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CP-200800125
Log # TXX-08014

Ref. # 10CFR50.90

February 28, 2008

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NOS. 50-445 AND 50-446
SUPPLEMENT TO LICENSE AMENDMENT REQUEST (LAR) 07-004
REVISION TO THE OPERATING LICENSE AND TECHNICAL SPECIFICATION 1.0, "USE AND APPLICATION" TO REVISE RATED THERMAL POWER FROM 3458 MWT TO 3612 MWT. (TAC NOS. MD6615 AND MD6616)

REFERENCES:
1. Letter logged TXX-07106 dated August 28, 2007 from Mike Blevins to the NRC submitting License Amendment Request (LAR) 07-004, proposing revisions to the Operating Licenses and to Technical Specifications 1.0, "USE AND APPLICATION" to revise rated thermal power from 3458 MWT to 3612 MWT
2. Letter logged TXX-08008 dated January 10, 2008 from Mike Blevins to the NRC submitting a supplement to License Amendment Request (LAR) 07-004
3. Letter logged TXX-08013 dated January 31, 2008 from Mike Blevins to the NRC submitting a supplement to License Amendment Request (LAR) 07-004
4. Letter logged TXX-08031 dated February 21, 2008 from Mike Blevins to the NRC submitting a supplement to License Amendment Request (LAR) 07-004

Dear Sir or Madam:

Per Reference 1, Luminant Generation Company LLC (Luminant Power) requested changes to the Comanche Peak Steam Electric Station, herein referred to as Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2 Operating Licenses and to Technical Specification 1.0, "USE AND APPLICATION" to revise rated thermal power from 3458 MWT to 3612 MWT. Luminant Power supplemented that request by responding to NRC Requests for Additional Information (RAI) per References 2, 3 and 4.

Per Reference 2 Luminant Power committed to provide a revised analysis of the loss of external electrical load/turbine trip event which demonstrates that the overpressure criteria continue to be met when the second, safety-grade reactor trip signal is credited (Commitment No. 3435228) by February 29, 2008. The attachment to this letter provides the information revising section 2.8.4.2 of WCAP-16840-P submitted in Reference 1 as required by the commitment.

A001
NRR

In accordance with 10CFR50.91(b), Luminant Power is providing the State of Texas with a copy of this proposed amendment supplement.

This communication closes commitment No. 3435228 and contains no new license basis commitments regarding CPNPP Units 1 and 2.

Should you have any questions, please contact Mr. J. D. Seawright at (254) 897-0140.

I state under penalty of perjury that the foregoing is true and correct.

Executed on February 28, 2008.

Sincerely,

Luminant Generation Company LLC

Mike Blevins

By: Fred W. Madden

Fred W. Madden
Director, Oversight & Regulatory Affairs

Attachment - Response to Request for Additional Information regarding Reactor Systems Branch Question 6 and 9

c - E. E. Collins, Region IV
B. K. Singal, NRR
Resident Inspectors, Comanche Peak

Ms. Alice Rogers
Environmental & Consumer Safety Section
Texas Department of State Health Services
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Attachment to TXX-08014

Response to Request for Additional Information Regarding Reactor Systems Branch Question 6 and 9

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 1). 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 (PORV) 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 (PSV), 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. The LOL/TT analysis described 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, demonstrates the adequacy of the RCS overpressure protection capability. An additional LOL/TT analysis, in which the automatic reactor trip on turbine trip and high pressurizer pressure trip functions are conservatively neglected, is summarized in this section to help demonstrate the adequacy of the RCS overpressure protection capability at CPNPP Units 1 and 2 for the SPU program.

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

One LOL/TT case from full-power conditions was analyzed for each unit to demonstrate the adequacy of the RCS overpressure protection. NSSS power, RCS temperature and pressure were assumed to be at their nominal values consistent with steady-state, full-power operation. The reactor coolant minimum measured flow was modeled.

The LOL/TT transient was conservatively analyzed with minimum reactivity feedback (beginning of core life), the least-negative Doppler power coefficient and a 0 pcm/°F moderator temperature coefficient. Minimum reactivity conditions are conservative since they maintain reactor power until the time of reactor trip, which exacerbates the maximum RCS pressure.

Manual rod control was modeled. If the reactor is in automatic rod control, the control rod banks would be driven into the core prior to reactor trip, thereby reducing the severity of the transient.

The LOL/TT transient was analyzed without pressurizer pressure control. The pressurizer PORVs and sprays were assumed inoperable in order to conservatively maximize the RCS pressure increase. The main steam safety valves (MSSVs) and pressurizer safety valves were assumed to be operable.

A total PSV setpoint tolerance of +1% was accounted for in the analysis in addition to a 0.9% setpoint shift and a 1.05-second purge time delay, which is associated with the existence of PSV water-filled loop seals.

