 CFR 20 and ensure that the occupational radiation exposures are kept ALARA.

CPNPP Technical Specification 5.6.3, "Radioactive Effluent Release Report," requires that an annual report be submitted in accordance with the requirements of 10 CFR 50.36a and consistent with the objectives outlined in the Offsite Dose Calculation Manual and the requirements of 10 CFR 50 Appendix I.

2.10.1.2 Technical Evaluation

The technical evaluation is presented in five subsections as listed below:

- Normal operation radiation levels and shielding adequacy
- Radiation monitoring setpoints
- Post-accident vital area accessibility

-
- Normal operation radwaste effluents and annual dose to the public
 - Ensuring that occupational and public radiation exposures are ALARA

2.10.1.2.1 Normal Operation Radiation Levels and Shielding Adequacy

Introduction

Cubicle wall thickness is specified not only for structural and separation requirements, but also to provide radiation shielding in support of radiological equipment qualification, and to reduce operator exposure during all modes of plant operation, including maintenance and accidents.

Conservative estimates of the radiation sources in plant systems and personnel access requirements form the bases of normal operation plant shielding and radiation zoning. These radiation source terms are primarily derived from conservative estimates of the reactor core and reactor coolant (also called primary coolant) isotopic activity inventory, and are referred to as design basis source terms. The SPU will impact the isotopic activity inventory in the core. In addition, since the design basis reactor coolant source term is based on 1-percent fuel defects, the SPU will result in an increase in the design basis reactor coolant activity concentration.

The expected radiation source terms in the coolant will also be impacted by the SPU. Expected source terms are less than that allowable by the plant Technical Specifications and are significantly less than the design basis source terms.

The impact of the SPU on the normal operation dose rates and the adequacy of existing shielding are evaluated to ensure safe operation within applicable regulatory limits. The assessment is broken into two parts – the impact of the SPU on 1) plant radiation levels during normal operation, and 2) adequacy of existing shielding for normal plant operation.

The shielding design basis for CPNPP is summarized in FSAR Section 12.3.2, with the radiation source terms summarized in FSAR Section 11.1.3. The original plant shielding design was based on a core power level of 3,565 MWt and a 1-year fuel cycle length. Currently, each cycle specific core design is evaluated for radiation impact to ensure the original design calculations supporting plant shielding remain adequate for current plant operations.

The SPU analysis is based on a conservative core power level of 3,684 MWt and an 18-month fuel cycle. An increase of fuel cycle length will increase the inventory of long-lived isotopes in the core and in the reactor coolant. The activity inventory of a few isotopes that are produced primarily by neutron activation of stable or long-lived fission products will also increase due to longer accumulation time.

The SPU results in an increase of the nuclear fission rate and consequently, an increase of neutron flux and the fission product generation rate. This leads to an increase of the fission product inventory in the core and spent fuel, and an increase of neutron and gamma flux leaking out of the reactor vessel.

The increase in the neutron flux results in an increase of neutron activation products in the reactor cooling system, control rod assemblies, reactor internals, and in the pressure vessel. The increase in the core inventory of fission products and actinides due to the SPU will also increase the activity concentrations in the reactor coolant due to fuel defects.

In the unlikely event that CPNPP experiences primary-to-secondary leakage, the activity concentrations in the secondary system will also increase due to primary-to-secondary leakage in the steam generators. The radiation source in the downstream systems will undergo a corresponding increase. This increase in the radioactivity levels, and the associated increase in the radiation source strength, results in an increase of radiation levels in the Containment Building, Safeguards Building, Auxiliary Building, Fuel Handling Building, Electrical and Control Buildings, and other locations, including offsite, that are subject to direct shine from radiation sources contained in these buildings.

Description of Analyses and Evaluations

The SPU evaluation utilizes scaling techniques to determine the impact of the SPU on plant radiation levels. This evaluation takes credit for conservatism in existing shielding analyses and the site ALARA Program to demonstrate adequacy of current plant shielding to support compliance with the operator exposure limits of 10 CFR 20.

1. Normal Operation Radiation Levels

For the same source-shield-detector configuration, the dose rate at a given detector point is directly proportional to the radiation source strength in the source region. The impact of increasing the reactor power from the current licensed level of 3,458 MWt to the conservatively analyzed core power level of 3,684 MWt on the neutron flux and gamma flux in and around the core, fission product and actinide activity inventory in the core and spent fuels, N-16 source in the reactor coolant, neutron activation source in the vicinity of the reactor core, and fission/corrosion products activity in the reactor coolant and downstream systems, was examined, and the increase quantified. This flux or activity increase factor for a given radiation source was determined to be the SPU scaling factor for the estimated dose rate due to that source.

The SPU assessment with regard to normal operation radiation levels is divided into four areas:

a. Areas Near the Reactor Vessel

During normal operation, the radiation source in the reactor core is made up of neutron and gamma fluxes that are approximately proportional to the core power level. The radiation sources during shutdown are the gamma fluxes in the core and the activation activities in the reactor internals, pressure vessel, and primary system piping walls, which also vary approximately in proportion to the reactor power.

The radiation dose rate near the reactor vessel is determined by the leakage flux from the reactor vessel. Therefore, an uprate from the current licensed core power of 3,458 MWt to an analyzed core power of 3,684 MWt is estimated to increase the normal operation radiation levels in areas near the reactor vessel by a factor of approximately 1.065, (that is, $3,684/3,458$).

b. In-Containment Areas Adjacent to the Reactor Coolant System

During normal operation, the major radiation source in the RCS components located within containment is N-16. With the core power increase from 3,458 MWt to the analyzed core power of 3,684 MWt, the fast neutron flux is estimated to increase by approximately 6.5 percent. The coolant residence time in the core and the transit time are not expected to change significantly due to the uprate. Therefore, the SPU scaling factor for the areas subjected to the N-16 source is 1.065.

The deposited corrosion product activity depends on the reactor coolant chemistry and the cobalt impurity in RCS and steam generator components. Since the water chemistry remains approximately the same, and the SPU will increase the neutron flux by approximately 6.5 percent, the corrosion product activity deposits and the associated shutdown dose rate are also estimated to increase by 6.5 percent.

c. Areas Near Irradiated Fuels and Other Irradiated Objects

These areas include the refueling canal, spent fuel pool, in-core instrumentation drive assembly area, and other areas housing neutron irradiated materials. The radiation source is the gamma rays from the fission products and activation products, which are determined by the fission rate, neutron flux level, and the irradiation time.

Since both the fission products and the activation products are estimated to increase by approximately 6.5 percent for a core power increase from 3,458 MWt to the analyzed core power level of 3,684 MWt, the SPU scaling factor for the areas subjected to irradiated fuels and other irradiated sources is 1.065.

d. Areas Outside Containment Where the Radiation Source is Derived from the Primary Coolant Activity

In most areas outside the reactor containment, the radiation sources are either the primary coolant itself or down-stream sources originating from the primary coolant activity. Following the SPU, both the fission products and the activated corrosion products in the primary coolant, and thus the down-stream sources, are estimated to increase by approximately 6.5 percent for a core power increase from 3,458 MWt to the analyzed power level of 3,684 MWt.

The SPU scaling factor for the areas outside containment where the radiation source is derived from the primary coolant activity is, in general, 1.065 with the exception of the area near the condensate polishing system. The radiation level near the condensate polishing system may increase slightly greater than the percentage of the SPU due to the increased steam flow rate and moisture carryover fraction associated with the SPU.

An assessment of the impact of the SPU on the activity accumulation on the condensate polishers concludes that, due to the SPU, the long-lived non-particulate halogen isotopes in the condensate polishers, at steady state, will increase by an estimated value of ≈ 6 percent, due to the 1-percent increase of halogen concentration in the steam generator liquid/steam and 5.1-percent increase of steam flow rate (1.01×1.051). The long-lived particulate isotopes in the condensate polishers at steady state are estimated to increase by up to a factor of 5, due to a 39-percent decrease of particulate concentration in the steam generator liquid/steam, increase in the moisture carryover fraction of 7.8, and 5.1-percent increase of steam flow rate ($0.61 \times 7.8 \times 1.051$).

For condensate polishers with fresh or newly regenerated resins, the activity is dominated by halogens and the radiation level for the same polisher operation time is estimated to increase by 6 percent. As the operation time increases, the contribution by the particulates will increase and the SPU impact will increase accordingly. However, this additional increase is limited because most of the particulates are removed by blowdown, and the particulate concentrations in the steam are many orders of magnitude less than those of halogens. Based on the above, it can be concluded that the dose rates near the condensate polishers increase by approximately the same percentage as the SPU or slightly higher.

2. Plant Shielding Adequacy

Shielding is used to reduce radiation dose rates in various parts of the station to acceptable levels consistent with operational and maintenance requirements and to maintain the dose rates at the site boundary to below those allowed for continuous non-occupational exposure.

The original CPNPP shielding design was based on plant operation at a core power level of 3,565 MWt/12-month fuel cycle, upon generalized occupancy requirements in various radiation zones of the station, and upon conservative reactor coolant source terms assuming 1-percent fuel defects.

The SPU evaluation takes into consideration that the occupancy requirements are not affected by the SPU. Similarly, the layout/configuration of systems containing radioactivity are unchanged by the SPU. Consequently, the SPU evaluation focused on determining a SPU scaling factor based on the design basis fission and corrosion product activity concentrations in the reactor coolant used in the original plant shielding design as documented in FSAR Table 12.2-3 and the corresponding SPU design basis

reactor coolant activity concentrations presented in LR Table 2.10.1-1 which reflects an analyzed core power level of 3,684 MWt, an 18-month fuel cycle length, and 1-percent fuel defects.

The source terms at the analyzed power are compared to the source terms used in the original shielding design to evaluate the adequacy of the shielding design. The SPU evaluation takes into consideration a) the conservative analytical techniques used to establish plant shielding design, b) the Technical Specification limits on the reactor coolant activity concentrations, and c) the station ALARA program that minimizes the radiation exposure to plant personnel.

a. Reactor Primary Shield

FSAR Section 12.3.2.4.1 discusses the primary shield which consists of a reinforced concrete structure that surrounds the reactor vessel. The primary function of the primary shield is to attenuate the neutron and gamma fluxes leaking out of the reactor vessel. Fuel cycle length has insignificant impact on the maximum dose rates around the reactor vessel, which are based on the neutron and gamma flux during power operation.

Luminant Power reviewed the fluence calculations and confirmed that the original design calculations remain bounding for SPU conditions. With continued use of low leakage fuel management following the SPU, the existing primary shielding remains adequate, and the estimated dose rates adjacent to the reactor vessel/primary wall remain within original design.

b. Reactor Secondary Shields

FSAR Sections 12.3.2.4.2 and 12.3.2.4.3 discuss the secondary shield and containment structure surrounding the reactor coolant loops and the NSSS. The primary function of these secondary shields is to attenuate the N-16 source, which emits high-energy gammas. These shields were designed to limit the full-power dose rate outside the Containment Building to acceptable levels. The N-16 source is estimated to increase by approximately 3.3 percent (3,684/3,565). The N-16 activity level is not impacted by fuel cycle length. The impact of the estimated 3.3 percent increase in source terms is bounded by the conservative analytical techniques used to establish plant shielding design (such as ignoring the shadow shielding effect of the neighboring sources, rounding up the calculated shield thickness to a higher whole number, and using a conservative infinite medium buildup factor), and the current reactor coolant loop shielding and containment structure is determined to be adequate for safe operation following the SPU.

c. Fuel Transfer Shield

FSAR Sections 12.3.2.4.5 and 12.3.2.4.7 discuss the fuel handling shielding which provides protection during all phases of removal and storage of spent fuel and control rods.

With the analyzed core power increase from 3,565 to 3,684 MWt, the gamma source from the irradiated fuel is estimated to increase by approximately 3.3 percent. The 18-month fuel cycle will also increase the inventory of long-lived isotopes in the irradiated fuel. However, this is not a concern as the dose rates near the refueling canal and the spent fuel pool are dominated by the shorter half-life isotopes in the freshly discharged spent fuel assemblies. The impact of the estimated 3.3-percent increase in source terms used in the SPU analysis versus the original shielding analysis is bounded by the conservative analytical techniques discussed earlier, which were used to establish plant shielding design. Consequently, the current spent fuel shielding is determined adequate for safe operation following the SPU.

d. All Other Shielding Outside Containment

FSAR Section 12.3.2.4.4 discusses the shielding provided outside the containment where the radiation sources are either the reactor coolant itself or down-stream sources originating from coolant activity. A review was performed of the SPU design primary coolant source terms (fission and activation products) versus the original design basis primary coolant source terms. It is noted that the analyzed design primary coolant source terms utilized for the SPU reflect a core power level of 3,684 MWt, operation with an 18-month fuel cycle, 1-percent fuel defects, a 4-percent margin to account for potential future fuel cycle variability, and more advanced fuel burnup modeling/libraries as compared to the computer codes used in the original analyses that addressed a core power level of 3,565 MWt and a one-year fuel cycle length.

The SPU assessment concluded that the estimated increase in the dose rate for shielded configurations based on the design SPU reactor coolant versus the pre-uprate coolant is compensated by the plant Technical Specifications that will limit the SPU RCS, degassed RCS, and RCS noble gas source terms and associated dose rates to less than the original design basis values. It is, therefore, concluded that the shielding design based on the original design basis primary coolant activity remains valid for the SPU condition.

Results

The normal operation radiation levels in most of the plant areas are estimated to increase by approximately 6.5 percent, that is, the percentage increase between the current licensed power level of 3,458 MWt, and the conservatively analyzed core power level of 3,684 MWt used for the

SPU assessment. The exposure to plant personnel and to the offsite public is also estimated to increase by the same percentage.

The increase in radiation levels will not affect radiation zoning or shielding requirements in the various areas of the plant. This is because the increase is offset by the:

- Conservative analytical techniques typically used to establish shielding requirements
- Conservatism in the original design basis reactor coolant source terms used to establish the radiation zones
- Plant Technical Specification Section 3.4.16 which limits the reactor coolant concentrations to levels at or below the original design basis source terms

As indicated in FSAR Sections 12.1 and 12.5, individual worker exposures will be maintained within the regulatory limits of 10 CFR 20 for occupational exposure by the site ALARA Program that controls access to radiation areas. In addition, the Offsite Dose Calculation Manual ensures that the radiation levels at the site boundary due to direct shine from radiation sources in the plant will be maintained within the regulatory limits of 10 CFR 20 and 40 CFR 190 for continuous non-occupational exposure.

The SPU assessment also demonstrates compliance with GDC-19 with regard to radiation protection, insofar that actions can be taken in the Control Room to operate the nuclear power unit safely during normal operation.

2.10.1.2.2 Radiation Monitoring Setpoints

As discussed in FSAR Sections 11.5 and 12.3.4, the radiation monitors installed at CPNPP can be classified into four categories: 1) area, 2) airborne, 3) process, and 4) effluent.

Area and airborne radiation monitors are included as radiation protection features and provide radiation/radioactivity monitoring to support control of radiation exposure of plant personnel. Process and effluent radiation monitors are provided in support of radioactivity monitoring in gaseous or liquid process streams, or effluent release points to unrestricted areas to detect system leakage and support control of radiation exposure of both plant personnel and the public. Post-accident monitoring is provided in accordance with Regulatory Guide 1.97 requirements to give notice of significant radioactive releases from the plant. The high alarm and alert setpoints for the radiation monitors are based on meeting the above objectives.

The function of area monitor alarm setpoints is to provide an early warning of changing radiological conditions in a specified area. The function of alarm setpoints for process/effluent monitors is to indicate leakage or malfunction of equipment; or a potential for an activity release that may exceed the release rate limit. The high alarm setpoint of selected effluent monitors will also initiate interlocks that terminate activity release to the environment. The function of the post-accident radiation monitors is to give notice of significant radiation levels within plant areas or in environmental releases from the plant.

The bases of the radiation monitor setpoints are usually a regulatory commitment (that is, the definition of a high radiation zone, or radioactivity in environmental releases that are fractions or multiples of the release rate limits and are intended to give notice of releases approaching the limits of 10 CFR 20), a multiple of the background, or a "high" value indicating an unusual event (such as leakage or malfunction of systems), that leads to a sudden increase of the activity level in the monitored stream. In some cases, the setpoint values are based on background levels/process fluids source terms that reflect plant operating data and are reviewed frequently and adjusted as required. In all of the above cases, the radiation monitor setpoint bases, and the methods of setpoint determination, remain valid following the SPU.

In some cases, a setpoint value may be based on "calculated" background levels/process fluid source terms. The SPU will increase the activity level of radioactive isotopes in most streams/components and the associated radiation levels by approximately the percentage of the uprate. However, the relative isotopic compositions in the process and effluent streams are not expected to change due to the SPU. The impact of the increase in background level (4.5 percent, i.e., 3,612/3,458) due to the SPU on monitor sensitivity is expected to be minimal and monitor settings are performed per existing plant procedure.

2.10.1.2.3 Post-Accident Vital Area Accessibility

Introduction

In accordance with NUREG-0737, II.B.2, vital areas are those areas within the station that will or may require access/occupancy to support accident mitigation following a loss-of-coolant accident (LOCA). In accordance with the above regulatory document, all vital areas and access routes to vital areas must be designed such that operator exposure while performing vital access functions remain within regulatory limits.

This section focuses on areas that may require short-term, one-time, or infrequent access following a LOCA. Onsite locations that require continuous occupancy and a demonstration of 30-day habitability are addressed in LR Section 2.9, Source Terms and Radiological Consequence Analyses.

The design basis vital area access review that supports CPNPP licensing basis relative to vital area dose rates/operator doses while performing post-LOCA vital missions is documented in FSAR Section TMI - II.B.2. FSAR Table II.B.2-4 summarizes the estimated operator mission doses for the 19 target areas that were determined to require short-term occupancy. As a result of a subsequent plant modification and per License Amendment No. 117, access is no longer required to the hydrogen recombiners (Access route No. 15).

SPU will typically increase the activity level in the core by the percentage of the uprate. The radiation source terms in equipment/structures containing post-accident fluids, and the corresponding environmental radiation levels, will increase proportionately to the uprate. In addition, factors that impact the equilibrium core inventory and consequently the estimated radiation environment, are fuel enrichment and burnup. These additional changes could result

in activity levels in the core that are typically higher than the core power ratio associated with the uprate.

Description of Analyses and Evaluations

The SPU assessment is based on an analyzed core power level of 3,684 MWt and use of an 18-month fuel cycle. The methodology utilized in the SPU evaluation is to demonstrate, using scaling techniques, compliance with the operator exposure dose limits of 5 rem provided in NUREG-0737, II.B.2.

The impact of the SPU on the post-LOCA gamma radiation dose rates utilized to determine operator exposure during vital area access is evaluated by comparing the gamma source terms, based on the original core inventory utilized to develop the post-LOCA dose rates, to the gamma source terms, based on the SPU core inventory. This approach takes into consideration that: a) the post-LOCA operator mission requirements, including the task description and required time/duration for access is not impacted by the SPU, and b) SPU does not impact the operation and layout/arrangement of plant radioactive systems.

Theoretically, following the SPU, the post-LOCA environmental gamma dose rates and the operator dose per identified mission should increase by approximately 3.3 percent (3,684 MWt/3,565 MWt). However, because the SPU analyzed core reflects: a) operation with an 18-month fuel cycle, b) more advanced fuel burnup modeling/libraries than used in the original analyses, and c) a 4-percent margin to account for variability in future fuel cycle designs, the calculated SPU scaling factor values will deviate from the core power ratio.

The SPU assessment is essentially a two-step process. The first develops a bounding SPU dose rate scaling factor versus time, and the second multiplies the pre-SPU personnel dose/dose rates at target areas identified in the licensing basis by the bounding SPU scaling factor.

The pre-SPU and the SPU core inventories are utilized to develop the post-LOCA gamma energy release rates (Mev/sec) per energy group vs. time for the various post accident sources, that is, containment atmosphere inside and outside containment, sump water, pressurized recirculating fluid, iodine plateout on surfaces and accumulation on heating, ventilation, and air conditioning (HVAC) filters.

For the "unshielded" case, the factor impact on post-accident gamma dose rates is estimated by ratioing the gamma energy release rates weighted by dose rates, as a function of time, for the SPU analyzed core power level, to the corresponding weighted source terms based on the pre-SPU analyzed core power level. To address the fact that the vital access locations are outside containment, the unshielded values include the shielding effect of a pipe wall thickness associated with a 2-inch nominal diameter pipe. This ensures that the results are not skewed by photons at energies less than 25 keV, which will be substantially attenuated by any piping sources.

To evaluate the factor impact of the SPU on post-LOCA gamma dose rates (versus time) in areas that are shielded, the pre-SPU as well as the SPU source terms discussed above were weighted by the concrete reduction factors for each energy group. The concrete reduction factors for 1 and 3 feet of concrete are used to provide a basis for comparison of the post-LOCA spectrum for the various source terms, with respect to time, for lightly shielded and heavily shielded cases.

Since the SPU gamma dose rate scaling factors vary with source, time, as well as shielding, to cover all types of analysis models/assessments, the maximum dose rate scaling factor with respect to time developed from the above assessments is used for all source/receptor combinations, with or without shields, for the time period identified in the vital access assessment.

Results

Review of FSAR Section II.B.2, Table II.B.2-4, indicates that the earliest required time of access for the 19 vital access missions evaluated, range from 2 minutes to 3 days (72 hours) after a LOCA.

The current post-LOCA vital area operator dose estimates presented in FSAR Section II.B.2, Table II.B.2-4, remain valid for SPU conditions, thus demonstrating continued compliance with the regulatory limit of 5 rem whole body listed in NUREG-0737, II.B.2.

2.10.1.2.4 Normal Operation Radwaste Effluents and Annual Dose to the Public

Introduction

Liquid and gaseous effluents released to the environment during normal plant operations contain small quantities of radioactive materials.

Liquid, gaseous, and solid radwaste systems are designed such that the plant is capable of maintaining normal operation offsite gaseous and liquid releases and doses within regulatory limits. The actual performance and operation of installed equipment, as well as reporting of actual offsite releases and doses, is controlled by the requirements of the Offsite Dose Calculation Manual (ODCM).

There are no specific regulatory limits associated with generation of solid radwaste other than those associated with transportation. However, onsite storage of solid radwaste may result in increased public exposure at the site boundary which is controlled by federal regulations.

SPU will increase the activity level of radioactive isotopes in the reactor and secondary coolant and steam. Due to leakage or process operations, fractions of these fluids are transported to the liquid and gaseous radwaste systems where they are held prior to discharge. As the activity levels in the coolants and steam are increased, the activity level of radwaste inputs, and subsequent environmental releases, are proportionately increased.

Description of Analyses and Evaluations

The methodology used in the SPU evaluation is to demonstrate, using scaling techniques, compliance with the annual dose limits to an individual in an unrestricted area set by 10 CFR 20, 10 CFR 50, Appendix I and 40 CFR 190 resulting from radioactive gaseous and liquid effluents released to the environment following the SPU. Note that limits on dose to the public resulting from normal operation are addressed in 10 CFR 20, 10 CFR 50 Appendix I, as well as 40 CFR 190. However, 10 CFR 50 Appendix I (which is based on the concept of "As Low As Reasonably Achievable") is the most limiting. 10 CFR 20 does have a release rate criteria that does not exist in 10 CFR 50 Appendix I, but the ODCM controls actual performance and operation of installed equipment and releases, thus maintaining compliance with that aspect of 10 CFR 20. In addition, if the projected increase in offsite doses due to radioactive gaseous and liquid effluents either approach or exceed 10 CFR 50, Appendix I guidelines, then the methodology in the ODCM is utilized to determine compliance with 40 CFR 190. Per Radiological Effluent Control 3/4.11.4 of the ODCM, compliance with the limits of 40 CFR 190 needs to be addressed if the calculated doses from gaseous or liquid radwaste effluents exceed the limits imposed by 10 CFR 50 Appendix I by a factor of 2.

There are no changes as a result of the SPU to existing radioactive waste systems (gaseous and liquid) design, plant operating procedures or waste inputs as defined by NUREG-0017, Revision 1. Therefore, a comparison of releases can be made based on current versus SPU inventories/radioactivity concentrations in the reactor coolant and secondary coolant/steam. As a result, the impact of the SPU on radwaste releases and Appendix I doses can be estimated using scaling techniques.

Scaling techniques based on NUREG-0017, Revision 1 methodology were utilized to assess the impact of the SPU on radioactive gaseous and liquid effluents at CPNPP Units 1 and 2. Use of the adjustment factors presented in NUREG-0017, Revision 1 allows development of coolant activity scaling factors to address the SPU.

The SPU analysis utilized the core power operating history during the years 2001 to 2005 for both CPNPP units, the reported gaseous and liquid effluent and dose data during that period, NUREG-0017, Revision 1, equations and assumptions and conservative methodology to estimate the impact of operation at the analyzed SPU core power level. The results were then compared to the comparable data from current operation on radioactive gaseous and liquid effluents and the consequent normal operation offsite doses.

For the current condition, the evaluation utilized offsite doses based on an average 5-year set of organ and whole body doses calculated from effluent reports for the years 2001 through 2005. The licensed core power level for both units changed during the 2001 to 2002 time frame to the MUR power level of 3,458 MWt. The MUR license amendment was effective October 2001. The MUR was implemented at Unit 2 in 2001 and at Unit 1 in 2002. For purposes of conservatism, the SPU evaluation assumes the original licensed core power level of 3,411 MWt for the years 2001 and 2002 at both Units. The average 5-year set of organ and whole body doses were adjusted to a core power level of 3,458 MWt.

For the SPU condition, the system parameters utilized in the SPU analysis reflected the flow rates and coolant masses at an analyzed NSSS power level of 3,628 MWt and a conservative core power level of 3,684 MWt.

The maximum potential percentage increase in coolant activity levels due to the SPU, for each chemical group identified in NUREG-0017, was estimated using the methodology and equations found in NUREG-0017, Revision 1, and based on a comparison of the change in power level and in plant coolant system parameters (such as reactor coolant mass, steam generator liquid mass, steam flow rate, reactor coolant letdown flow rate, flow rate to the cation demineralizer, letdown flow rate for boron control, steam generator blowdown flow rate, or steam generator moisture carryover) for both current and SPU conditions. To estimate an upper bound impact on offsite doses, the highest factor found for representative isotopes in any chemical group (including corrosion products) in either unit, pertinent to the release pathway was applied to the average doses previously determined as representative of operation at current conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the SPU and to demonstrate that the estimated offsite doses following the SPU, although increased, will remain below the regulatory limits.

The impact of the SPU on solid radwaste generation was qualitatively addressed based on NUREG-0017, Revision 1, methodology, engineering judgment and the understanding of radwaste and affected plant system operation on the generation of solid radwaste.

The analysis concluded the following:

- Expected Reactor Coolant Source Terms

Based on a comparison of current versus SPU input parameters, and the methodology outlined in NUREG-0017, Revision 1, the maximum expected increase in the reactor coolant source is approximately 9.5 percent for noble gases, 6.6 percent for I-131, and 6.5 percent for long half-life activity. The above change is primarily due to the estimated decrease in RCS mass (~3 percent) and increase in effective core power level (~6.5 percent, that is, 3,684 MWt (power level conservatively analyzed for the uprate)/3,458 (MWt pre-uprate licensed power level)) between current and SPU conditions.

- Liquid Effluents

As discussed above, there is a maximum 6.5-percent increase in the radioactivity content of the liquid releases since input activities are based on long-term reactor coolant activity that is proportional to the core power uprate percentage increase, and on waste volumes that are essentially independent of power level within the applicability range of NUREG-0017.

Tritium releases in liquid effluents are assumed to increase approximately 6.5 percent (corresponding to the effective core power uprate percentage) since the analysis is based on changes in an existing facility's power rating without changing its mode of

operation. The Annual Radiological Environmental Operating Reports for 2005 and 2006 indicated that the highest aqueous tritium concentrations due to CPNPP operation are located at the Squaw Creek Reservoir (SCR). Increasing the average tritium concentration in releases by approximately 6.5 percent would leave the resulting long-term average tritium concentration at the SCR well below the reportable limit identified in Table 3.12-2 of the ODCM.

- Gaseous Effluents

For all noble gases, there will be a bounding maximum 9.5-percent increase of radioactivity content in effluent releases due to the effective core power uprate percentage increase and decrease in primary coolant mass.

Tritium releases in the gaseous effluents increase in proportion to their increased production (6.5 percent), which is directly related to core power and is allocated in this analysis in the same ratio as current releases.

The impact of the SPU on iodine releases is approximated by the effective core power level increase with the calculated increase in the reactor coolant and secondary coolant I-131 concentration of 6.6 and 12.6 percent, respectively. The 12.6 percent will be used as the limiting increase in thyroid doses due to iodine releases.

For particulates, the methodology of NUREG-0017 specifies the release rate per year per unit per building ventilation system. This is not dependent on power level within the range of applicability. Particulates released via the Turbine Building due to leakage of main steam and air ejector exhaust are generally considered to be a small fraction of total particulate releases. Therefore, minimal change would be expected for the SPU operations. However, a conservative approach is dictated by the fact that the annual effluent release reports do not delineate the source of particulates or iodines released. In addition, at CPNPP, tritium is included in the category with iodines and particulates (radionuclides with half-lives greater than 8 days).

On the secondary side, moisture carryover is a major factor in determining the non-volatile activity in the steam. The multiplier (~4.3) applicable to the particulates (cesium) released via the Turbine Building due to main steam leaks and condenser vacuum pump exhaust is higher than the percentage of the core power uprate (primarily due to a conservatively estimated 4.2-fold increase in moisture carryover due to the SPU, coupled with a 6.5-percent increase in coolant concentration). However the contribution of particulates to the "iodine and particulate" category was insignificant compared to the dose contribution from tritium. Therefore, the scaling factor for the entire "particulate and iodine" category was conservatively estimated at 6.5 percent for all organs except for the thyroid which was estimated at 12.6 percent. Since iodine dominated the dose impact and affected the average organ dose for two quarters of the 5-year effluent dose data evaluated, the average maximum organ dose in the particulate and iodine category was scaled to the thyroid/iodine increase.

- Estimated Impact on Effluent Doses - Compliance with 10 CFR 50 Appendix I

Table 2.10.1-2 shows that, based on operating history, the maximum estimated dose due to liquid and gaseous radwaste effluents following the SPU is significantly below the 10 CFR 50, Appendix I limits.

- Solid Radioactive Waste

For CPNPP, the volume of solid waste would not be expected to increase proportionally because the SPU neither appreciably impacts installed equipment performance, nor does it require drastic changes in system operation or maintenance. Only minor, if any, changes in waste generation volume are expected. However, it is estimated that the activity levels for most of the solid waste would increase proportionately to the increase in long half-life coolant activity bounded by the 6.5-percent maximum increase.

Taking into consideration the average capacity factor during the 5-year evaluation period of 0.8985, the total long-lived activity contained in the waste following SPU is estimated to be bounded by approximately 7.2 percent (that is, 6.5 percent/0.8985) over that currently stored on site.

In the long term, the direct shine dose due to radwaste stored onsite could be conservatively estimated to increase by approximately 7.2 percent as: a) current waste decays and its contribution decreases, b) the radwaste is routinely moved offsite for disposal, c) waste generated post-uprate enters into storage and d) plant capacity factor approaches the target of 1.0.

As the impact on direct shine doses is cumulative from wastes generated from all units onsite over the plants' lifetime and stored onsite, procedures and controls in the ODCM monitor and control this component of the offsite dose and would limit, through administrative and storage controls, the offsite dose to ensure compliance with the 40 CFR 190 direct shine dose limits.

- Impact of SPU on Direct Shine

As demonstrated by the CPNPP Annual Radioactive Effluent Reports, the annual direct shine dose during the pre-SPU 5-year period evaluated was negligible. For the SPU, the direct shine dose due to plant operation would increase by the increase percentage of the power level, that is, 6.5 percent. However, as discussed above, the direct shine contribution due to accumulation of stored solid radwaste, could increase by approximately 7.2 percent. A conservative bounding scaling factor of 7.2 percent would not change the estimated SPU direct shine dose which would remain negligible.

- Compliance with 40 CFR 190

The discussion below regarding compliance with 40 CFR 190 is provided for completeness even though, per the ODCM, demonstration of compliance with

40 CFR 190 is not required unless the dose limits of 10 CFR 50 Appendix I are exceeded by a factor of 2. Table 2.10.1-2 demonstrates that the SPU dose estimates are well below the design objectives of 10 CFR 50 Appendix I.

The 40 CFR 190 whole body dose limit of 25 mrem to any member of the public includes: a) contributions from direct radiation (including skyshine) from contained radioactive sources within the facility, b) the whole body dose from liquid release pathways, and c) the whole body dose to an individual via airborne pathways.

Taking into consideration the estimated annual SPU whole body dose of 0.19 mrem due to gaseous and liquid effluent releases ($7.00\text{E-}02$ mrem/yr and $1.20\text{E-}01$ mrem/yr, respectively), and the negligible direct shine dose contribution, it is concluded that the 40 CFR 190 whole body dose limit of 25 mrem/yr will not be exceeded by the SPU.

Results

CPNPP is required to meet the requirements of 40 CFR 190, 10 CFR 20 and 10 CFR 50, Appendix I. However, 10 CFR 50 Appendix I is the most limiting.

10 CFR 20 does have a release rate criteria that does not exist in 10 CFR 50 Appendix I, but the ODCM control actual performance and operation of installed equipment and releases thus maintaining compliance with that aspect of 10 CFR 20.

If the normal operation doses due to radioactive gaseous and liquid effluents either approach or exceed 10 CFR 50, Appendix I guidelines, the ODCM will ensure compliance with 40 CFR 190.

The SPU has no significant impact on the annual radwaste effluent doses (that is, this analysis demonstrates that the estimated doses following SPU will remain a small percentage of allowable Appendix I doses - see Table 2.10.1-2). It is therefore concluded that following SPU, the liquid and gaseous radwaste effluent treatment systems, in conjunction with the procedures and controls provided by the ODCM, will remain capable of maintaining normal operation offsite doses within the regulatory requirements.

2.10.1.2.5 Ensuring that Occupational and Public Radiation Exposures are ALARA

Introduction

As discussed in FSAR Sections 12.1 and 12.5, it is CPNPP policy to implement a Radiation Protection Program that meets the requirements of 10 CFR 20 and ensures that the occupational radiation exposures at CPNPP are kept ALARA.

Implementation of the overall requirements of 10 CFR 50, Appendix I relative to utilization of radwaste treatment equipment to ensure that radioactive discharges and public exposure are ALARA is formalized via controls imposed by the ODCM.

Description of Analyses and Evaluations

As noted in FSAR Section 12.5.3, ALARA procedures currently in place govern all activities in restricted areas at CPNPP. Design features credited to support CPNPP's commitment to ALARA operator exposures include shielding, which is provided to reduce levels of radiation; ventilation, which is arranged to control the flow of potentially contaminated air; an installed radiation monitoring system, which is used to measure levels of radiation in potentially occupied areas and measure airborne radioactivity throughout the plant; and respiratory protective equipment, which is used as prescribed by the Radiation Protection Program.

Compliance with the requirements of the ODCM ensures that radioactive discharges and public exposure are ALARA.

The SPU assessments documented in LR subsections 2.10.1.2.1 through 2.10.1.2.4 demonstrate that the dose limits imposed by regulatory requirements are met following the SPU. The SPU does not impact the effectiveness of the design features credited to support CPNPP commitment to ALARA operator exposures. The intent of the ALARA procedures remain unchanged, specifically, a) the allowable limits on operator and public exposure and b) the intent to keep operator and public exposure at a minimum.

Results

It is concluded that no additional steps are necessary to ensure that dose increases are maintained ALARA.

2.10.1.3 Conclusion

Luminant Power has assessed the effects of the proposed SPU on radiation source terms and plant radiation levels, the associated impact on shielding adequacy, radiation monitoring setpoints, post-accident vital area accessibility, and normal operation radwaste effluents. Luminant Power concluded that the evaluation adequately accounts for the effects on the proposed SPU on occupational and public radiation doses such that no additional steps are required to ensure that radiation doses will be maintained ALARA. Based on this, Luminant Power concluded that the occupational and public radiation dose controls will meet the CPNPP licensing basis with respect to the requirements of GDC-19; 10 CFR 20; 10 CFR 50, Appendix I; 40 CFR 190 and NUREG-0737, II.B.2. The proposed SPU is acceptable with respect to radiation protection and ensuring that occupational and public radiation exposures will be maintained ALARA.

Table 2.10.1-1

CPNPP SPU Design Reactor Coolant Activity Concentrations @ 3,684 MWt⁽¹⁾

Nuclide	Primary Coolant Activity Conc. (μCi/g)	Nuclide	Primary Coolant Activity Conc. (μCi/g)	Nuclide	Primary Coolant Activity Conc. (μCi/g)
Kr 83m	4.77E-01	Rb 86	2.28E-02	Te129	1.32E-02
Kr 85m	1.95E+00	Rb 88	4.54E+00	Te131m	3.49E-02
Kr 85	8.73E+00	Rb 89	2.07E-01	Te131	1.79E-02
Kr 87	1.29E+00	Sr 89	4.46E-03	Te132	3.07E-01
Kr 88	3.61E+00	Sr 90	1.77E-04	Te134	3.36E-02
Kr 89	1.04E-01	Sr 91	8.13E-03	Cs134	1.78E+00
Xe131m	3.20E+00	Sr 92	1.37E-03	Cs136	2.01E+00
Xe133m	4.81E+00	Y 90	4.93E-05	Cs137	1.10E+00
Xe133	2.80E+02	Y 91m	4.34E-03	Cs138	1.08E+00
Xe135m	5.36E-01	Y 91	6.12E-04	Ba137m	8.63E-01
Xe135	8.18E+00	Y 92	1.18E-03	Ba140	4.43E-03
Xe137	1.96E-01	Y 93	3.86E-04	La140	1.47E-03
Xe138	7.25E-01	Zr 95	6.94E-04	Ce141	6.69E-04
		Nb 95	6.86E-04	Ce143	5.28E-04
Br 83	1.02E-01	Mo 99	8.14E-01	Ce144	4.76E-04
Br 84	5.24E-02	Tc 99m	7.51E-01	Pr143	6.57E-04
Br 85	6.17E-03	Ru103	5.77E-04	Pr144	3.89E-04
I127 ⁽²⁾	7.45E-11	Ru106	1.61E-04	Cr 51	5.00E-03
I129	4.43E-08	Rh103m	5.78E-04	Mn 54	3.90E-04
I130	2.31E-02	Rh106	1.15E-04	Mn 56	1.40E-02
I131	2.87E+00	Ag110m	9.89E-04	Fe 55	2.10E-03
I132	3.07E+00	Te125m	3.20E-04	Fe 59	5.10E-04
I133	4.60E+00	Te127m	3.36E-03	Co 58	1.20E-02
I134	6.65E-01	Te127	1.19E-02	Co 60	1.40E-03
I135	2.53E+00	Te129m	1.16E-02	H-3	3.50E+00

Note:

1. Listed activity concentration includes 4-percent margin for potential future fuel cycle variability
2. Gram of I-127 per gram of coolant

Table 2.10.1-2				
Estimated Annual SPU Doses to the Public Due to Normal Operation Gaseous and Liquid Radwaste Effluents				
Type of Dose	Appendix I Design Objectives per unit	Base Case 100% Capacity Pre-Power-Uprate Case	Scaled Doses SPU Case	Percentage of Appendix I Design Objectives for Uprate Case
Liquid Effluents				
Dose to Total Body from All Pathways	3 mrem/yr	1.12E-01 mrem/yr	1.20E-01 mrem/yr	1.99%
Dose to Any Organ from All Pathways	10 mrem/yr	1.29E-01 mrem/yr	1.37E-01 mrem/yr	0.69%
Gaseous Effluents				
Gamma Dose in Air	10 mrad/yr	1.36E-02 mrad/yr	1.49E-02 mrad/yr	7.47E-02%
Beta Dose in Air	20 mrad/yr	3.25E-02 mrad/yr	3.56E-02 mrad/yr	8.91E-02%
Dose to Total Body of an Individual	5 mrem/yr	6.57E-02 mrem/yr	7.00E-02 mrem/yr	0.700%
Dose to Skin of an Individual	15 mrem/yr	8.58E-02 mrem/yr	9.13E-02 mrem/yr	0.304%
Radioiodines and Particulates Released to the Atmosphere				
Dose to any Organ from all Pathways	15 mrem/yr	1.64E-01 mrem/yr	1.84E-01 mrem/yr	0.614%

2.11 HUMAN PERFORMANCE

2.11.1 Human Factors

2.11.1.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The Comanche Peak Nuclear Power Plant (CPNPP) human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed stretch power uprate (SPU). The review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed SPU. The acceptance criteria for human factors are based on General Design Criterion (GDC)-19, 10 CFR 50.120, 10 CFR Part 55, and the guidance in Generic Letter (GL) 82-33 (NUREG-0737).