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

Only the overtemperature N-16 (OTN-16) reactor trip function was assumed to be operable. As shown in the results, the high pressurizer pressure reactor trip, although not credited, would actuate before the OTN-16 reactor trip setpoint is reached. The high pressurizer pressure reactor trip function was assumed to be inoperable so as to show that the second safety-grade reactor trip signal would initiate reactor trip, thereby demonstrating the adequacy of the RCS overpressure protection capability.

The MSSV model included 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.

Maximum (10-percent) steam generator tube plugging is assumed since it maximizes the RCS temperature transient following event initiation.

For the LOL/TT RCS overpressure protection analysis, the primary system pressure must remain below 110% of the design pressure (an RCS pressure limit of 2,748.2 psia) at all times during the transient. Demonstrating that the primary pressure limit is 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, as it relates to designing the RCS and associated auxiliary, control, and protection systems 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 (AOOs).
- GDC-31, as it relates to designing the RCPB with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fractures is minimized.
- 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) require that RCS SRVs meet Three Mile Island (TMI) Action Plan Items II.D.1 and II.D.3 of NUREG-0737.
- 10 CFR 52.47(a)(8) provides the requirement for design certification reviews to comply with the technically relevant portions of the TMI requirements in 10 CFR 50.34(f).
- 10 CFR 52.79(a)(17) provides the requirements for combined license (COL) applications to comply with the technically relevant information in 10 CFR 50.34.

- 10 CFR 52.47(b)(1) requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

- 10 CFR 52.80(a) requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

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) (requirements 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi)), (2) design certification applications (not applicable) (requirement 10 CFR 52.47(a)(8)), and (3) COL applications (not applicable) (requirements 10 CFR 52.79(a)(17), 10 CFR 52.47(b)(1) and 10 CFR 52.80(a)).

2.8.4.2.2.3 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 power operated relief valves (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 the LOL/TT transient (Reference 2).

In addition, the allowable power levels with inoperable main steam safety valves have been determined and specified in Technical Specification Table 3.7.1-1. This table is being revised as described in License Amendment Request (LAR) 07-006 submitted in a letter dated August 16, 2007, logged 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. To preclude secondary side overpressurization during 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 calculated time sequence of events for both units is listed in Table 2.8.4.2-1 and the limiting values for each unit are presented in Table 2.8.4.2-2.

The transient response of the LOL/TT RCS overpressurization analysis is plotted in Figures 2.8.4.2-1 through 2.5.4.2-6. The following results discussion is applicable to both units.

The LOL/TT event is initiated by tripping the turbine. Because of the beginning of life reactivity model, the nuclear power remained essentially constant at full power until the reactor was tripped on the overtemperature N-16 reactor trip function. Due to the turbine trip, the primary and secondary side pressures rapidly increase. The PSVs actuate to maintain the primary side pressure below 110 percent of the design value. The MSSVs also actuate to maintain the secondary side pressure below 110 percent of the design pressure. The peak pressurizer water volume remains below the total volume of the pressurizer, demonstrating that this event does not generate a more serious plant condition.

The results of the LOL/TT analyses documented in LR subsection 2.8.5.2.1 and in this section 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, the conclusions of the Overpressure Protection Report referenced in the FSAR remain valid.

Table 2.8.4.2-3 provides the maximum allowable power range neutron flux high setpoints with inoperable MSSVs described in LAR 07-006 submitted in a letter dated August 16, 2007, logged 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.

2.8.4.2.3 References

1. WCAP-7769, "Topical Report Overpressure Protection for Westinghouse PWR," October 1971.

2. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurizer Water Reactor Non-LOCA Safety Analyses," April 1999.

Table 2.8.4.2-1 Time Sequence of Events – Loss of External Electrical Load and/or Turbine Trip RCS Overpressurization Analysis		
Event	Time (sec)	
	Unit 1	Unit 2
Loss of Electrical Load/Turbine Trip Occurs	0.0	0.0
High Pressurizer Pressure Reactor Trip Setpoint Reached (not credited)	5.1	5.4
Peak RCS Pressure Occurs	7.6	7.8
OTN-16 Reactor Trip Setpoint Reached	11.3	10.9
Rods Begin to Drop	13.3	12.9

Table 2.8.4.2-2 Limiting Results – Loss of External Load and/or Turbine Trip RCS Overpressurization Analysis		
Unit 1	Peak RCS Pressure (psia)	2733.8
Unit 2	Peak RCS Pressure (psia)	2744.2