Current Licensing Basis

- GDC-19, Control Room, is described in Final Safety Analysis Report (FSAR) Section 3.1.2.10.

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is a seismic Category I structure. Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA), as detailed in FSAR Sections 12.1 and 15.6. The Control Room air conditioning system features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in FSAR Sections 6.4, 6.5, and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

In the unlikely event that access to the Control Room is restricted, the hot shutdown panel, local control stations, or manual operation of critical components can be used to affect hot shutdown from outside the Control Room for an extended period.

Before evacuation takes place, the reactor can be manually tripped and neutron level and control rod position can be verified. By use of appropriate procedures and equipment, the unit can also be brought to cold shutdown conditions from outside the Control Room. (See FSAR Section 7.4)

- 10 CFR 50.120 is described in FSAR Sections 13.2.1.2, Training Programs for Nonlicensed Personnel; and 13.2.2, Replacement and Retraining. As described in these FSAR sections, the qualifications and training of the personnel who provide offsite and

onsite support to CPNPP is based on a systems approach to training and is in accordance with 10 CFR 50.120.

- 10 CFR 55 is described in FSAR Section 13.2.1, Nuclear Training Program. As discussed in this FSAR section, licensed operators are qualified in accordance with 10 CFR 50 and 10 CFR 55. Their training is based on a systems approach to training (SAT). The program was credited by the National Nuclear Accrediting Board (10/25/90).
- The CPNPP plant procedure program is discussed in FSAR Section 13.5, Plant Procedures and Instructions. This section identifies the activities that must be conducted by procedures and instructions and provides an appropriate method to develop and approve these procedures and instructions.
- In addition to the above FSAR sections, human performance and human factors related elements are discussed in FSAR Section 7.7.1.12, Control Room Operating Console; and FSAR Section 7.7.1.11, Plant Computer System.

2.11.1.2 Technical Evaluation

Introduction

Human factors engineering and human performance initiatives are foundational characteristics that help ensure that the plant operators can effectively and safely operate the facility as well as mitigate emergency conditions. When initiating a plant change, completion of a human factors review for changes that may impact the Control Room layout (alarms, indication, appearance or performance) is required. In addition, plant operations staff has been represented and participated in SPU planning and modification development studies. Operating experience in the form of lessons learned from prior major modifications (such as CPNPP Unit 1 steam generator replacement) has provided valuable insights with respect to the human factors issues associated with major changes in operating methods. To ensure that changes associated with the SPU do not introduce any unanticipated consequences, a careful review of the effects of those changes on human performance was performed.

Description of Analysis and Evaluations

1. Changes in Emergency and Abnormal Operating Procedures:

The existing CPNPP emergency and abnormal operating procedures provide adequate guidance to cover the spectrum of anticipated plant events. The SPU will result in changes to the applicable abnormal operating procedures (ABNs) to address changes in setpoints, alarm response setpoints, and physical plant changes as a result of the SPU. The applicable emergency operating procedures (EOPs) may involve changes to setpoints. However no changes are required to the procedure steps and mitigation actions required as a result of the SPU.

The changes to EOPs and ABNs as a result of the SPU do not significantly impact operator actions and mitigation strategies. The changes will be appropriately proceduralized and the operators will receive appropriate classroom and/or simulator training for implementation.

2. Changes to Operator Actions Sensitive to SPU:

No changes are required to the EOP procedure steps and mitigation actions as a result of the SPU. Minor changes are expected to the ABN procedures to address additional heat loads generated by the main generator and auxiliaries due to SPU.

The changes in procedures do not significantly impact operator actions.

3. Changes to Control Room Controls, Displays, and Alarms:

Changes to Control Room controls and displays will not be extensive and will generally include rebanding indicators and rescaling loops for identified instrumentation. There will also be changes to several control board and computer alarms and limited changes to plant control systems.

Below is a summary of the changes identified (refer to LR sections 2.4.1, 2.4.2 and 2.5.5.4 for additional details):

- a. The following instrument loops are affected by the SPU (indicator banding, calibration range, and or scaling):
- Units 1 and 2 main steam flow
 - Unit 1 moisture separator reheater steam supply flow (calibration range and scaling)
 - Unit 1 and 2 Turbine first stage pressure (calibration range)
 - Unit 1 and 2, extraction steam pressure at the first point heaters (banding)
 - Unit 1 and 2, extraction steam pressure at the second point heaters (banding)
 - Unit 1 and 2 main feedwater flow (banding and scaling)
 - Unit 1 and 2 main feedwater pump turbine speed control (scaling)
 - Unit 1 and 2 condensate flow to gland steam condenser (banding)
 - Unit 1 and 2 heater drain pump discharge flow (indicator scale/banding, calibration range, and scaling)

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- Unit 1 and 2 feedwater pump suction flow (indicator scale/banding, calibration range, and scaling)
 - Unit 1 and 2 blowdown heat exchanger condensate water outlet temperature (banding)
- b. Alarm response procedures (Unit 1 and 2) will require revision as a result of setpoint changes:
- Feedwater pump low feedwater pump suction pressure, open condensate polishing bypass valve
 - Feedwater pump low feedwater pump suction pressure, open low pressure heater bypass valve
 - Feedwater pump low feedwater pump suction pressure, trip main feedwater pump
- c. Some plant computer setpoints will be changed for the following parameters:
- Reactor coolant system (RCS) N16 alarm and protection
 - RCS T_{avg}
 - Pressurizer level
 - Turbine first stage pressure
 - Other various alarm changes
- d. Changes to controls and control systems:
- Control rod speed program (power mismatch) in AUTO
 - Pressurizer level program
 - RCS T_{avg} program
 - Turbine first stage pressure
 - Manual turbine runback
 - Feedwater pump low feedwater pump suction pressure, open condensate polishing bypass valve
 - Feedwater pump low feedwater pump suction pressure, open low pressure heater bypass valve

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- Feedwater pump low feedwater pump suction pressure, trip main feedwater pump
 - Turbine bypass (steam dump) control system changes

Instruments associated with turbine first stage pressure will require scaling changes for various nuclear steam supply system (NSSS) inputs (anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC)), Permissive P7, rod control, block auto rod withdrawal, and steam dump control).

The SPU also involves other BOP hardware modifications as discussed in LR Section 1.0. These modifications include the main HP turbine upgrade, main transformer replacements, heater drain pump upgrades, Iso-phase bus duct cooling modifications, and turbine-generator component cooling modifications. Such hardware modifications may involve associated control system modifications and are evaluated for Human Factors as part of normal plant procedures in the design change process.

The operators will be provided detailed training related to the SPU modifications and resulting control board and procedure changes. Operators are provided station modification review packages as well as classroom and simulator training where appropriate. The initial plant startup of the uprated plant will be performed as an infrequent evolution and will be controlled by a power ascension testing program. (See LR Section 2.12)

4. Changes on the Safety Parameter Display System:

No significant safety parameter display system changes are anticipated as a result of SPU. Critical safety function status trees will be reviewed and revised as necessary for related changes to setpoints and decision points.

Any changes identified to the safety parameter display system will be captured through the normal update process, modification process, and interdepartmental reviews.

5. Changes to the Plant Simulator and Operating Training

The existing CPNPP Licensed/Non-Licensed Operator training programs employ the systematic approach to training process that has provisions for ensuring that adequate training is provided for plant modifications prior to implementation. Training will be initiated prior to the SPU implementation in 2008 for Unit 1 and 2009 for Unit 2. The training will focus on the SPU modifications and impact of the SPU on plant system operating conditions following implementation of the SPU. Through classroom and simulator training, the operators will be able to demonstrate understanding of the integrated plant response due to the SPU. Additional power ascension training will cover the testing and monitoring that will be performed during power ascension to the uprated power level.

Plant uprate analyses and modifications will be incorporated in the plant simulator software modeling. The control board physical changes as a result of SPU modifications and setpoint and scaling changes will be also be incorporated in the simulator. The simulator upgrades required for SPU will be evaluated, implemented, and tested in accordance with ANSI/ANS-3.5 1985. Simulator fidelity will be revalidated in accordance with Section 5.4.1 of ANSI/ANS-3.5 1985. These changes will be scheduled to accommodate the operator training to support SPU implementation.

Results

The changes to EOPs and ABNs as a result of the SPU do not significantly impact operator actions and mitigation strategies. The changes will be appropriately proceduralized and the operators will receive appropriate classroom and/or simulator training for implementation.

The results of the SPU human factors review show that changes to plant procedures, when prepared in accordance with the current procedure change control process, will not alter the basic mitigation strategies with which the operators are familiar. Changes associated with instrument scaling and setpoints will not introduce a level of complexity that would lead to misunderstanding the parameter. Operator training will provide effective reinforcement of procedure and plant physical changes as well as build proficiency with the required operator action changes.

2.11.1.3 Conclusion

Luminant Power has reviewed the changes to operator actions, human-system interfaces, procedures, and training required for the proposed SPU and concluded CPNPP has: (1) appropriately accounted for the effects of the proposed SPU on the available time for the operator actions, and (2) taken appropriate actions to ensure that the operator performance is not adversely affected by the proposed SPU. Luminant Power further concluded that CPNPP will continue to meet the current licensing basis with respect to the requirements of GDC-19, 10 CFR 50.120, and 10 CFR 55 following implementation of the proposed SPU. Therefore, Luminant Power finds the proposed SPU acceptable with respect to human factors.

2.12 POWER ASCENSION AND TESTING PLAN

2.12.1 Approach to SPU Power Level and Test Plan

2.12.1.1 Regulatory Evaluation

A stretch power uprate (SPU) test plan (LR Section 2.12.1.2.3) has been developed to ensure that the structures, systems, and components (SSCs) impacted by the proposed Comanche Peak Nuclear Power Plant (CPNPP) SPU have been identified, and to identify the testing required that will ensure all completed modifications allow continued safe and reliable operation of the plant within design limits. The SPU test plan will be implemented by a SPU power ascension test program (SPU test program) to validate that CPNPP Units 1 and 2 can be operated safely and reliably at the proposed SPU power level.

Development of the SPU test plan has considered the following evaluations:

- Conformance with applicable regulations including 10 CFR 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service.
- Plans for the initial approach to the proposed maximum licensed thermal power level, including verification of acceptable primary and secondary plant performances based upon meeting established acceptance criteria.
- Comparison of the SPU proposed tests with the original initial startup tests identified in the CPNPP Final Safety Analysis Report (FSAR). Planned tests and justification for excluding any original test from the SPU test program is provided in Table 2.12-1.
- Dynamic and Transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level.

The purpose of the SPU test program is to establish the programmatic and administrative requirements for CPNPP Units 1 and 2 power ascension testing, currently planned following the refueling outage in which the SPU is initiated (Unit 1/1RF13 (Fall 2008), Unit 2/2RF11 (Fall 2009)). The SPU test program will implement the SPU test plan, identify all proposed testing, procedures, and methodologies to accomplish the required tests and ensure test results meet established acceptance criteria and standards.

The SPU test program is intended to provide the following test methodologies:

- Normal startup and surveillance testing to verify acceptance of system performance following a routine post outage refueling
- Post-modification testing to demonstrate acceptance following implementation of an individual modification

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- Additional testing to ensure that SPU design conditions (such as piping vibration) satisfy design conditions for the uprated condition
 - Dynamic and transient testing, as needed to demonstrate that plant systems are capable of performing safely and reliably in the uprated condition.

Current Licensing Basis

The initial startup test program described in Table 14.2-3 of the FSAR was performed for Units 1 and 2 to ensure the safe and reliable operation of both units up to their initial core thermal power rating of 3,411 MWt. Following testing that introduced transient and dynamic perturbations to both the primary and secondary systems at all power levels up to 3,411 MWt, each unit recovered successfully within design limitations. Transients included 10 percent load swings at 50- and 75-percent rated thermal power (RTP), large-load reduction of 50 percent from approximately 100-percent RTP and full-load rejection and turbine trip from 100 percent. Control rods were used for a dynamic rod drop test, ejected and dropped rod worth measurements, and a xenon oscillation test. For both Units 1 and 2, maximum and/or minimum values of critical primary and secondary system parameters including core thermal-hydraulic limits were established to be within allowable limits.

Following the initial rating of 3,411 MWt, License Amendment No. 72 was approved for Unit 2 Cycle 5 (Spring 1999), which allowed for a measurement uncertainty recapture (MUR) uprate of approximately 1 percent to 3,445 MWt. Uprating was based upon reduction of uncertainty associated with the core thermal power measurement from 2 percent RTP to 1-percent RTP. A change to the nominal trip setpoint/allowable value of the overpower N-16 reactor trip and high neutron flux trip function were implemented.

Under the scope of License Amendment No. 89, Unit 1, Cycle 10 (Fall 2002) and Unit 2, Mid Cycle 8 (October 2001) were approved for MURs of 1.4 percent and 0.4 percent, respectively to 3,458 MWt, the current Units 1 and 2 license basis. A change to the nominal trip setpoint/allowable value of the overpower N-16 reactor trip and high neutron flux trip function were implemented.

Currently, core operating limits for Unit 1 and Unit 2 are based upon 100.6 percent of the licensed power level (3,458 MWt) assuming the feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the ultrasonic flowmeter.

2.12.1.2 Technical Evaluation

2.12.1.2.1 Introduction

CPNPP is proposing an SPU to increase both Unit 1's and 2's core thermal power to 3,612 MWt. This uprate involves changes to each plant's configuration to accommodate the higher reactor power limit as well as the larger steam and feedwater flows commensurate with the power increase. As a result of these changes, testing is required to ensure that each unit can be operated safely in its uprated condition.

2.12.1.2.2 Background

The proposed SPU at CPNPP will result in each reactor operating at a new core thermal power of 3,612 MWt. The current licensed core thermal power is 3,458 MWt for each Unit. CPNPP has significant operating experience, over 16 years, at its initial and current operating condition. Each unit is a Westinghouse four-loop design. A similar SPU power level has been successfully achieved recently by another Westinghouse four-loop design plant, Seabrook Station, which increased its power level from 3,411 to 3,587 MWt and 3,587 to 3,648 MWt in two successive cycles, each with no adverse affects.

In a pressurized water reactor (PWR), the biggest change in system operating parameters occurs in the secondary side where mass flow is increased commensurate with the uprate. Minor changes may also occur in primary side temperatures to provide additional heat transfer in the steam generators. Based on heat balance models (LR Section 2.5 Tables 2.5.5.1-1, 2.5.5.4-1, and 2.5.5.4-2) for a nuclear steam supply system (NSSS) power level of 3,628 MWt, CPNPP Unit 1 and 2 main steam and condensate/feedwater flows will increase by approximately 5-percent. The Unit 1 main steam pressure is calculated to remain unchanged, whereas the Unit 2 main steam pressure is calculated to decrease slightly. Unit 1 T_{avg} is expected to increase slightly while Unit 2 T_{avg} is expected to remain the same following the uprate.

In order to accommodate this new thermal power, changes in plant operating parameters have to occur. However, the fundamental operating characteristics at uprate remain consistent with the operating characteristics prior to the uprate, and also consistent with other similar units that have been uprated. This means that pre-uprate plant operating experience and industry operating experience provide valuable insight to the viability of the CPNPP SPU. This operating experience has been incorporated into the SPU test plan.

As discussed in LR Section 1.0, plant modifications are required to support power operation at the proposed uprated condition. Post-modification testing of these modifications will be performed to ensure proper installation and performance. In addition to the post-modification tests, routine post refuel and system surveillance tests will be performed to verify acceptance of primary and secondary system operation up to and including operation at an NSSS power level of 3,628 MWt.

Development of the SPU test plan has drawn on the results of the original startup and test program, the plant operating characteristics following previous SPUs, the startup and test program for Unit 1 following steam generator replacement, and applicable industry experience with similar uprates as a means of ensuring safe and reliable operation at the new uprated condition. Additionally, the potential consequences of unnecessarily challenging the plant and safety systems through transient and dynamic testing have been considered during development of the test plan. Where applicable, comparison will be made between recent operating data and the data that will be gathered during the proposed uprate testing to ensure that the test results are satisfactory. For example, pre-SPU piping vibration data will be compared to post-SPU piping vibration data.

In summary, the SPU test plan is comprised of a mixture of power ascension monitoring, post-modification testing, load reduction testing, surveillance testing, and analytical evaluation to ensure that Units 1 and 2 can operate safely and reliably at the new uprated core thermal power condition. The following subsections describe the CPNPP Units 1 and 2 SPU test plan and proposed test program.

2.12.1.2.3 SPU Test Plan

2.12.1.2.3.1 General Discussion

The SPU test plan has been developed based on review of similar test programs performed at other plants, detailed system and integrated plant analyses performed in support of the SPU, the results of plant response and performance following previous core power uprates of Units 1 and 2 from 3,411 to 3,458 MWt and the test results obtained during the return to service testing program performed for the Unit 1 steam generator replacement project.

Prior to the commencement of power ascension startup testing, the SPU test program will require the completion of numerous activities, which include:

- Review and revision of applicable plant operating procedures, administrative procedures, surveillance test procedures, calibration procedures, chemical and radiological procedures, and other similar procedures.
- Review and revision of computer software programs as required to support the SPU test plan and the new SPU power level.
- Incorporation of applicable plant instrumentation setpoint changes and recalibration of instrumentation as required.
- Implementation and successful post-modification testing of all scheduled plant design modifications associated with the uprate. (Some testing may be completed at various power levels during power ascension.)
- Review of temporary modifications to assure there is no impact on the ability of the effected equipment to support the uprate, and that uprate will not have an adverse impact on any existing plant condition.

Additionally, commitments which were the result of the SPU License Amendment Request, the NRC Safety Evaluation Reports (SER), and any other actions associated with the CPNPP SPU implementation, will be verified as being closed, included in the SPU test program, or evaluated as not impacting power ascension.

The SPU test program is being developed in advance of the SPU outage and will be implemented with the start of the SPU outage to verify the following:

- Plant systems and equipment affected by SPU are operating within design limits

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- Nuclear fuel thermal limits are maintained within expected margins and the core is operating as designed
 - Reactor control systems are stable and controlling parameters are within acceptable limits
 - Feedwater, condensate, and heater drain pumps are stable and operating within acceptable limits
 - Piping vibration measurements are within acceptable levels
 - Equipment cooled by turbine plant cooling water is operating within acceptable limits
 - Moisture separator reheater and feedwater heater drains and level control are stable
 - Turbine-Generator electrical performance is acceptable
 - Area radiation levels are acceptable and stable
 - General area and local environmental conditions are acceptable

2.12.1.2.3.2 Test Conditions

During the SPU startup, power will be increased in accordance with a pre-established schedule, stopping at pre-determined power levels for steady-state data gathering and test data evaluation. These pre-determined power levels are referred to as power plateaus. The typical post-refueling power plateaus include core mapping at approximately 30 percent, 80 percent, and reactor thermal power condition of 100 percent.

The power ascension test program will include the gathering of steady-state data at approximately 3,458 MWt (current licensed core power), 3,535 MWt, and 3,612 MWt. This will provide extrapolated verification of the performance of the SPU and SPU related modifications. In particular, by comparison of the plant data with established acceptance criteria, the SPU test program will provide assurance that unintended interactions between the various modifications have not occurred such that integrated plant performance is not adversely affected. Additionally, a piping vibration walk down of selected balance-of-plant (BOP) piping outside the Containment Building that could be impacted by the SPU will be performed at the selected power levels.

Upon attaining a steady-state core power level of 3,612 MWt, test data will be evaluated against its performance acceptance criteria (that is, design predictions or limits). If the test data satisfies the acceptance criteria, then system and component performance will be considered to have satisfied their design requirements.

A flux map will be obtained at the SPU 100 percent RTP (3,612 MWt) to accomplish the following:

- Demonstrate by measurement that the operating characteristics of the core are consistent with the design predictions at SPU
- Demonstrate by measurement that the plant is capable of operating within the limits imposed by the operating license and Technical Specifications at SPU
- Determine by measurement, the data required to establish appropriate operating limits at SPU (100-percent plateau only)
- Determine by measurement, the data required to calibrate plant instrumentation dependent upon measured physics parameters at SPU

The large transient tests described in FSAR Table 14.2-3 such as unit load transients, reactor trip, and large-load rejections are not planned to be re-performed during the SPU power ascension testing as justified in LR subsection 2.12.1.2.3.6 below. However, any significant differences between analytical predictions and steady-state test data obtained during power ascension testing will be evaluated and reconciled before proceeding to higher power levels.

Post-modification testing will be performed in accordance with normal plant procedures as identified through normal plant processes (see Section 2.12.1.2.3.3). Specific examples, for balance-of-plant systems include:

- Hydraulic interactions between the heater drain pumps modification and the feedwater and condensate interface, as well as the impact of the higher main feed flow and the associated increased piping pressure loss.
- Impact of the uprate conditions on secondary cooling capacity of the turbine plant cooling water system including upgrade to the iso-phase bus duct coolers and exciter air coolers.
- Monitoring of individual control systems such as moisture separator and feedwater heater drain level control.
- Turbine-Generator performance testing.

The SPU test program consists of a combination of normal startup and surveillance testing, post-modification testing, and power ascension testing deemed necessary to support acceptance of the proposed SPU. The following system and equipment testing has been evaluated for inclusion into the SPU test plan and test program:

- Initial startup testing identified in FSAR Table 14.2-3 (See Table 2.12-1 for SPU planned testing)
 - Reactor coolant system flow test
 - Reactor coolant system flow coastdown test
 - Control rod drive tests
 - Rod position indication
 - Reactor trip system
 - Auxiliary startup instrumentation
 - Calibration of process temperature and nuclear instrumentation
 - Chemical tests
 - Radiation surveys
 - Process and effluent radiation monitoring test (Unit 1)
 - Process and effluent radiation monitoring test (Unit 2)
 - Moderator temperature reactivity coefficient
 - Control rod reactivity worths
 - Boron reactivity worth
 - Core reactivity balance
 - Loss of offsite power
 - Rod drop tests
 - Flux distribution measurements
 - Core performance evaluation
 - Unit load transients
 - Remote shutdown
 - Turbine trip/generator load rejection
 - Reactor coolant leak
 - Rod control system
 - Automatic reactor control system
- Pre-modification baseline testing
 - Turbine performance test (high-pressure turbine replacement)
 - Piping vibration monitoring (balance of plant)
 - Monitoring of plant parameters
- Post-modification testing (as required by the design change process) see LR Section 1.0 for list of Plant Modifications

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- Power ascension testing
 - Monitoring of plant parameters
 - Piping vibration monitoring (balance of plant)

2.12.1.2.3.3 Post Modification Testing Requirements

Plant modifications will be implemented at CPNPP in order to achieve and support the SPU rated power. Plant modifications are controlled by administrative procedures. These procedures provide configuration control, installation instructions, and testing requirements. Post modification testing verifies satisfactory performance of the modification in accordance with the design documentation. The performance of post-modification testing is addressed by existing programmatic controls within the design modification process. Functional and operational post modification testing will be performed for each modification to verify satisfactory installation and performance.

2.12.1.2.3.4 Acceptance Criteria

The acceptance criteria for the CPNPP power ascension test program will be established as discussed in Regulatory Guide 1.68 and will be included in the applicable test procedures performed for the SPU.

Level 1 acceptance criteria are values for process parameters assigned in the design of the plant that are safety significant. If a Level 1 criterion is not satisfied, the power ascension will be stopped and the plant will be placed in a condition that is judged to be safe based upon prior testing. The power ascension test procedure and Technical Specifications will provide direction for actions to be taken to assure the plant is safe and stable. Resolution of the issue that resulted in exceeding the Level 1 criterion must be resolved by equipment changes or through engineering evaluation, as appropriate. Following resolution, the applicable test portion must be repeated to verify that the Level 1 requirement is satisfied. A description of the problem must be included in the report documenting successful completion of the test.

Level 2 acceptance criteria are values that relate to plant functions or parameters that are not safety significant. If Level 2 criteria are not met, power ascension testing may continue. Investigation of the issue that resulted in exceeding the Level 2 criterion may continue in parallel with the power ascension. These investigations would be handled by existing plant processes (such as corrective action program) and procedures.

2.12.1.2.3.5 Vibration Monitoring

A piping vibration monitoring program has been developed to ensure that any steady-state flow induced piping vibrations following SPU implementation are not detrimental to the plant, piping, pipe supports or connected equipment.

Observed piping vibrations will be evaluated to ensure that damage will not result. The predominant way of assessing these vibrations is to monitor the piping during the plant heat up

and power ascension. The methodology to be used for monitoring and evaluating piping vibration is summarized below.

Piping outside the Containment Building that will experience increases in their flow rates or are directly attached to lines that will experience increases in their flow rate as a result of SPU conditions will be observed. Piping of probable interest are identified in the vibration monitoring program. Branch piping attached to these lines will also be observed as experience has shown that branch lines are as susceptible (or more susceptible) to flow induced vibration. All listed lines and their branches are a part of one of the following systems:

- Main steam, reheat, and steam dump (outside of containment)
- Steam generator feedwater (outside of containment)
- Condensate
- Heater drains
- Extraction steam
- Steam generator blowdown cleanup

The vibration monitoring program scope will also include lines or equipment within the monitored systems that have been modified or otherwise identified through the CPNPP corrective action program as having already experienced vibration issues.

The main steam and feedwater piping inside containment is not considered to be a target area based upon on the following:

- The main steam and feedwater piping is well supported and seismically designed.
- The piping is large diameter, not overly flexible, with large radius bends and few elbows.
- There are no long cantilever branch lines or branch lines with heavy unsupported valves.
- There is no history of vibration problems in these lines at CPNPP, nor at other 4 loop Westinghouse designed reactors.

The piping and equipment within the scope of the vibration monitoring program will be observed at the selected power levels. The initial walkdown at 3,458 MWt will establish the baseline piping vibration level. A final walkdown is planned at the SPU power level (3,612 MWt).

By comparing the observed pipe vibrations/displacements at the selected power levels with previously established Level 1 and 2 acceptance criteria, potentially adverse pipe vibrations will be identified, evaluated and resolved.

2.12.1.2.3.6 Transient and Dynamic Tests

A small load reduction test of at least 50 MWe will be performed to confirm the expected integrated response of the following automatic control systems at SPU conditions;

- Rod Control System
- Steam Generator Water Level Control System
- Pressurizer Level Control System

This load reduction test, along with routine startup and surveillance testing, post modification testing, and power ascension testing and monitoring will provide the bases for confirmation of predicted and extrapolated system dynamic behavior. The results of this testing and monitoring, combined with SPU analyses, will be used to ensure that the plant systems, including the above identified automatic control systems are capable of performing safely and reliably in the uprated condition.

Based upon the following, large load rejections and turbine trip transient tests are not planned for the Unit 1 and Unit 2 uprate testing program.

- A LOFTRAN computer model was developed for CPNPP Units 1 and 2 at the proposed SPU conditions using best-estimate analyses. The analyses were performed using the multi-loop version of the Westinghouse LOFTRAN computer code. This computer model simulates the overall thermal-hydraulic and nuclear response of the NSSS as well as various control and protection systems. The LOFTRAN code has been reviewed and approved by the NRC (refer to LR subsection 2.4.2.1).
- SPU analyses (LR Section 2.6 and subsections 2.4.2.1 and 2.8.5) indicate that the system dynamic behavior is satisfactory and that no new thermal-hydraulic phenomena or adverse system interactions are created by the proposed uprate. The SPU analyses determined that the NSSS and balance of plant (BOP) instrumentation ranges, scaling, and setpoints used in the reactor protection, engineered safety features actuation system (ESFAS), and reactor control instrumentation remained adequate for the SPU. The SPU analyses confirmed that departure from nucleate boiling (DNB), reactor coolant system (RCS) pressure, and secondary system pressure remains within the allowable design margins and the response to the design basis operational transients remains acceptable.
- During the initial startup program for each unit, both Unit 1 and Unit 2 primary and secondary systems demonstrated stability at all power levels up to 3,411 MWt when subjected to dynamic and transient tests at pre-determined power plateaus. Transients included 10-percent load swings at 50- and 75-percent RTP, large-load reduction of 50-percent from approximately 100-percent RTP and full-load rejection and turbine trip from 100 percent. In all cases, both plants responded within acceptance criteria limitations.

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- Transient and dynamic testing performed during Unit 1 startup testing following the replacement of D-4 steam generators with $\Delta 76$ steam generators demonstrated that both primary and secondary systems responded within acceptance criteria limitations, verifying plant system interaction and automatic control system response. Transient testing included a 8-percent (90 MWe) step load reduction at 80-percent RTP followed by an equal step load increase of 8 percent and a large step load reduction of 275 MWe from 1,000 MWe.
 - Experiences at similar plants with similar uprates indicate that large transient and dynamic testing is not required for this magnitude of uprate.
 - Not performing these types of test will preclude unnecessary challenges to plant and safety systems.

2.12.1.3 Conclusion

Luminant Power has prepared an SPU test plan that includes plans for the initial approach to the proposed SPU thermal power level including the testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased licensed thermal power level. The post-modification tests that are considered in the verification of plant integrated response are identified above in the SPU test program. Luminant Power concludes that the proposed SPU test program that will implement the SPU test program will provide adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed SPU, or modified to support the proposed SPU, will perform satisfactorily in service. Further, Luminant Power provides assurance that the SPU test program will satisfy the requirements of 10 CFR 50, Appendix B, Criterion XI. Therefore, Luminant Power finds the SPU test plan and the proposed SPU test program acceptable with respect to the proposed SPU.

**Table 2.12-1
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
1	Reactor Coolant System Flow Test ----- Sheet No. 2	Yes	To verify predicted Reactor Coolant System cold leg volumetric flow rates in hot standby and during power ascension testing are within design flow limits at normal operating temperature and pressure with all reactor coolant pumps running, to verify flow transmitters are satisfactorily aligned for zero flow and full flow conditions and to demonstrate that pressurizer spray is within acceptable limits.	RCS flow testing will be performed as part of the SPU Power Ascension Test Program in order to validate the calorimetric flow measurement test This test is performed routinely to satisfy Tech Spec Surveillance requirements. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. The flow rate through the reactor coolant system will not change as a result of SPU. Additionally, the operation of this system is verified by routine surveillance testing. RCS full flow verification is performed per applicable plant procedures. Spray flow is not impacted by SPU. However the adequacy of pressurizer spray flow is routinely verified during normal operations.
2	Reactor Coolant System Flow Coastdown Test ----- Sheet No. 3	No	To measure the reactor coolant system flow rate decrease subsequent to a simultaneous trip of all four reactor coolant pumps, and to measure the delay times associated with assumptions of the loss of flow accident analysis. Satisfies the NSSS vendor acceptance criteria which verifies the loss of flow analysis of the FSAR, and the associated delay times are within acceptable limits.	This test is <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the RCS flow is not changed by the SPU, there is no modification to the RCPs and there is no impact to the coastdown times.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
3	Control Rod Drive Test ----- Sheet No. 4	Yes	To verify control rod bank start and stop setpoints, verify proper slave cyclers timing and drive mechanism operation, check rod speeds are within acceptable limits, and demonstrate the capability of the CRDMs to respond to signals from the Reactor Control System.	This test is performed routinely to satisfy Tech Spec Surveillance requirements. The SPU impacted rod drop times (LR Section 2.2.3). The operation of these systems is verified by routine surveillance testing. Rod Drop testing and CRDM step trace testing will be performed per applicable plant procedures.
4	Rod Position Indication ----- Sheet No. 5	No	To verify that the Digital Rod Position Indication system provides proper rod position indication and alarms based on simulated and/or actual inputs over the entire length of travel of each rod cluster control assembly. Specifically, test was performed to verify the rod position indicators perform satisfactorily for each full-length rod cluster control assembly over its entire length of travel, to verify the rod-bottom alarms function in accordance with design specifications and that indication accuracy and alarm setpoints are within design criteria.	This test is performed routinely to satisfy Tech Spec Surveillance requirements, but is <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the rod control system has performed its intended function during all phases of plant operation. Additionally, the operation of this system is verified by routine surveillance testing. DRPI testing is performed per applicable plant procedures.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
5	Reactor Trip System ----- Sheet No. 6	No	To demonstrate the proper functioning of the Reactor Trip System, including the capability to test the operation of the reactor trip breakers using bypass breakers. Specifically, each control rod drive mechanism unlatches upon opening of the trip breakers, associated reactor trip breaker bypass breaker remains closed when each reactor trip breaker is opened for test, and interlocks which prevent closing both reactor trip breaker bypass breakers simultaneously function in accordance with design requirements.	This test is performed routinely to satisfy Tech Spec Surveillance requirements, but is <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the logic of the Reactor Trip System will not be changed as a part of this SPU and the test does not need to be repeated since the initial testing had satisfactory results. New reactor trip setpoints (e.g. OT N16, OP N16) for SPU will be verified by instrument calibration tests. Additionally, the operation of these systems is verified by routine surveillance testing. Reactor Trip System Tests are performed per the normal surveillance procedures.
6	Auxiliary Startup Instrumentation ----- Sheet No. 7	No	To demonstrate the proper response of the temporary neutron detectors and proper functioning of their associated indicating and recording functions. Specifically, the auxiliary startup instrumentation responds properly to a neutron source providing indication and an optional audible signal. The instrumentation indicates increasing neutron levels during early stages of fuel loading.	This test is <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the SPU does not alter this system as it is no longer used.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
7	Calibration of Process Temperature and Nuclear Instrumentation ----- Sheet No. 8	Yes	<p>To calibrate and adjust the operational settings of the source, intermediate, and power range detectors. Specifically, the source range channel operating voltages are set for optimum neutron response, the source, intermediate, and power range channels exhibit satisfactory overlap, the power range excore detectors display a linear output over the range of normal power operation, accurately indicate the actual reactor power level as determined by calorimetric measurements and selected alarm features function properly and to verify channel indication overlap and power range detector output linearity.</p> <p>Also, to calibrate and adjust the operational settings of the N-16 power detectors and reactor coolant average temperature instrumentation system. Specifically, the N-16 power and T_{ave} channels accurately indicate the actual reactor power and reactor coolant average temperature as determined by calorimetric measurements.</p>	<p>These tests are performed routinely to satisfy Tech Spec Surveillance requirements. The operation of these systems will be verified by routine surveillance testing.</p> <p>Process Temperature Instrumentation: N16 instrumentation and Cold Leg Temperature RTDs are routinely calibrated and adjusted by changes to N16/T_{avg} loop scaling calculations and RTD cross calibrations, respectively.</p> <p>Nuclear Instrumentation: Overlaps of excore Source Range/Intermediate Range detectors and Intermediate Range/Power Range detectors are verified during routine startups using normal integrated plant operating procedures. Power Range detectors are aligned within +2% of RTP per applicable surveillance test procedure (plant calorimetric).</p>
8	Chemical Test (RCS and Secondary Coolant Chemistry) ----- Sheet No. 10	No	To establish the water chemistry of the Reactor Coolant System, and to verify the capability to maintain Reactor Coolant System chemistry at power and during power escalation.	This test is not required to support the proposed SPU. SPU does not change normal operating chemistry specifications. Normal chemistry monitoring during power ascension will be performed.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
9	Radiation Surveys ----- Sheet No. 11	Yes	To verify radiation shielding effectiveness by measuring radiation dose levels at preselected locations within the plant during low, intermediate and high reactor power level operation.	A radiation survey will be conducted at 3,612 MWt to establish a new baseline dose level and to verify expected dose rate increases associated with the SPU. The Initial Startup Test program performed surveys at 0 – 5%, 45 – 55% and 95 – 100% power. Measurement of radiation dose levels at low and intermediate power levels will not be required as a baseline has already been established for the current power level of 3,458 MWt.
10	Process and Effluent Radiation Monitoring ----- Sheets No. 12 & 13	Yes	To verify the proper performance of the process and effluent radiation monitoring equipment under actual nuclear operating conditions.	Radiation monitor settings and surveillance will be performed per existing plant procedures. The SPU impacts these systems by increasing the amount of activity processed through them. However, the basic function of the system is not impacted and the capacity of the system remains acceptable.
11	Moderator Temperature Reactivity Coefficient ----- Sheet No. 14	Yes	To confirm that the actual moderator temperature coefficient of reactivity is within acceptable limits.	These tests are performed routinely to satisfy Tech Spec Surveillance requirements. The SPU may slightly affect MTC therefore routine surveillance testing will be performed to determine ITC and MTC and the Rod Withdrawal Limit.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
12	Control Rod Reactivity Worth ----- Sheet No. 15	No	To verify the rod worths of the control and shutdown RCCA banks are within design specifications.	These tests are performed routinely to satisfy Tech Spec Surveillance requirements, but are <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this process and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, this process is verified by routine surveillance testing to determine differential and integral rod worths of individual RCCAs using rod swaps.
13	Boron Reactivity Worth ----- Sheet No. 16	Yes	To measure the reactivity worth of the boron in the reactor coolant for determining that it is within acceptable limits.	These tests are performed routinely to satisfy Tech Spec Surveillance requirements. Routine surveillance testing will be performed to determine differential boron worth (pcm/ppm), boron endpoint and all rods out (ARO), just critical boron concentration.
14	Core Reactivity Balance ----- Sheet No. 17	Yes	To verify that actual core reactivity effects are in agreement with design values and within acceptable limits.	These tests are performed routinely to satisfy Tech Spec Surveillance requirements. Routine surveillance testing will be performed to measure ARO-hot zero power, ARO-hot full power, core reactivity balance.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
15	Loss of Offsite Power ----- Sheet No. 18	No	To demonstrate the proper plant response following a plant trip with no offsite power available. The on-site power supplies (i.e., diesel generators) shall auto-start and operate the necessary controls, equipment and indication to remove decay heat and maintain the Reactor Coolant System in a shutdown condition for the duration of the test.	These tests are performed routinely to satisfy Tech Spec Surveillance requirements, but are <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. SPU modifications have no effect on diesel generator loading and decay heat removal equipment. Therefore, this test is not required to be performed at the uprated power conditions. Additionally, the operation of plant systems is verified by routine surveillance testing. Loss of Offsite Power tests (Integrated Test Sequence tests) are performed each refueling outage with the core offloaded.
16	Rod Drop Tests ----- Sheet No. 19	Yes	To determine the rod drop time of each full-length rod cluster control assembly (RCCA) under hot full flow Reactor Coolant System conditions is acceptable in accordance with plant Technical Specifications.	This test is performed routinely to satisfy Tech Spec Surveillance requirements. The SPU analysis impacted rod drop times (LR Section 2.2.3). The operation of these systems is verified by routine surveillance testing. Rod Drop testing will be performed per applicable plant procedures
17	Flux Distribution Measurements ----- Sheet No. 20	Yes	To determine the reactor core power distribution. The core flux distributions indicated by the flux map are acceptable in accordance with plant Technical Specifications where applicable.	This test is performed routinely to satisfy Tech Spec Surveillance requirements. The Tech Spec surveillance will be performed during power ascension.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
18	Core Performance Evaluation ----- Sheet No. 21	Yes	To verify the operating characteristics of the core and the calibration of the flux and temperature instrumentation during power escalation. Specifically, The core performance margins are within design predictions for normal rod configurations and the calibration of the flux and temperature instrumentation has been verified.	<p>This test is performed routinely to define the activities for the startup testing program from 0% to 100% power. The testing sequence originally performed consisted of a group of plant procedures to establish plant conditions and to change reactor power, to stabilize power for testing at specific pre-determined power levels including instrumentation tuning, and to extrapolate power to ensure parameter indicative of DNBR and linear heat rate are acceptable up to the power level afforded protection by the power range trip setpoints. Correct operational alignment of the NIS and Reactor Control Systems is verified. This process will continue to be performed for normal restart following a refueling outage. For SPU power ascension, the following testing is planned using existing plant procedures:</p> <ul style="list-style-type: none"> • Demonstrate by measurement that the operating characteristics of the core are consistent with the design predictions.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
18 (cont.)	Core Performance Evaluation ----- Sheet No. 21	Yes		<ul style="list-style-type: none"> • Demonstrate by measurement that the plant is capable of operating within the limits imposed by the operating license and Technical Specifications. • Determine by measurement the data required to establish appropriate operating limits. • Determine by measurement the data required to calibrate plant instrumentation dependent upon measured physics parameters.
19	Unit Load Transients ----- Sheet No. 22	Yes	<p>To demonstrate satisfactory plant transient response to various specified load changes and trips, to monitor the behavior of reactor control systems during these transients, and, if necessary, optimize the reactor control system setpoints. Plant response to the unit load transients is acceptable in accordance with design specifications, and the Reactor Control System parameters reach steady state values without appreciable overshoot or oscillation subsequent to a step change. Specifically,</p> <p>a. Initiate a step change in power level of 10 percent and monitor Reactor Coolant System behavior in response to the transients. For Unit 1, this test will be performed at approximate power levels of</p>	<p>A load transient test of at least 50 MWe will be performed to demonstrate the integrated response to the control system setting changes. There are no major hardware modifications planned for NSSS components as part of the SPU that would affect plant transient responses any differently than demonstrated during Initial Startup Testing. Since the NSSS control system functional design and hardware are not impacted and the analyzed (LR Section 2.4.2.1) 10% load swings and 50% load rejection Condition I operating transients show acceptable stability, setpoints, and margin to reactor trip and ESF actuation, the NSSS control systems are acceptable for operation at full power SPU conditions.</p>