Table 2.8.4.2-3 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

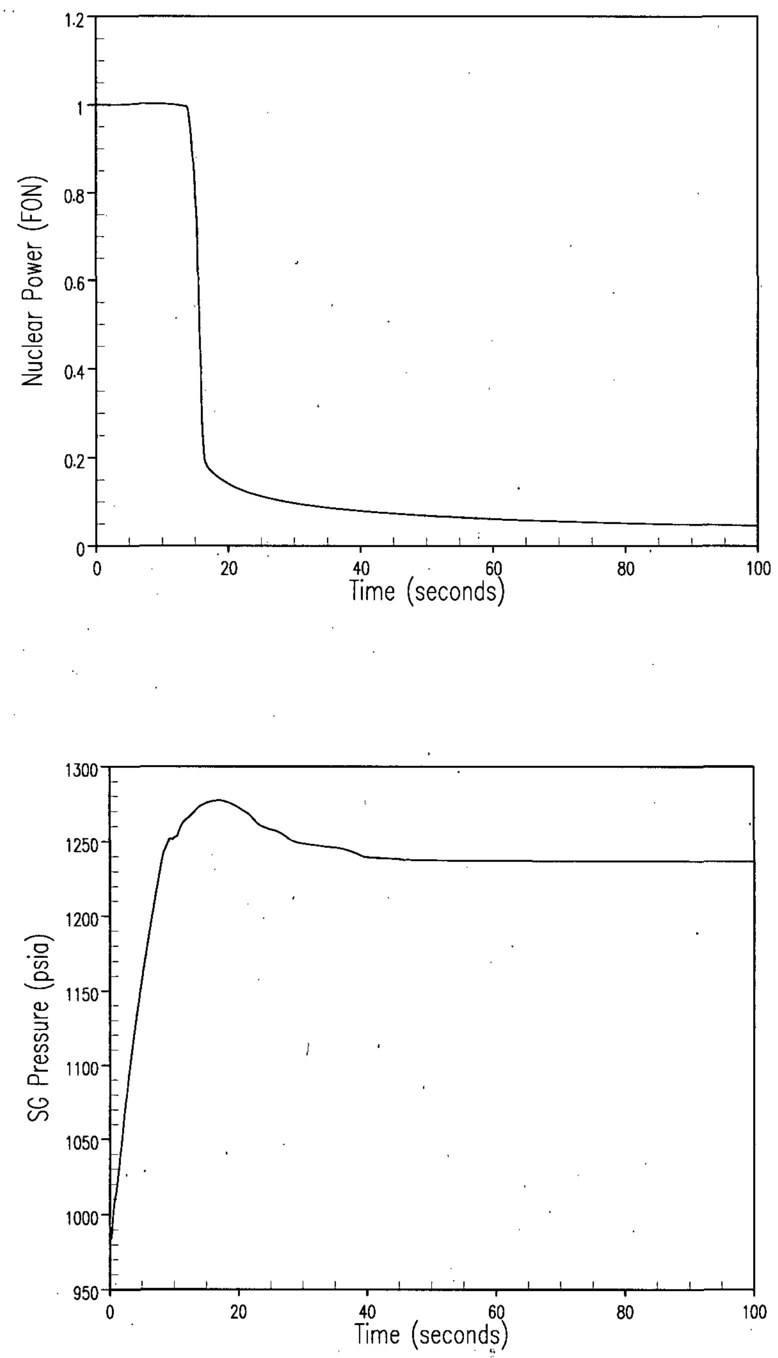


Figure 2.8.4.2-1 Unit 1 Loss of Load/Turbine Trip RCS Overpressurization Analysis Nuclear Power and Steam Generator Pressure vs. Time