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
19 (cont.)	Unit Load Transients ----- Sheet No. 22	Yes	<p>50 percent, 30 percent (following completion of 50 percent testing) and 100 percent. For Unit 2, this test will be performed at approximate power levels of 50 percent and 75 percent.</p> <p>b. Monitor plant response to a 50 percent load reduction, from power levels of approximately 75 percent and 100 percent.</p> <p>c. Monitor plant response to a plant trip from power levels up to 100 percent.</p>	<p>A reactor trip, or the potential for a reactor trip, from high power level results in an unnecessary plant transient and the risk associated with such a transient, while small, should not be incurred. Based on the completed analyses and the avoided risk of an unnecessary plant transient from 100% SPU, 10% step changes and a large load reduction of 50% from 75% and 100% SPU power to verify proper operation of the plant and automatic control systems, is not required in the CPNPP Power Ascension Test Program.</p> <p>The following tests were completed successfully during the CPNPP Initial Startup Test Program:</p> <p>a. Unit 1 10% load swing tests were performed successfully at approximately 35%, 50% and 100% RTP. Unit 2 10% load swing tests were successfully performed at 46% and 75% RTP.</p>

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
19 (cont.)	Unit Load Transients ----- Sheet No. 22	Yes		<p>b. For Unit 1, a 50% rapid turbine load reduction was successfully performed from 75% and 100% RTP. For Unit 2, a 50% rapid turbine load reduction was successfully performed from 100% RTP.</p> <p>c. A Turbine Trip/Generator Load Reject from 100% RTP was performed successfully for both Unit 1 and Unit 2.</p> <p>See Section 2.12.1.2.3.6 for additional justification for not re-performing these tests.</p>
20	Remote Shutdown ----- Sheet No. 23	No	To demonstrate the capability of performing a safe plant shutdown in accordance with design requirements, maintain the plant in a hot standby condition, and to demonstrate the ability to cooldown from hot standby to cold shutdown conditions from outside the Control Room, using the minimum shift crew. Verify that the Remote Shutdown Panel selector switches properly transfer control from the Control Room to the Remote Shutdown panel.	This test is <u>not</u> required to support the proposed SPU. Test objectives were successfully demonstrated for both Unit 1 and Unit 2. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
21	Turbine Trip/Generator Load Rejection ----- Sheet No. 24	No	To demonstrate the ability of the plant to sustain a full load rejection of the turbine generator at full power, and to evaluate plant response to the transient demonstrating that the resultant plant trip is sustained and the control systems function properly within specified limits to preclude lifting of the pressurizer or main steam safety valves, and that subsequent to the trip, the plant can be maintained in a hot standby condition.	This test is <u>not</u> required to support the proposed SPU. See Test Basis for Item No. 19 and LR Section 2.12.1.2.3.6 for additional justification for not re-performing these tests.
22	Reactor Coolant Leak ----- Sheet No. 25	No	To demonstrate the leak tightness of Reactor Coolant System pressure boundary and of the reactor vessel flange after the system has been closed following fueling. To determine the leak rate for primary-to-secondary leakage, Reactor Coolant Pump seal leakage, other identified leakage, and any unidentified leakage are within Technical Specification limitations.	These tests are performed routinely to satisfy Tech Spec Surveillance requirements, but are <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this process and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, this process is verified by routine surveillance testing to measure both identified and unidentified leakage.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
23	Rod Control System ----- Sheet No. 26	No	Demonstrate that the Full Length Rod Control System performs the required control and indication functions in order to verify it is operational and ready for use prior to initial criticality. Specifically, the Rod Control System responds properly to normal input signals in the manual mode, providing correct bank overlapping, rod speed, direction and indication.	This test is performed routinely to satisfy a combination of Tech Spec Surveillance requirements and scheduled Preventive Maintenance procedures, but is <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the rod control system has performed its intended function during all phases of plant operation. Additionally, the operation of this system is verified by routine surveillance testing and preventive maintenance procedures.
24	Automatic Control System Test ----- Sheet No. 27	No	Demonstrate the ability of the Automatic Reactor Control System to return reactor coolant temperature to the programmed setpoint and to maintain it. Specifically, The Reactor Coolant System Temperature (T_{avg}) returns to within 1.5°F of the setpoint (T_{ref}) following the positive and negative temperature transient.	This test is <u>not</u> required to support the proposed SPU. Test objectives were successfully demonstrated for both Unit 1 and Unit 2. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. However, the automatic control system will be observed during a load reduction test of at least 50 MWe. See Item 19 of this table.

**Table 2.12-1 (cont.)
CPNPP SPU Test Plan**

Item No.	TEST DESCRIPTION ----- FSAR Table 14.2-3 Sheet No.	TEST PLANNED FOR SPU ⁽¹⁾ (Yes/No)	INITIAL STARTUP TEST OBJECTIVE	SPU TEST BASIS (1) The scope of SPU planned testing is described in this column
25	Incore Nuclear Instrumentation ----- Sheet No. 28	No	To demonstrate the capability of the Incore Nuclear Instrumentation to remotely position the incore neutron detectors for the purpose of core flux mapping, and to supply the appropriate digital and analog signals to the plant computer. Specifically, the Incore Nuclear Instrumentation channels function properly and supply appropriate outputs to the plant computer. The indexing system functions in accordance with design requirements. After fuel loading, free passage to all positions is demonstrated. Response to neutron flux at power meets design criteria.	These tests are performed routinely to satisfy Tech Spec Surveillance requirements, but are <u>not</u> required to support the proposed SPU. The SPU has no adverse affect on this system and does not invalidate the test as originally performed. Therefore, this test is not required to be performed at the uprated power conditions. Specifically, the In-Core Detector System is used during normal plant operation and has proven itself to be reliable. Additionally, the operation of these systems is verified by regular surveillance testing.

2.13 RISK EVALUATION

2.13.1 Regulatory Evaluation

The impact of a stretch power uprate (SPU) for Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 has been evaluated by reviewing the plant changes associated with the SPU and the plant specific Probabilistic Risk Assessment (PRA) for internal and external events at power, and the available shutdown risk evaluation information. This included the changes made to the plant PRA model due to the $\Delta 76$ steam generators for Unit 1. The changes made due to the $\Delta 76$ steam generators will not be discussed here but are used in this evaluation.

The guidance provided in Regulatory Guide (RG) 1.174 (Reference 1), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and "Review Standard (RS) -001 (Reference 2), Review Standard for Extended Power Uprates" was used to perform an analysis of the plant changes associated with the SPU. The evaluation focused on the impacts of SPU on core damage frequency (CDF) and large early release frequency (LERF) due to internal events, external events, and shutdown operations. Although the SPU is not a risk informed application, this risk analysis is provided for information. The analysis made use of the current CPNPP Units 1 and 2 PRA models.

Current PRA Basis

To ensure a high-quality PRA and to provide quality control to the PRA process, two types of independent reviews were conducted during the development of the PRA model used to support the individual plant examination (IPE) submittal. One was an internal review and one was conducted by outside PRA experts. The PRA was subsequently reviewed as part of the Westinghouse Owners Group (WOG) peer review. In addition to the peer review, a focused, independent industry peer review of the PRA Revision 3 changes was completed in the spring of 2005. The major model features addressed in this review included the reactor coolant pump (RCP) seal loss-of-coolant-accident (LOCA) model update to the WOG 2000 model, thermal-hydraulic (T-H) analyses associated with RCP seal LOCA scenarios, loss-of-offsite-power (LOOP) model changes, and the quantification process. This review was completed based on the American Society of Mechanical Engineers (ASME) PRA Standard (Reference 3).

2.13.2 Technical Evaluation

2.13.2.1 Level 1 Internal Events

An evaluation was performed to analyze the changes due to the SPU implementation for their potential impact on the PRA models for internal events in the following key areas: initiating event frequency, component reliability, system success criteria, and operator response. The SPU related modifications listed in Table 1.0-2, in LR Section 1.0 were considered in this evaluation. Each of these areas is specifically addressed in the following subsections, followed by a description of the overall impacts on CDF and LERF from internal events for the SPU.

2.13.2.1.1 Initiating Event Frequency

The CPNPP Units 1 and 2 Internal Events PRAs address LOCAs, steam generator tube rupture (SGTR), LOOP, transients, loss of support systems, and anticipated transients without scram (ATWS), among others. The underlying contributors to these initiating events were reviewed to determine the potential effects of the SPU on the initiating event frequencies.

The loss-of-coolant inventory category includes five LOCA events ranging from excessive LOCAs to very small LOCAs (isolable and non-isolable). The LOCA events (including excessive, large, and medium LOCA) are based on potential passive structural failures. Also included are the interfacing system LOCA (ISLOCA) and SGTR events. The operating pressure for the RCS has not changed. Therefore, the pressure used for the LOCA initiating events remains the same. The initiating event frequencies for these events have been derived from industry data and are not affected by the SPU.

For the small and very small LOCA (non-isolable) pipe break, the frequency has been derived from industry data. The reactor coolant system (RCS) pressure will not change due to the SPU and no modifications to the RCS piping are planned. Therefore, the SPU should have no impact on the small LOCA frequency.

The small LOCA (isolable) event represents a pressurizer power operated relief valve (PORV) that has opened and failed to close, but is isolated by the operator action to close the block valve. Plant changes associated with the SPU would not affect the probability of failure to reclose a PORV. The normal RCS operating pressure and the PORV setpoints remain the same, so additional challenges to the valves are not expected due to the SPU. Therefore, the small LOCA (isolable) initiating event frequency remains applicable for the SPU conditions.

The ISLOCA event is initiated by a failure in the interface between the RCS and low-pressure system piping or components, resulting in a failure that allows reactor coolant to be released outside containment. This event is quantified using a simplified fault tree. The RCS normal operating pressure does not change for the SPU; the valve arrangements for the interfacing systems do not change; the RCS leakage surveillance and testing frequencies do not change; and the number of challenges to the isolation valves as a result of normal plant operation is not expected to change. Therefore, the ISLOCA initiating event frequency is not affected by the SPU.

The current Unit 1 SGTR initiating event frequency is derived from industry data. The Unit 1 $\Delta 76$ steam generators (SGs) have been designed and evaluated for the SPU conditions. The evaluations demonstrate that the $\Delta 76$ SG will perform acceptably at the SPU power level. The evaluations for the $\Delta 76$ SG include T-H performance, structural integrity, and tube wear. The Alloy 690 tubes in the $\Delta 76$ SG are less susceptible to tube ruptures than the tubes in the Unit 1 original steam generators. Therefore, the Unit 1 SGTR initiating event has been requantified specific to the Alloy 690 tubes. This resulted in a reduction of the SGTR initiating event frequency due to the $\Delta 76$ steam generators. No change to the SGTR initiating event frequency was required as a result of the SPU evaluation.

The Unit 2 SGTR initiating event frequency is based on generic industry data, and the Unit 2 steam generators have been evaluated at the SPU conditions. The evaluation indicated that operating steam generator pressure will remain essentially the same as the pre-SPU pressure. The SPU evaluation showed that the SGTR initiating event frequency will not be impacted by Unit 2 plant specific modifications.

The transient initiating event frequencies were derived by using plant specific data and generic industry data through a Bayesian updating process. As part of the SPU effort, plant systems have been reviewed for continued operability at the SPU conditions. In some cases, system changes have been made so that systems will adequately perform their functions at the SPU conditions. An example of these changes is resetting control and protection system instrument setpoints so that adequate analysis and operating margins are maintained for both units. The evaluation performed for the SPU shows that the transient initiating event frequencies calculated for CPNPP Units 1 and 2 will not be significantly affected by SPU conditions.

The LOOP initiating event frequency was derived by using plant specific data and generic industry data (Reference 4a through 4c) through a Bayesian updating process. The mean CPNPP LOOP frequency is 3.24E-02 per year. The frequency of LOOP events is dictated by the reliability of the switchyard and grid. The evaluations of the plant electrical systems provide assurance that the frequency of a LOOP event is not adversely affected by the SPU.

The support system fault initiating event frequencies were quantified using fault trees that model plant components. The initiating event frequencies quantified in this manner include those for loss of service water, loss of component cooling water, and loss of heating, ventilation, and air conditioning (HVAC). There are no changes related to the SPU that would affect system success criteria. Therefore, initiating event frequency, as modeled in the PRA are not affected. It has been concluded that the components and their reliability are not affected by the SPU conditions; therefore, the calculated initiating event frequencies remain applicable.

Failure of the reactor to trip automatically following an initiating event (ATWS) is considered in the PRA models in the course of developing plant response scenarios. Therefore, ATWS events are not defined as a separate initiating event category. A separate set of event trees is developed instead for transients and support system faults that are followed by a failure of reactor trip. NSSS control systems were evaluated for stability and operability, and the control rod drive mechanisms were evaluated. These evaluations show that the frequency of an ATWS event from rod control failures is not affected by the SPU conditions.

It is concluded that the SPU will have no adverse effect on the internal events PRA initiator frequencies. Any future deviations in initiating event frequencies will be identified by existing monitoring processes, such as licensee event reports, condition reporting, and industry events databases and will be included in the periodic PRA updates. In addition, safety system actuations are trended under the Maintenance Rule as an indicator of unnecessary challenges of safety-related equipment.

2.13.2.1.2 Component Availability and Reliability

As summarized in this licensing report, numerous SPU analyses were performed to assure that the plant components can be operated reliably, meet design requirements, and address impacts on operating margins. Component performance is assured through analysis of margins and upgrading or replacing components as required. Mitigating system equipment reliability is assured through analysis of equipment performance to demonstrate that it meets the design requirements.

A review was performed of the changes associated with the SPU that might affect systems and associated equipment that are important to plant risk. The balance-of-plant equipment that is being added or modified does not require any additional modeling or changes to the PRA plant model to assess risk.

The SPU may result in some components being refurbished or replaced more frequently. This may result in increased unavailability if performed while the plant is on-line. To assess potential future impacts, reliability and availability of the equipment will be monitored by plant programs. The existing component monitoring programs (such as preventive maintenance and the Maintenance Rule) will identify any additional degradation as a result of the SPU. The PRA maintenance and update process in place at CPNPP provides the means for identifying any future impact on unavailability and component failure rates and addressing them in the PRA model. In addition, changes to reliability that impact initiating events will be identified and incorporated in future updates.

In summary, modifications are being made to improve the performance of certain plant equipment and systems for the SPU. Therefore, it is anticipated that they will continue to be operated within design constraints and component failure rates are not expected to change with the implementation of the SPU. Monitoring programs provide the appropriate means to identify and account for future changes in equipment performance.

2.13.2.1.3 Important Systems and Functions

An evaluation was performed to identify the effects of the SPU on the functionality of plant systems. Systems related to decay heat removal, containment, steam conversion, RCS inventory, engineered safety features, reactivity control, and electrical/instrumentation and controls were evaluated.

The functionality of those systems is not expected to be impacted by the SPU. Some system components are being modified or added, and some instrumentation setpoints are being changed to accommodate the SPU. In some instances, these changes are being made specifically to preserve functionality of the system under SPU conditions.

The evaluation of the potential for plant changes associated with the SPU to affect the plant PRA model concluded that the SPU modifications will have no adverse affect on the system functions important to plant risk. Plant modifications were made to maintain or improve the performance of certain equipment under SPU so that plant systems and equipment will continue

to be operated within design constraints and that component failure rates and unavailability will not significantly change with the implementation of SPU. Therefore, system functionality is not impacted.

2.13.2.1.4 Success Criteria and LOOP Recovery

Detailed analyses were performed to account for the effect of the increase in the CPNPP thermal power level on the internal events PRA success criteria. Success criteria are defined for the accident sequences modeled in the PRA to establish, via analysis or other suitable means, whether or not core damage occurs. The success criteria specify the plant systems and equipment required to function to address critical safety functions. These critical safety functions include: reactivity control, RCS pressure control/pressure boundary integrity, RCS and core heat removal, RCS inventory control, and long-term RCS inventory control and heat removal. The following paragraphs discuss the affects of the SPU on success criteria and LOOP recovery.

Electric Power Recovery/Station Blackout (SBO)

The electric power recovery model in the PRA considers the non-recovery of electric power in sequences for which emergency AC power is lost. MAAP thermal-hydraulic analyses for SBO provided time to core damage (see Tables 2.13-1 and 2.13-2 for the electric power recovery sequences). The convolution approach was applied, using the time to core damage results of the MAAP thermal-hydraulic analyses as input, to calculate an electric power non-recovery probability for each SBO sequence.

The CPNPP model for recovery of the electric power during a specific event scenario accounts for the causes and timing of the power failure events, the sequencing of failures and recovery actions, and the available time window for success before the onset of core damage. Equipment failures and recovery can occur at any time during the 23-hour period after event initiation. Therefore, the CPNPP electric power recovery model is a time-integrated model for failures and recovery actions that are necessary to assess the effect of diverse failure causes and to model corresponding responses that require different amounts of time to complete and that are started at different times after event initiation.

To assure that the PRA model accurately reflects the actual plant configuration and recoveries, the recovery model was requantified. The new values were calculated and used as part of the uprate model even though the change was not risk significant. As shown in Tables 2.13-1 and 2.13-2, the times to core damage for most of the SBO sequences were not significantly reduced.

Small LOCA

Small LOCA sequences are defined in the PRA as those smaller than 2 inches equivalent diameter but larger than breaks for which the normal charging system could provide continuous makeup. Those small LOCA sequences that are dependent on bleed-and-feed cooling for decay heat removal, or dependent on steam generator cooldown and depressurization could be

affected by increases in core power. The bleed-and-feed success criteria, as currently modeled in the PRA, require a single operable PORV. The analyses show that a single PORV will still be adequate after the SPU. The steam generator cooldown and depressurization success criteria will not change because the steam generator atmospheric steam dump valve capacity has been shown to be adequate for SPU conditions.

MAAP analyses have been performed that show that after SPU implementation, for a small LOCA with failed high head safety injection (HHSI), operators will still be able to depressurize the RCS with sufficient time available to use low head safety injection (LHSI) to prevent core damage.

Medium and Large LOCA

The medium LOCA break spectrum includes breaks ranging from 2 inches up to 6 inches equivalent diameter. The smaller end of the break spectrum may have sensitivities similar to those found for the small LOCA scenarios. The success criteria for the larger end of the medium-break spectrum and the large-break spectrum are not affected by more power given that the PRA models the design basis criteria (such as one of two LHSI pumps delivering flow through one of four intact cold leg injection paths and injection from two of four intact cold leg accumulators) that remain unchanged.

Anticipated Transient Without Scram

There are changes associated with the SPU that could potentially affect ATWS-related CDF. Changes in the core design for the SPU conditions may result in increased unfavorable exposure time (UET), which is a measure of the time from the beginning of the cycle when total negative reactivity feedback, including moderator temperature coefficient, may be insufficient to prevent exceeding the ASME Service Level C stress limit for the RCS. This could result in an increase in ATWS-related CDF. According to LR subsection 2.8.5.7 of this report, analyses have been performed that provide support for a 5-percent UET. Therefore, this change is not risk significant.

Human Reliability Analysis (HRA)

In the CPNPP PRA model, the probability of human reliability actions are based in part on timing derived from MAAP analyses for various accident sequences. Because of the higher decay heat levels for the SPU, there is a potential for reducing the time available for the operators to complete recovery actions. Therefore the timings of these actions were reviewed for the new decay heat levels expected after SPU implementation.

A comprehensive review of the post-initiator operator actions was performed using new MAPP runs based on the $\Delta 76$ SGs and power uprate. Both the time window for initiating the actions and the time for performing the actions were reviewed. Based on this review, it was determined that none of the operator failures modeled in the PRA changed as a result of the SPU. Though there were minor changes in the MAPP calculated timing for some actions, the analyses showed that the current timings used in the HRA were conservative for both the window of time

to initiate the action and for performing the action. Therefore, no changes were made to the HRA values due to the SPU for this assessment.

2.13.2.1.5 PRA Level 2/LERF Analysis

An evaluation was performed to identify the effects of the SPU on the CPNPP containments and the contributors to the large early release conditional probability. The dominant contributors to large early release frequency (LERF) in the CPNPP PRA include events that are either faulted SGTR events or interfacing system LOCAs outside containment. The plant changes associated with the SPU were determined to have an insignificant or no adverse impact on these contributors.

It should be noted that the steam generators in CPNPP Unit 1 are Westinghouse Model $\Delta 76$ with Alloy 690 material tubes. The Alloy 690 material has shown resiliency against the failure mechanisms that occur in the original steam generator design, such as primary water stress corrosion cracking. Therefore, the Unit 1 SGTR initiating event frequency was calculated to be $9.24E-04$ per year per steam generator. Credit was taken for this decrease for CPNPP PRA for Unit 1 due to the $\Delta 76$ SGs. The SPU did not require any changes.

Late containment failures modeled in the CPNPP PRA include those attributed to overpressurization due to steam, overpressurization due to non-condensables, late hydrogen burn, and failure due to overtemperature. The plant changes reviewed were determined to have an insignificant or no adverse impact on the total failure contribution from these contributors.

2.13.2.1.6 Summary of Internal Events Evaluation Results

The internal events risk assessment of the CPNPP SPU was performed via the revision and quantification of the existing CPNPP PRA model to account for the above noted changes in SBO non-recovery probabilities. Included in the quantification are the updated model due to the replacement of the steam generators in Unit 1 and any added components due to that replacement. Two PRA models were used in the assessment of the risk impact of the SPU:

- SPU model – The SPU SBO non-recovery probabilities were used to update the current internal events PRA models to create the SPU models.
- Current (pre-SPU) Model – The current internal events PRA models of record were used as a baseline for the SPU model.

All models were quantified and their results compared in order to assess the effect of the SPU on the internal events Level 1 risk. The Unit 1 CDF changed from $9.41E-06$ to $9.55E-06$ and Unit 2 CDF changed from $9.31E-06$ to $9.81E-06$. The Unit 1 LERF changed from $4.87E-07$ to $4.91E-07$ and Unit 2 LERF changed from $6.11E-07$ to $6.32E-07$. As a result of this assessment, there was a very small (less than $1E-06$ /year) increase in the internal events CDF and a very small (less than $1E-07$) increase in internal events LERF due to changes in the SBO non-recovery probabilities.

2.13.2.2 External Events

Qualitative evaluations were performed to assess the impact of the changes due to SPU on external events analyses, specifically seismic events, fires, high winds, floods, and other external events. Each of these is individually addressed in the following subsections.

2.13.2.2.1 Fire

The CPNPP external events analysis fire assessment employs the Electric Power Research Institute (EPRI) fire PRA methodology. This method consists of determining plant-wide ignition sources, establishing fire ignition frequencies on a building and fire zone basis, and performing an analysis on the effects of fires associated with these ignition sources on the components important to mitigating systems that affect accidents or transients resulting from fires. Since the SPU has very few changes important to mitigating systems, a qualitative evaluation of the effects of SPU on fire risk was done, using the knowledge of the fire PRA as input.

An evaluation of the plant changes resulting from the SPU shows that the fire risk impacts are very small to insignificant. The modifications to the plant were reviewed for their placement in the plant. The exciter air cooling and heater drain pump modifications are to be placed in the Turbine Building. The new output transformers will replace the existing ones. The addition of the plant equipment associated with these modifications is an insignificant increase in the fire loading of the Turbine Building. Therefore, this and other modifications are deemed to have minimal changes to ignition sources and fire loading that can impact mitigating equipment. Therefore this plant change will not cause a large increase in risk due to fire.

A review of plant response to Appendix R to 10 CFR 50 fire response and operator timing show that the impacts on timing for operator actions are minimal and, therefore, pose no additional risk to the plant (see LR Section 2.5.1.4.2).

2.13.2.2.2 Seismic Analysis

The CPNPP external events analysis seismic assessment employs the Seismic Margin Methodology that is based on the EPRI methodology described in the EPRI NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (Reference 5). This methodology consists of defining the equipment required to safely shut down the plant following a review-level seismic event and then evaluating the equipment through walkdown and margin analysis to show that the equipment will, in fact, survive at the review-level seismic accelerations. Since CPNPP is categorized as a reduced scope plant, the review-level earthquake is the same as the design basis for the Seismic Category 1 structures, systems, and components (SSCs), namely the safe shutdown earthquake (SSE).

Since there are no changes being made to the RCS, the original analysis remains applicable to the SPU conditions. An analysis of the plant changes resulting from the SPU indicates that these changes will not result in changes in SSC response to a seismic initiator, nor do they result in any new seismic core damage or large early release scenarios. The plant changes have a negligible impact on the structural response of the plant, and they have a small impact

on the availability and performance of necessary mitigation systems for a seismic event. Equipment installed or modified as a result of the SPU will meet the applicable seismic design criteria.

2.13.2.2.3 High Winds

The occurrence of high winds that can cause plant damage is location specific. Damaging winds in the coastal region are typically due to hurricanes, and in the interior of the country, tornadoes become important. The dominant CDF sequences associated with high winds typically involve LOOP in combination with random failures of emergency AC power. LOOP events from high winds are already accounted for in the LOOP initiating event frequency used in the internal event PRA model. The LOOP initiating event frequency includes a contribution for severe weather.

None of the modifications to the plant due to a SPU involve equipment that affects offsite power. Therefore, the plant modifications due to the SPU do not impact the high winds analysis.

2.13.2.2.4 Flood

The occurrence of floods that can cause plant damage is location specific. The CPNPP Individual Plant Examination of External Events (IPEEE) concludes that, for external flooding of the Turbine Building a pipe break in the circulating water system was the dominant contributor. The amount of piping and additional water added to the Turbine Building due to the SPU plant modifications is insignificant when compared to the circulating water system flows. The replacement of the output transformer will not change the flood analysis of the switchyard. Therefore, the plant modifications due to the SPU do not impact the flood analysis.

2.13.2.2.5 Other External Events

Other external events include transportation and nearby facility accidents, and the other external events listed in Table 4.1 of NUREG-1742 (Reference 6). As concluded in the NUREG, these events do not account for a significant risk contribution in any of the IPEEE submittals. This conclusion is consistent with the conclusions and insights from the CPNPP IPEEE.

2.13.2.3 Shutdown Risk

The impact of the SPU on plant risk at low power and shutdown risk was evaluated in a qualitative manner by addressing the questions posed in Table III-1 of Standard Review Plan (SRP) 19.0, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decision-Making: General Guidance" (Reference 7) to determine if the impacts on shutdown risk would be important. Based upon the responses to the SRP 19.0 questions on shutdown risk, increase in decay heat is expected to result in a small decrease in the time available for operator actions during shutdown operations.

CPNPP has shutdown administrative procedures that provide guidance for evaluation of shutdown safety and for surveillance during plant operating Modes 5 and 6, for planned and

forced outages. These procedures are consistent with the guidance in NUMARC 91-06 (Reference 8). They require monitoring of the plant defense-in-depth features available during these operating modes, and provide guidance for evaluating the adequacy of protective measures and specify actions to be taken to ensure that there are adequate protective measures in place. The administrative procedures require development of a pre-outage safety review of key shutdown safety functions. This guidance will continue to be used following the SPU.

2.13.2.3.1 SRP 19.0 Evaluation

Does the application introduce new initiating events or change the frequencies of existing events?

- Typical initiating events during shutdown conditions include:
 - Loss of residual heat removal – No plant modifications are being made to the residual heat removal system. Therefore, there is no expected change in frequency or additional initiating event categories due to the SPU.
 - Loss of offsite power – There is no expected change in the frequency of LOOP.
 - Loss of support systems – There are no changes to support systems that require any change in modeling. Therefore, there is no change to any loss of support system initiating event.
 - Loss of inventory – There is no proposed modification to the NSSS. Therefore, there is no change to any loss of inventory initiating event frequency.
 - Reactivity accidents – There is no proposed modification to the NSSS (including chemical and volume control and safety injection). Therefore, there is no change to any reactivity accident initiating event frequency.

None of the changes per the SPU are expected to introduce or change initiating event frequencies.

- Does the application affect the scheduling of outage activities?

There are no planned modifications to the residual heat removal system (RHRS). This system is expected to continue to meet its design requirements.

Since the decay heat levels are expected to be slightly higher at SPU conditions, it may take a few hours longer to achieve cold shutdown. This will cause very little change in the shutdown schedule, and has no direct safety impacts on the schedule.

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- Does the application affect the ability of the operator to respond to shutdown events?

The following shutdown safety functions are tracked during an outage:

- RCS decay heat removal
- RCS inventory control
- AC power availability
- Reactivity control
- Containment integrity
- Spent fuel pool cooling

The possible initiating events during shutdown are generally defined as loss of the shutdown safety functions. The SPU does not increase the frequency of these initiators, but may impact the operators' response to loss of shutdown safety functions.

RCS Decay Heat Removal

The RHRS performance evaluation notes that RHR remains adequate to maintain refueling temperatures and a uniform boron concentration in the RCS. The increase in decay heat due to SPU will decrease the time for the operators to respond to a loss of shutdown cooling.

Maintaining an adequate defense-in-depth for this safety function, in accordance with the CPNPP procedures, minimizes the impact of this decreased response time. A pre-outage shutdown safety review is prepared for each refueling outage according to the guidance of the CPNPP procedures. As expected, the increased power level results in an increased decay heat level. For SPU conditions, the estimated time to boiling is typically on the order of a few minutes less than that for the pre-SPU conditions. These small changes in time to boil occur many hours after shutdown. Therefore, they do not impact shutdown operations. Prior to every planned shutdown at CPNPP, a new time-to-boil calculation is performed. Also, the requirements for the defense-in-depth are tracked during the outage with the use of the Outage Risk Assessment and Management (ORAM) software.

RCS Inventory Control

The increase in RCS temperature and the increase in decay heat will decrease the time for the operators to respond to a loss of RCS inventory control. Maintaining an adequate defense-in-depth for this safety function, at all times in accordance with CPNPP procedures, minimizes the impact of this decreased response time. A pre-outage shutdown safety review is prepared for each refueling outage according to the CPNPP procedures. The current procedures include guidance for evaluating the defense in depth for RCS heat removal. Also the requirements for the defense in depth are tracked during the outage with the use of the ORAM software.

Reactivity Control

The effect on reactivity control is minimal since the RHRS remains adequate to maintain a uniform boron concentration in the RCS. Maintaining an adequate defense in depth for this safety function, at all times in accordance with CPNPP procedures, minimizes the impact of this decreased response time. A pre-outage shutdown safety review is prepared for each refueling outage according to the CPNPP procedures. The current procedures include guidance for evaluating the defense in depth for reactivity control. Also the requirements for the defense in depth are tracked during the outage with the use of the ORAM software.

AC Power Availability

The increase in RCS temperature and the increase in decay heat will decrease the time for the operators to respond to a loss of electrical systems. Since the electrical systems support the systems required for the other safety functions, maintaining an adequate defense in depth for this safety function minimizes the impact of this decreased response time. The current procedures include guidance for evaluating the defense in depth for electrical systems. Also the requirements for the defense in depth are tracked during the outage with the use of the ORAM software.

Containment Integrity

The containment integrity safety function provides the capability to close the containment following a loss of another safety function. Therefore, the response time for this safety function is decreased by the decreased response time for the other safety functions. Maintaining an adequate defense in depth for this safety function minimizes the impact of this decreased response time. Administrative closure controls are required to be implemented to ensure closure capability prior to core boiling.

The increased power level results in an increased decay heat level. For SPU conditions, the estimated time to boiling is typically on the order of a few minutes less than that for the pre-SPU conditions. These small changes in time to boil occur many hours after shutdown. Therefore, they do not impact shutdown operations. Prior to every planned shutdown at CPNPP, a new time to boil calculation is performed.

Spent Fuel Pool Cooling

LR Section 2.5.4.1 of this report notes that the spent fuel cooling systems were evaluated with consideration of the SPU. Under normal and abnormal conditions, the fuel pool water temperature does not exceed limits associated with the pool structure, liner, cooling system, or system component.

-
- Does the application affect the reliability or availability of equipment used for shutdown conditions?

Existing component monitoring programs can account for any additional equipment wear as a result of a SPU. While the SPU may result in some components being refurbished or replaced more frequently, the functionality and reliability of components can be maintained to the current standards.

- Does the application affect the availability of equipment or instrumentation used for contingency plans?

Existing component monitoring programs can account for any additional equipment wear as a result of a SPU. While the SPU may result in some components being refurbished or replaced more frequently, the functionality and reliability of components can be maintained to the current standards.

2.13.2.3.2 Conclusion

The increase in decay heat will result in a small decrease in the time available for operator actions during shutdown. However, maintaining an adequate defense in depth for the shutdown safety functions at all times, in accordance with CPNPP procedures, minimizes the impact of this decreased response time. The SPU will have no unique or significant impacts on shutdown risk.

2.13.2.4 Quality of Probabilistic Risk Assessment

To ensure a high-quality PRA and to provide quality control to the PRA process, two types of independent reviews were conducted during the development of the PRA model used to support the individual plant examination (IPE) submittal. One was done internally by Luminant Power, and the other was done externally by outside PRA experts. In general, both reviews were applied to the entire examination process except when it was not possible due to the availability of resources or required skills. In those few cases, as a minimum, each task was reviewed thoroughly by either an internal or external independent reviewer. Further, a final independent review was performed after the IPE study was completed. A team of PRA experts was selected from the industry to independently review the entire IPE study and its supporting analyses. The review team spent one week at the Luminant Power offices where documents, procedures, and supporting calculations and analyses were available for use. The results of all independent review activities performed by internal and external reviewers were documented as part of the IPE documentation requirements. This process has been continued since the IPE with the WOG peer review and the external peer review of the CPNPP updated models. A discussion of the WOG and another subsequent peer review is provided below.

WOG and Other Peer Reviews

A WOG peer review of the CPNPP PRA model was performed during the spring of 2002. The conclusion of the peer assessment is that the CPNPP PRA can be effectively used to support

risk significance evaluations with deterministic input, subject to addressing the items identified as significant in the technical element summary and Facts & Observations (F&O) sheets. There were three Level A F&Os.

Two Level A F&Os involved steam generator tube rupture and the application of the 24-hour mission time concept for both CDF and LERF considerations. The basis and success paths for the steam generator tube rupture model were clarified to provide for actions beyond the 24-hour mission time to assure that the plant is in a stable condition. To address this, it was determined that changes to the PRA event and fault trees were needed for long-term cooling after a steam generator tube rupture. These changes were incorporated into, and are part of, the current PRA model.

A third Level A F&O was written to address cutsets with multiple human errors and to revise dependency calculations if necessary. This item was found not to adversely affect the technical adequacy of the PRA. To address this, a PRA utility program was used to identify unique combinations of multiple human actions. These combinations were reviewed on a scenario basis to assure that dependencies were identified and handled as appropriate. Changes were made to the model where required to address these dependencies.

There were several Level B F&Os. CPNPP addressed each of the Level B F&Os and incorporated those items into the PRA model where appropriate. In summary, all of the Levels A and B F&Os were fully resolved.

In addition to the above described peer review, a focused, independent industry peer review of the Revision 3 changes was completed in the spring of 2005. The major model features addressed in this review included the RCP seal LOCA model update to the WOG 2000 model, T-H analyses associated with seal LOCA scenarios, LOOP model changes, and the quantification process. This review was completed based on the ASME PRA Standard (Reference 3). No Category A or B F&Os were identified by this peer review. All other F&O items were resolved and incorporated into Revision 3B of the model as appropriate.

2.13.2.5 Summary of Risk Impacts

The impact of the SPU is small for initiating event frequencies, component reliability, important systems and system functions, and Level 2/LERF, as modeled in the internal events at power PRA and for external events. The SPU will affect certain PRA success criteria, and timing of some modeled human actions, due to the increased decay heat level and other factors previously discussed. Further, there is a detailed process for managing the plant during shutdown operations, and the risk impact due to the SPU during these operations is expected to be small.

2.13.3 Conclusion

Luminant Power has assessed risk implications associated with the implementation of the proposed SPU and have concluded that the potential impacts associated with the implementation of the proposed SPU are adequately modeled and/or addressed. The results of

the risk analysis indicate that the risks associated with the proposed SPU are acceptable and do not create the "special circumstances" described in Appendix D of the Standard Review Plan, Chapter 19. Therefore, the risk implications of the proposed SPU are acceptable.

Based on this evaluation, it is concluded that the risk increases due to the impacts of the SPU conditions for internal events, external events, and shutdown operations are very small and within the acceptance criteria of RG 1.174.

2.13.4 References

1. U.S. Nuclear Regulatory Commission Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.
2. U.S. Nuclear Regulatory Commission Review Standard RS-001, "Review Standard for Extended Power Uprates," Revision 0, December 2003.
3. American Society of Mechanical Engineers ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," April 5, 2002.
4. Electric Power Research Institute Technical Reports:
 - a. TR-110398, "Losses of Off-Site Power at US Nuclear Power Plants-Through 1997," Final Report, April 1998.
 - b. TR-110398, "Losses of Off-Site Power at US Nuclear Power Plants-Through 2001," Final Report, April 2002.
 - c. TR-110398, "Losses of Off-Site Power at US Nuclear Power Plants-Through 2004," Final Report, April 2004.
5. Electric Power Research Institute EPRI NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1.
6. U.S. Nuclear Regulatory Commission NUREG-1742, "Perspectives Gained from IPEEE," Draft, April 2001.
7. U.S. Nuclear Regulatory Commission NUREG-0800, Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decision making: General Guidance," Revision 1, November 2002.
8. Nuclear Management and Resources Council, Inc., NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.

Table 2.13-1 Summary of SBO MAAP Cases (Unit 1)				
MAAP Case	Seal LOCA (gpm/pump) ⁽¹⁾	TDAFW Pump Run Time (hours) ⁽²⁾	SPU Core Uncovery Time (hours)	Original Core Uncovery Time (hours) ⁽⁵⁾
SBO1	76	0	1.8 ⁽³⁾	1.8
SBO3	21	0	1.7 ⁽³⁾	2.1
SBO4	182	0	1.8 ⁽³⁾	1.8
WOG Run	480	0	1.8 ⁽³⁾	1.8
CHRIS1	21	0.68	4.3	5
RCP21A	21	4	18.5	20.6
RCP76A	76	4	17.7	19
RCP76D	76	4	17.6	16.8
RCP182A	182	4	16.3	19.0
RCP182B	182	4	9.9	9.9
RCP480A	480	4	8.7	9.6
RCP480C	480	4	2.2	2
RCP480B	480	4	7.5	6.8
SB1E1	SLOCA	0 ⁽⁴⁾	1.0 ⁽⁴⁾	1.25

Notes:

1. RCP Seal LOCA occurs 13 minutes after loss of seal cooling unless otherwise noted.
2. Time to core uncovery with turbine-driven auxiliary feedwater (TDAFW) available varies with the number of steam generators available for depressurization
3. For simplification of recovery application, a time of 1.7 hours to core uncovery will be used.
4. Availability of auxiliary feedwater (AFW) does not influence time to core uncovery due to size of leak. The time to core uncovery was conservatively revised to 1.0 hrs based on the revised MAAP run. At this time, RCS inventory falls to the top of the core prior to accumulator injection. Following accumulator injection, inventory again drops back down to the top of the core at approximately 1.25 hrs. The analysis shows that damage may not occur at the 1-hour time due to the short duration of this configuration. However, the shorter time to initial core exposure was conservatively used in this analysis.

Table 2.13-2

Summary of SBO MAAP Cases (Unit 2)

MAAP Case	Seal LOCA (gpm/pump) ⁽¹⁾	TDAFW Pump Run Time (hours) ⁽²⁾	SPU Core Uncovery Time (hours) ⁽³⁾	Original Core Uncovery Time (hours)
SBO1	76	0	1.8 ⁽⁴⁾	1.8
SBO3	21	0	1.7 ⁽⁴⁾	2.1
SBO4	182	0	1.8 ⁽⁴⁾	1.8
WOG Run	480	0	1.8 ⁽⁴⁾	1.8
CHRIS1	21	0.68	4.3	5
RCP21A	21	4	18.5	20.6
RCP76A	76	4	17.7	19
RCP76D	76	4	16.8	16.8
RCP182A	182	4	16.3	19.0
RCP182B	182	4	9.9	9.9
RCP480A	480	4	8.7	9.6
RCP480C	480	4	2.0	2
RCP480B	480	4	7.5	6.8
SB1E1	SLOCA	0 ⁽⁵⁾	1.0 ⁽⁵⁾	1.25

Notes:

1. RCP Seal LOCA occurs 13 minutes after loss of seal cooling unless otherwise noted.
2. Time to core uncovery with TDAFW available varies with the number of steam generators available for depressurization
3. The values shown in the table are based on expert engineering judgment. The values were derived from a review of these calculations and the information they contained for the Δ76 SG and Δ76 SG plus uprate. In general, if the existing calculations showed that the Δ76 SG plus uprate provided a more restrictive or equivalent time period as the Δ76 SG alone case, that value was chosen. Those results were then compared against the D-5 SG cases and the more restrictive of the two was then chosen as the representative timing for the D-5 SG plus uprate.
4. For simplification of recovery application, a time of 1.7 hours to core uncovery will be used.
5. Availability of AFW does not influence time to core uncovery due to size of leak. The time to core uncovery was conservatively revised to 1.0 hrs based on the revised MAAP run. At this time, RCS inventory falls to the top of the core prior to accumulator injection. Following accumulator injection, inventory again drops back down to the top of the core at approximately 1.25 hrs. The analysis shows that damage may not occur at the 1-hour time due to the short duration of this configuration. However, the shorter time to initial core exposure was conservatively used in this analysis.

**Table 2.13-3
Description of MAPP Run**

Case	Description
SBO1	The turbine driven auxiliary feedwater fails to inject. A LOCA of 76 gpm per Reactor Coolant Pump (4 RCPs) is assumed 45 minutes after loss of all RCP seal cooling.
SBO3	The turbine driven auxiliary feedwater fails to inject. A LOCA of 21 gpm per Reactor Coolant Pump (4 RCPs) is assumed 10 minutes after loss of all RCP seal cooling.
SBO4	The turbine driven auxiliary feedwater fails to inject. A LOCA of 182 gpm per Reactor Coolant Pump (4 RCPs) is assumed 45 minutes after loss of all RCP seal cooling.
WOG Run	The turbine driven auxiliary feedwater fails to inject. A LOCA of 480 gpm per Reactor Coolant Pump (4 RCPs) is assumed 45 minutes after loss of all RCP seal cooling.
CHRIS1	The turbine driven auxiliary feedwater injects with control failure at time 0. SG overfills with the turbine driven auxiliary feedwater failure. A LOCA of 21 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
RCP21A	The SG safeties were forced open after SG overfill when SG pressure reached the setpoint. The SG safeties are assumed to stick open. A LOCA of 21 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
RCP76A	The SG safeties were forced open after SG overfill when SG pressure reached the setpoint. The SG safeties are assumed to stick open. A LOCA of 76 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
RCP76D	The SG safeties are not assumed to stick open after overfill and only 1 SG was used in the depressurization. A LOCA of 76 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
RCP182A	The SG safeties were forced open after SG overfill when SG pressure reached the setpoint. The SG safeties are assumed to stick open. A LOCA of 182 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
RCP182B	The SG safeties are not assumed to stick open after overfill and only 1 SG was used in the depressurization. A LOCA of 182 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
RCP480A	The SG safeties were forced open after SG overfill when SG pressure reached the setpoint. The SG safeties are assumed to stick open. A LOCA of 480 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
RCP480C	The SG safeties are not assumed to stick open after overfill and no SG depressurization. A LOCA of 480 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
RCP480B	The SG safeties are not assumed to stick open after overfill and one SG was used for depressurization. A LOCA of 480 gpm per Reactor Coolant Pump (4 RCPs) is assumed.
SB1E1	Containment Spray pumps are credited but fail at switchover to recirculation. No credit for RHR, CS, SI, or AFW is taken.

Note:

All cases were station blackouts except where noted. The turbine driven auxiliary feedwater was assumed to be available until SG overfill, unless otherwise noted. In all cases power was assumed to be available for 4 hours. At the time of battery depletion, i.e. at 4 hours, all calculations assumed loss of level control in the SGs. The SG atmospheric relief valves were assumed to fail closed at the time of battery depletion in all runs.

APPENDIX A CODES AND METHODS

The following is provided in support of the principal computer codes (See Licensing Report (LR) Table 2.8.5.0-6) used to support the Comanche Peak Nuclear Power Plant (CPNPP) stretch power uprate (SPU). Westinghouse COBRA/TRAC is not addressed since it is part of a separate best-estimate loss-of-coolant-accident (BELOCA) submittal.

Computer Code Description
<p>ANC</p> <p>ANC is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, and so forth. In addition, 3-D ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.</p> <p><u>Reference:</u></p> <ol style="list-style-type: none">1. WCAP-10965, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986. <p>Date of NRC Acceptance: June 23, 1986, Carl Berlinger to E. P. Rahe</p> <p>Safety Evaluation Report (SER) Conditions: There are no SER restrictions applicable to this application.</p>
<p>DORT/BUGLE-96</p> <p>The DORT discrete ordinates transport module of the DOORS 3.1 code package, in conjunction with the BUGLE-96 cross-section library, is used to determine the neutron flux and gamma-ray heating rate environment. This code and the associated cross-section library have been used by Westinghouse to calculate vessel fluences and reactor internals heating rates for other projects that have been submitted to, and approved by, the Nuclear Regulatory Commission (NRC). Furthermore, these calculation tools are specified in Regulatory Guide 1.190 for this type of work.</p> <p><u>References:</u></p> <ol style="list-style-type: none">1. RSICC Computer Code Collection CCC-650, "DOORS 3.1, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.2. RSICC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.3. Regulatory Guide RG-1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001. <p>Date of NRC Acceptance: There is no formal NRC acceptance.</p> <p>SER Conditions: N/A</p>

Computer Code Description

FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad, uranium dioxide fuel rod and the transient heat flux at the surface of the cladding using the time-dependent input of nuclear power and reactor coolant parameters of pressure, flow, temperature and density. The code uses a fuel model containing a sufficiently large number of radial-spatial nodes to adequately model very fast transients. FACTRAN uses material properties, which are a function of temperature, and has the capability to perform a detailed fuel-to-cladding gap heat transfer calculation. Two sets of transient equations, representing an energy balance and the heat conduction for each radial node, are solved simultaneously. The solutions to these equations consist of the heat flux at the surface of the fuel rod and the fuel rod temperatures at the end of each time step.

Reference:

1. WCAP-7908, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO₂ Fuel Rod," December 1989.

Date of NRC Acceptance: September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))

SER Conditions and Justification

1. *"The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."*

Justification

The bounding initial fuel temperatures for transients were calculated using the PAD 4.0 computer code (see WCAP-15063). As indicated in WCAP-15063, the method of determining uncertainties for PAD 4.0 fuel temperatures has been approved by the NRC.

2. *"Table 2 presents the guidelines used to select initial temperatures."*

Justification

In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: Loss of Flow, Locked Rotor, and Rod Ejection. The assumed fuel temperatures, which were based on bounding temperatures calculated using the PAD 4.0 computer code (see WCAP-15063), include uncertainties and are conservatively high.

3. *"The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."*

Justification

The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the Rod Cluster Control Assembly (RCCA) Withdrawal from a Subcritical Condition transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for departure from nucleate boiling (DNB) concerns) and a low constant value is assumed to maximize transient fuel temperatures. For the RCCA Ejection transients, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate cladding collapse onto the fuel pellet.

Computer Code Description

4. *"...the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."*

Justification

Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed.

5. *"The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."*

Justification

At least 6 concentric rings were assumed in FACTRAN for each transient analyzed.

6. *"Although time-independent mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-dependent mechanical behavior (e.g., plastic deformation) is not considered in the code. ...for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet."*

Justification

The two transients that were analyzed with FACTRAN for CPNPP (RCCA Withdrawal from a Subcritical Condition (Final Safety Analysis Report (FSAR) 15.4.1) and RCCA Ejection (FSAR 15.4.8)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the RCCA Withdrawal from a Subcritical Condition transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the RCCA Ejection transient, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.

7. *"The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When $(T(\text{centerline}) - T(\text{surface}))$ is on the order of 3,000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable."*

Justification

The condition of concern ($T(\text{centerline}) - T(\text{surface})$ on the order of 3,000°F) is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA Ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA Ejection transient.

Computer Code Description

FORCE2 (See also MULTIFLEX)

The FORCE2 program calculates the hydraulic forces that the fluid exerts on the vessel internals in the vertical direction by utilizing a detailed geometric description of the vessel components and the transient pressures, mass velocities, and densities computed by the MULTIFLEX code. The analytical basis for the derivation of the mathematical equations employed in the FORCE2 code is the conservation of linear momentum (one-dimensional). Note that the computed vertical forces in the LOCA forces analyses do not include body forces on the vessel internals, such as dead-weight or buoyancy. The dead-weight and other factors are part of the dynamic system model to which the LOCA forces are provided as an external load. When the vertical forces on the reactor pressure vessel internals are calculated, pressure differential forces, flow stagnation on, unrecoverable orifice losses across, and friction losses on, the individual components are considered. These force types are then summed together, depending upon the significance of each, to yield the total vertical force acting on a given component.

References:

1. WCAP-8708 and WCAP-8709, "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," September 1977.
2. WCAP-8252, Revision 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," May 1977.

Date of NRC Acceptance: See MULTIFLEX.

SER Conditions: See MULTIFLEX.

LATFORC (See also MULTIFLEX)

The LATFORC computer code utilizes MULTIFLEX generated field pressures, together with geometric vessel information (component radial and axial lengths), to determine the horizontal forces on the vessel wall and core barrel. The LATFORC code represents the vessel region with a model that is consistent with the model used in the MULTIFLEX blowdown calculation. The downcomer annulus is subdivided into cylindrical segments, formed by dividing this region into circumferential and axial zones. The results of the MULTIFLEX/LATFORC analysis of the horizontal forces are typically stored on magnetic tape and are calculated for the initial 500 msec of the blowdown transient. These forcing functions serve as required input in determining the resultant mechanical loads on primary equipment and loop supports, vessel internals, and fuel grids.

References:

1. WCAP-8708 and WCAP-8709, "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," September 1977.
2. WCAP-8252, Revision 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," May 1977.

Date of NRC Acceptance: See MULTIFLEX.

SER Conditions: See MULTIFLEX.

Computer Code Description

LOFTRAN

The LOFTRAN computer program is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. LOFTRAN simulates up to four-loop systems by modeling the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and pressurizer. The pressurizer heaters, spray, relief, 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 sides of the steam generators utilize a homogeneous, saturated mixture for the thermal transients, and a water level correlation for indication and control. The reactor protection system simulation includes reactor trips on neutron flux, overpower and overtemperature ΔT , high and low pressure, low flow, and high pressurizer water level. Control systems, including rod control, steam dump, feedwater control, and pressurizer pressure controls are also simulated. The safety injection system, including the accumulators, is also modeled.

LOFTRAN is a versatile program suited to accident evaluation and control studies as well as parameter sizing. It is also used in performing loss of normal feedwater anticipated transient without scram (ATWS) and loss-of-load ATWS evaluations and control systems analysis.

Reference:

1. WCAP-7907 and WCAP-7907, "LOFTRAN Code Description," April 1984.

Date of NRC Acceptance: July 29, 1983 (SER from C. O. Thomas (NRC) to E. P. Rahe (W)).

SER Conditions and Justification:

"LOFTRAN is used to simulate plant responses to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis."

MULTIFLEX (See also LATFORC, FORCE2, and THRUST)

The analysis for LOCA hydraulic forces used the NRC-accepted MULTIFLEX computer code, which is the current Westinghouse analytical tool used for analyzing LOCA hydraulic forces. The code was used to generate the transient hydraulic forcing functions on the vessel and internals. This code was previously used for LOCA hydraulic forces analyses.

MULTIFLEX 3.0 is an engineering design tool that is used to analyze the coupled fluid-structural interactions in a PWR system during the transient following a postulated pipe rupture in the main reactor coolant system (RCS). The thermal-hydraulic portion of the MULTIFLEX code is based on the one-dimensional homogeneous model expressed in a set of mass, momentum, and energy conservation equations. These equations are quasi-linear, first order, partial differential equations solved by the method of characteristics. The employed numerical method utilizes an explicit time scheme along the respective characteristics. MULTIFLEX considers the interaction of the fluid and structure simultaneously, whereby the mechanical equations of vibration are solved through the use of the modal analysis technique. MULTIFLEX 3.0 generates the input for the post-processing codes LATFORC, FORCE2, and THRUST. All applicable MULTIFLEX SER items have been addressed in this application.

References:

1. WCAP-8708 and WCAP-8709, "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," September 1977.
2. WCAP-8252, Revision 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," May 1977.

Computer Code Description

3. WCAP-9735, Revision 2 and WCAP-9736, Revision 1, "MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model," February 1998.
4. WCAP-15029, Revision 0 and WCAP-15030, Revision 0, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," January 1999.

NRC Acceptance

U.S. NRC review and approval for the use of MULTIFLEX (1.0), LATFORC and FORCE2 codes documented in Reference 1 for PWR LOCA hydraulic forces calculations was originally provided in Reference 2. Reference 3 is an example of U.S. NRC reviewed and approved application of MULTIFLEX to steam generator LOCA hydraulic force calculations, as provided in Reference 4. Reference 5 is an example of the U.S. NRC reviewed and approved application of MULTIFLEX to the analysis of fuel assembly LOCA hydraulic force calculations, as provided in Reference 6. Reference 7 documents the changes in MULTIFLEX modeling features from version 1.0 to version 3.0. References 8 and 9 were supplemental submittals on behalf of Beaver Valley Unit 2 regarding the use of MULTIFLEX 3.0 in the LOCA hydraulic forces analysis. The MULTIFLEX 3.0 analysis was subsequently accepted as the analysis of record for Beaver Valley Unit 2. Subsequently, MULTIFLEX 3.0 was accepted by the U.S. NRC as part of the methodology to confirm acceptable baffle-barrel-bolting patterns, Reference 10, in the Reference 11 evaluation report. MULTIFLEX 3.0 was again accepted by the U.S. NRC as part of the methodology to confirm control rod insertion for D. C. Cook Units 1 and 2, Reference 12, in the Reference 13 evaluation report. Reference 14 documents the STHRUST code which has been used in loop piping LOCA hydraulic forces analyses since before the MULTIFLEX code was developed. There is no specific acceptance date for the THRUST code (the S was dropped when BLOWDN-2 and MULTIFLEX replaced SATAN in providing the hydraulic data to THRUST).

References:

1. WCAP-8708, "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," September, 1977.
2. Letter, John F. Stolz (U.S. NRC) to C. Eicheldinger (Westinghouse), "Evaluation of Westinghouse Topical Reports WCAP-8708 and WCAP-8709," June 17, 1977 (Enclosure – Topical Evaluation Report).
3. WCAP-7832, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions," April 1978.
4. Letter, John F. Stolz (U.S. NRC) to C. Eicheldinger (Westinghouse), "Safety Evaluation of WCAP-7832 and WCAP-8709," March 2, 1978 (Enclosure – Safety Evaluation Report).
5. WCAP-9401, "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," August 1981.
6. Letter, Robert L. Tedesco (U.S. NRC) to T. M. Anderson (Westinghouse), "Acceptance for Referencing Topical Report WCAP-9401/WCAP-9402," May 7, 1981.
7. WCAP-9735, Revision 2 and WCAP-9736, Revision 1, "MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model," February 1998.

Computer Code Description

8. WCAP-11004/WCAP-11005, "Comparison of Data for Beaver Valley Power Station, Unit 2 with WCAP-9735 Data, Prepared for NRC Review in Conjunction with Review of WCAP-9735, Docket No. 50-412," November 1985.
9. WCAP-11522/WCAP-11523, "Response to NRC Questions on the LOCA Hydraulic Forces Analysis of the Beaver Valley Power Station, Unit 2, Prepared for NRC Review in Conjunction with Review of WCAP-9735, Docket No. 50-412," June 1987.
10. WCAP-15029, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," December 1998.
11. Letter, T. H. Essig (U.S. NRC) to Lou Liberatori (WOG), Safety Evaluation of Topical Report WCAP-15029, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," (TAC No. MA1152), November 10, 1998 (Enclosure 1 – Safety Evaluation Report).
12. WCAP-15245, "Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2," May 28, 1999.
13. Letter, John F. Stang (U.S. NRC) to Robert P. Powers (Indiana Michigan Power Company), "Issuance of Amendments – Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474)," December 23, 1999.
14. WCAP-8252, Revision 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," May 1977.

SER Conditions:

Note there are no specific SER restrictions on MULTIFLEX 3.0, LATFORC, FORCE2, and THRUST. However, the analyses have complied with the applicable SER restrictions on MULTIFLEX 1.0 and the Reference 10 methodology, including the use of the conservative 1-millisecond break opening time.

NOTRUMP/SBLOCTA (LOCTA-IV)

The approved codes for Appendix K small-break LOCA analyses are NOTRUMP and SBLOCTA. The NOTRUMP computer code is a one-dimensional general network code consisting of a number of advanced features. Among these features is the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flow limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat-transfer correlations. Additional features of the code are condensation heat-transfer model applied in the steam generator region, loop seal model, core reflux model, flow regime mapping, etc.

The SBLOCTA computer code is used to model the fuel rod response to the small-break LOCA transient. It models two rods in the hot assembly (hot and average), modeling simultaneous radial and axial conduction. Other modeling features include assembly blockage model due to cladding swell, and rupture and zirc/water reaction.

NOTRUMP is used to model the thermal-hydraulic behavior of the system and thereby obtain time-dependent values of various core region parameters, such as system pressure, temperature, fluid levels, and flow rates. These are provided as boundary conditions to SBLOCTA. SBLOCTA then uses these conditions and various hot channel inputs to calculate the rod heatup and ultimately, the peak cladding temperature (PCT) for a given transient. Additional variables calculated by SBLOCTA are cladding pressure, strain, and oxidation.

All applicable SER restrictions and limitations have been addressed in this application.

Computer Code Description

References:

1. WCAP-10079, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.
2. WCAP-10054, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
3. WCAP-10054, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.
4. WCAP-8301, "LOCTA-IV Program: Loss of Coolant Transient Analysis," June 1974.

Dates of NRC Acceptance:

WCAP-10079, May 23, 1985; WCAP-10054, May 21, 1985; WCAP-10054, August 12, 1996.

SER Conditions: WCAP-10054

SER Wording (Page 8)

"To assure the validity of this application, the bubble diameter should be on the order of 10-1-2 cm. As long as steam generator tube uncover (concurrent with a severe depressurization rate) does not occur, this option is acceptable."

SER Compliance

Westinghouse complies with this restriction for all Appendix K licensing basis calculations. Typical Appendix K calculations do not undergo a significant secondary side system depressurization in conjunction with steam generator tube uncover due to the modeling methodology utilized.

SER Wording (Page 14)

"The two phase multiplier used is the Thom modification of the Martinelli-Nelson correlation. This model is acceptable per 10 CFR Part 50 Appendix K for LOCA analysis at pressure above 250 psia".

SER Compliance

The original NOTRUMP model was limited to no less than 250 psia since the model, as contained in the NOTRUMP code, did not contain information below this range. Westinghouse extended the model to below 250 psia, as allowed by Appendix K paragraph I-C-2, and reported these modifications to the NRC via the 1995 annual reporting period (NSD-NRC-96-4639).

SER Wording (Pages 16-17)

"Axial heat conduction is not modeled." and "Deletion of clad axial heat conduction maximizes the peak clad temperature."

SER Compliance

The Westinghouse small-break LOCA is comprised of two computer codes, the NOTRUMP code (which performs the detailed system wide thermal hydraulic calculations) and the LOCTA code (which performs the detailed fuel rod heatup calculations). The NOTRUMP code does not model axial conduction in the fuel rod and, therefore, complies. The LOCTA code has always accounted for axial conduction as is clearly stated in WCAP-14710, which supplements the original NOTRUMP documentation.

Computer Code Description

SER Wording (Page 21)

"The standard continuous contact model is not appropriate for vertical flow,..."

SER Compliance

The standard continuous contact flow links are not utilized when modeling vertical flow in the Appendix K NOTRUMP Evaluation Model analyses. Therefore, compliance is demonstrated.

SER Wording (Page 7 of enclosure 2)

"Per generic letter 83-35, compliance with Action Item II. K.3.31 may be submitted generically. We require that the generic submittal include validation that the limiting break location has not shifted away from the cold legs to the hot or pump suction legs."

SER Compliance

Westinghouse submitted WCAP-11145 in support of Generic Letter 83-35 Action Item 11.K.3.31. As part of this effort, verification was provided which documented that the cold leg break location remains limiting.

WCAP-10054, Addendum 2, Revision 1

SER Wording (Page 3)

"It is stated in Ref. 5 that the range of injection jet velocities used in the experiments brackets the corresponding rates in small break LOCAs for Westinghouse plants and that the model will be used within the experimental range. Also in References 1 and 5 Westinghouse submitted analyses demonstrating that the condensation efficiency is virtually independent of RCS pressure and state that the COSI model will be applied within the pressure range of 550 to 1,200 psia."

SER Compliance

The coding implementation of the COSI model correlation in the NOTRUMP model restricts the application of the COST condensation model to a default pressure range of 550 to 1,200 psia and limits the injection flow rate to a default value of 40 lbm/sec-loop. The value of 40 lbm/sec-loop corresponds to the 30 ft/sec velocity utilized in the COSI experiments. As such, the default NOTRUMP implementation of the COSI condensation model complies with the applicable SER restrictions.

NUPIPE-SWPC

The NUPIPE-SWPC program is used to perform detailed pipe stress analysis. This program is designed to perform analyses in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III Nuclear Power Plant Components and the American National Standards Institute (ANSI)/ASME B31.1 Power Piping Code.

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for use in evaluating piping systems in accordance with the ASME B&PV Code, Section III and ANSI/ASME B31.1 Power Piping Codes. This computer code has been used by Shaw Stone & Webster on many recent SPU projects, as well as being used for past pipe stress evaluations efforts at CPNPP.

Computer Code Description

ORIGEN-S

Industry code ORIGEN-S is part of the SCALE 4.3 suite of codes that was developed for the NRC by Oak Ridge National Laboratory. ORIGEN-S calculates the isotopic composition as a function of time for nuclear reactor fuel irradiation and decay models. The code output includes isotopic concentrations and radiation source spectra and strengths. For CPNPP, ORIGEN-S is used to address the impact of the SPU on radiological equipment qualification. Specifically, it is used to calculate the gamma energy release rates for selected post-accident recirculating fluids for various time periods after the LOCA (which are then used to establish SPU dose rate scaling factors).

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for the use discussed above. It has been used extensively in prior licensing applications, including SPUs, and its results accepted by the NRC.

PAD 4.0

The NRC-approved PAD code, with NRC-approved models for in-reactor behavior, is used to calculate the fuel rod performance over its irradiation history. PAD is the principal design tool for evaluating fuel rod performance. PAD iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power. Fuel rod design and safety analyses are based on updated values (up to 100-percent helium gas release) for the integral fuel burnable absorber (IFBA) helium gas release model.

PAD is a best-estimate fuel rod performance model. In most cases, the design criterion evaluations are based on a best-estimate plus uncertainties approach. A statistical convolution of individual uncertainties due to design model uncertainties and fabrication dimensional tolerances is used. As-built dimensional uncertainties are measured for some critical inputs (e.g., fuel pellet diameter), and when available, can be used in lieu of the fabrication uncertainties.

References:

1. WCAP-12610, "VANTAGE + Fuel Assembly Reference Core Report," April 1995.
2. WCAP-10851, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," August 1988.
3. WCAP-15063, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000.

Date of NRC Acceptance:

Letter from S. Richards (NRC) to H. A. Sepp (Westinghouse), "Safety Evaluation Related to Topical Report WCAP-15063, Revision 1, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," (TAC No. MA2086), April 24, 2000.

SER Conditions: There are no SER restrictions applicable to this application.

Computer Code Description

PERC2

Stone and Webster (S&W) proprietary computer code PERC2 calculates the radiological dose consequences at the site boundary and Control Room following a postulated accident. It includes the following major features:

- Provision of time-dependent releases from the RCS to the containment atmosphere.
- Provision for airborne radionuclides other than noble gas and iodine, including daughter ingrowth.
- Provision for calculating organ doses other than thyroid.
- Provisions for tracking time-dependent inventories of all radionuclides in all control regions of the plant model.
- Provision for calculating instantaneous and integrated gamma radiation source strengths as well as activities for the inventoried radionuclides to permit direct assessment for equipment qualification and vital area access.

For CPNPP, it is used to address the impact of SPU on radiological equipment qualification and vital area access doses. Specifically, it is used to calculate energy release rates and integrated gamma energy releases versus time for various post-accident radiation sources, and relative gamma and beta doses inside containment, which are then used to establish SPU dose scaling factors.

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for the use discussed above. It has been used extensively in prior licensing applications, including SPUs, and its results accepted by the NRC.

PC-PREPS

PC-PREPS is a PC-based computer program that performs a complete structural analysis, performing an American Institute of Steel Construction (AISC) code check, weld qualification, and baseplate/anchor bolt qualifications.

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for use in evaluating piping systems in accordance with the ASME B&PV Code, Section III and ANSI/ASME B31.1 Power Piping Codes. This computer code has been used by Shaw Stone & Webster on many recent SPU projects, as well as being used for past pipe stress evaluations efforts at CPNPP.

PHOENIX-P

PHOENIX-P is a two-dimensional, multi-group transport theory computer code. The nuclear cross-section library used by PHOENIX-P contains cross-section data based on a 70-energy-group structure derived from ENDF/B-VI files. PHOENIX-P performs a two-dimensional 70-group nodal flux calculation which couples the individual subcell regions (pellet, cladding, and moderator) as well as surrounding rods via a collision probability technique. This 70-group solution is normalized by a coarse energy group flux solution derived from a discrete ordinates calculation. PHOENIX-P is capable of modeling all cell types needed for PWR core design applications.

Reference:

1. WCAP-11596, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

Date of NRC Acceptance: May 17, 1988, Ashok Tadani to W. J. Johnson

SER Conditions: There are no SER restrictions applicable to this application.

Computer Code Description

PILUG-PC

PILUG-PC is a PC-based stress analysis program used to calculate stress intensity at the junction of a rectangular attachment perpendicular to round pipe.

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for use in evaluating piping systems in accordance with the ASME B&PV Code, Section III and ANSI/ASME B31.1 Power Piping Codes. This computer code has been used by Shaw Stone & Webster on many recent SPU projects, as well as being used for past pipe stress evaluations efforts at CPNPP.

SW-QADCGGP

SW-QADCGGP is a variant of the QAD point kernel shielding program originally written at the Los Alamos Scientific Laboratory by R. E. Malenfant. The SW-QADCGGP version implements combinatorial geometry and the geometric progression build-up factor algorithm. For CPNPP, it is used to develop dose/dose rate scaling factors for-shielded configurations to support the evaluation of impact of the SPU on radiological equipment qualification, vital area access, and shielding adequacy.

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for the use discussed above. It has been used extensively in prior licensing applications, including SPUs, and its results accepted by the NRC.

THRUST (See also MULTIFLEX)

The THRUST program calculates the hydraulic forces that the fluid exerts on the reactor coolant loop. The THRUST code uses the MULTIFLEX LOCA pressure transient as input in the calculation of the loop forces. In the THRUST computer code, the loop piping is represented by a series of control volumes. The pressure forces are calculated by THRUST wherever there are changes in either loop area or direction. The LOCA loop forces are then transmitted to the appropriate structural analysis group where they are then combined with the other design-basis loads (i.e., seismic, thermal and system shaking loads) where they are used to qualify the reactor coolant loops under the design-basis loads.

References:

1. WCAP-8708 and WCAP-8709, "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," September 1977.
2. WCAP-8252, Revision 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," May 1977.

Date of NRC Acceptance: See MULTIFLEX

SER Conditions: See MULTIFLEX

TWINKLE

TWINKLE is a neutron kinetics code that solves the multi-dimensional, two-group transient diffusion equations using a finite-difference technique. The code contains a detailed six-region fuel-clad-coolant transient heat transfer model at each spatial point for calculating Doppler and moderator feedback effects. The code handles up to 8,000 spatial points in one-, two- or three- dimensional rectangular geometry.

Computer Code Description

Reference:

1. WCAP-7979, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code," January 1975.

Date of NRC Acceptance: July 29, 1974 (SER from D. B. Vassallo (U.S. Atomic Energy Commission) to R. Salvatori (Westinghouse))

SER Conditions & Justification: There are no conditions, restrictions, or limitations cited in the TWINKLE SER.

VIPRE

VIPRE-01 (VIPRE) is a three-dimensional subchannel code that has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core and hot channels. The VIPRE code is based on a knowledge and understanding of the heat transfer and hydrodynamic behavior of the coolant flow and the mechanical characteristics of the fuel elements. The use of the VIPRE analysis provides a realistic evaluation of the core performance and is used in the thermal-hydraulic analysis.

The VIPRE core model as approved by the NRC (Reference 1) is used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core under all expected operating conditions. The VIPRE code is described in detail in Reference 2, including discussion on code validation with experimental data. The VIPRE modeling method is described in Reference 1, including empirical models and correlations used. The effect of crud on the flow and enthalpy distribution in the core is not directly accounted for in the VIPRE evaluations. However, conservative treatment by the VIPRE modeling method has been demonstrated to bound this effect in DNBR calculations.

References:

1. WCAP-14565 and WCAP-15306, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.
2. NP-2511-CCM-A, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Core, Volume 1-3 (Revision 3, August 1989, Volume 4 (April 1987)," Electric Power Research Institute, C. W. Stewart et al.

Date of NRC Acceptance:

Letter from T. H. Essig (NRC) to H. Sepp (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis," (TAC No. M98666)," January 19, 1999.

SER Conditions & Justification

1. *"Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."*

Justification

The WRB-2 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the CPNPP 17x17 VANTAGE+ fuel type. The use of the WRB-2 DNB correlation for VANTAGE+ fuel was approved September 1985 (Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-10444, VANTAGE+ Fuel Assembly," Reference 19). WCAP-12610 extended the use of the WRB-2 correlation to VANTAGE+ fuel and was approved July 1, 1991 (Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610 VANTAGE+ Fuel Assembly Reference Core Report," Reference 20).

Computer Code Description

The use of the plant-specific hot channel factors and other fuel dependent parameters in the DNB analysis for the CPNPP VANTAGE+ fuel were justified using the same methodologies as for previously approved safety evaluations of other Westinghouse four-loop plants using the same fuel design.

2. *"Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."*

Justification

The core boundary conditions for the VIPRE calculations for the CPNPP fuel are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272.

3. *"The NRC Staff's generic SER for VIPRE (Reference 2 of the SER) set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."*

Justification

As discussed in response to Condition 1, the WRB-2 correlation with a limit of 1.17 was used for the DNB analyses of the CPNPP fuel. For conditions where WRB-2 is not applicable, the W-3 DNB correlation was used with a limit of 1.30 (1.45, for pressures between 500 psia and 1,000 psia).

4. *"Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE (Reference 2 of the SER) did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."*

Justification

For application to CPNPP safety analysis, the usage of VIPRE in the post-critical heat flux region is limited to the PCT calculation for the locked rotor transient. The calculation demonstrated that the PCT in the reactor core is well below the allowable limit to prevent cladding embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565 and included the following conservative assumptions:

- DNB was assumed to occur at the beginning of the transient.
- Film boiling was calculated using the Bishop-Sandberg-Tong correlation.
- The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction.

Computer Code Description

Conservative results were further ensured with the following input:

- Fuel rod input based on the maximum fuel temperature at the given power
- The hot spot power factor was equal to or greater than the design linear heat rate

Uncertainties were applied to the initial operating conditions in the limiting direction.

WATHAM-PC

1. General-PC Program Description

The WATHAM is a PC-based computer program that is used to determine the flow-induced forcing functions acting on piping systems due to waterhammer. These forcing functions may then be used as input to a structural dynamic analysis such as a NUPIPE-SWPC program run. WATHAM is applicable to a waterhammer problem or more generally, to any unsteady, incompressible fluid flow. These events may be caused by normal or abnormal operational changes of piping components, such as the startup and trip of pumps or the rapid opening and closing of valves.

2. Program Qualification

Using an approved Quality Assurance Program, this computer program has been verified and validated and shown to be accurate and acceptable for use in evaluating piping systems in accordance with the ASME B&PV Code, Section III and ANSI/ASME B31.1 Power Piping Codes. This computer code has been used by Shaw Stone & Webster on recent SPU projects, as well as being used for past piping evaluation efforts at CPNPP.