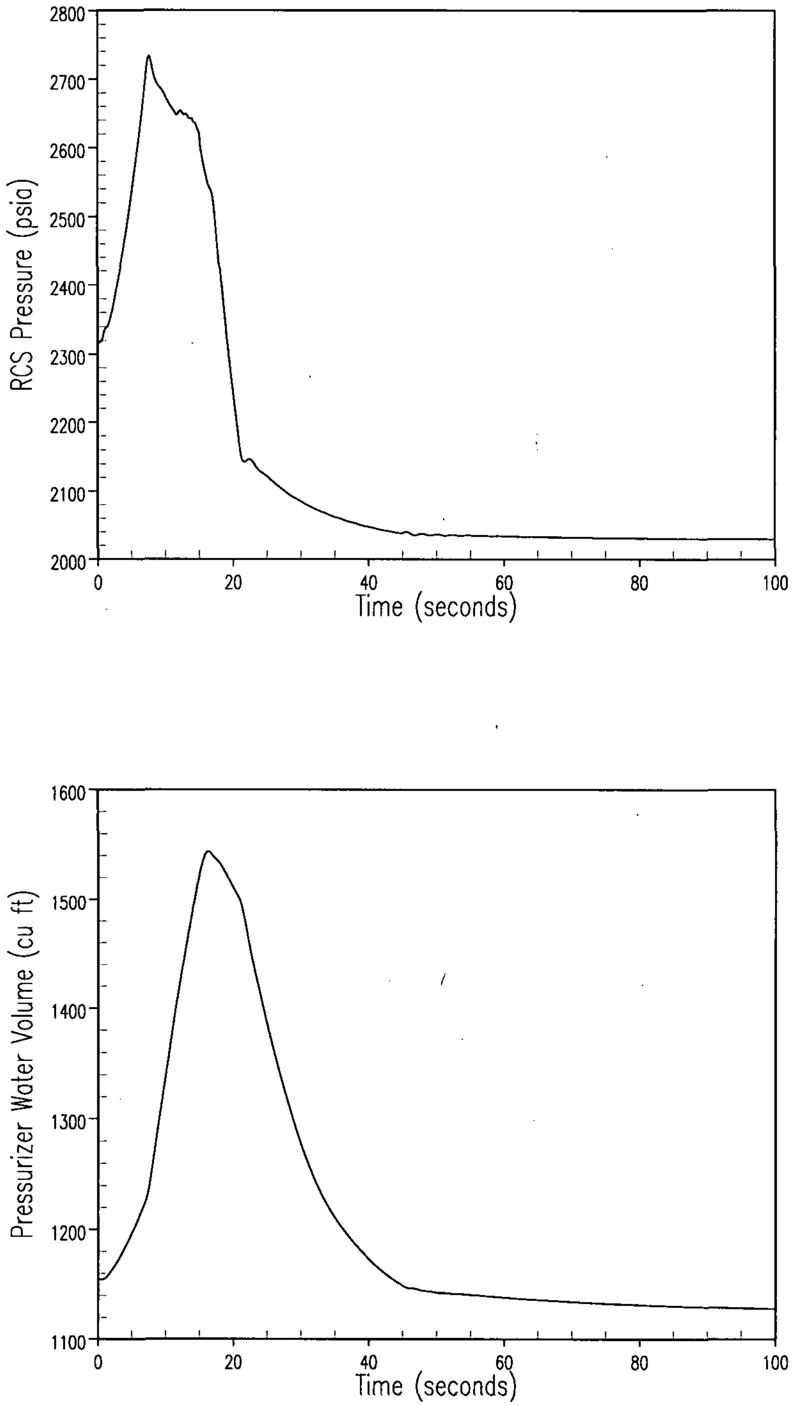


Figure 2.8.4.2-2 Unit 1 Loss of Load/Turbine Trip RCS Overpressurization Analysis RCS Pressure and Pressurizer Water Volume vs. Time

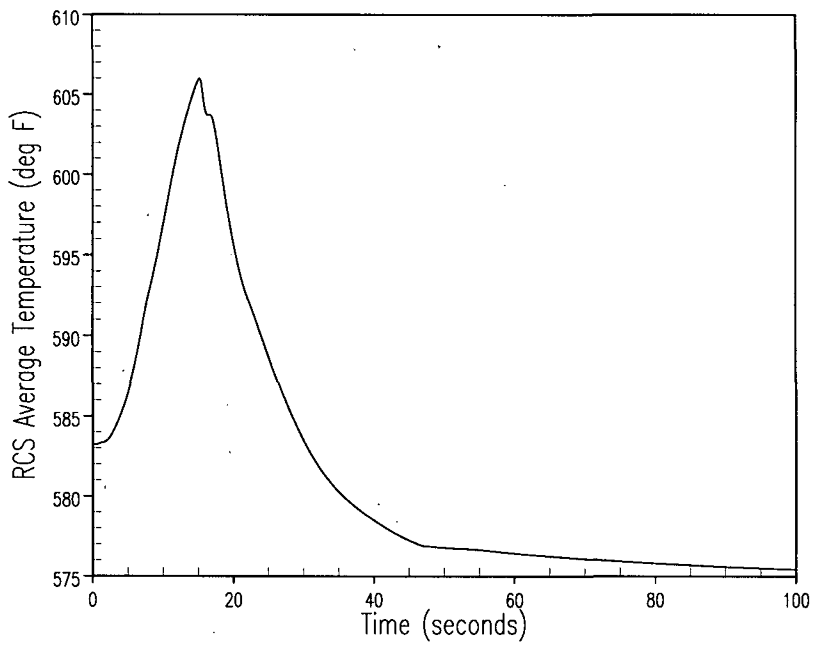


Figure 2.8.4.2-3 Unit 1 Loss of Load/Turbine Trip RCS Overpressurization Analysis RCS Average Temperature vs. Time

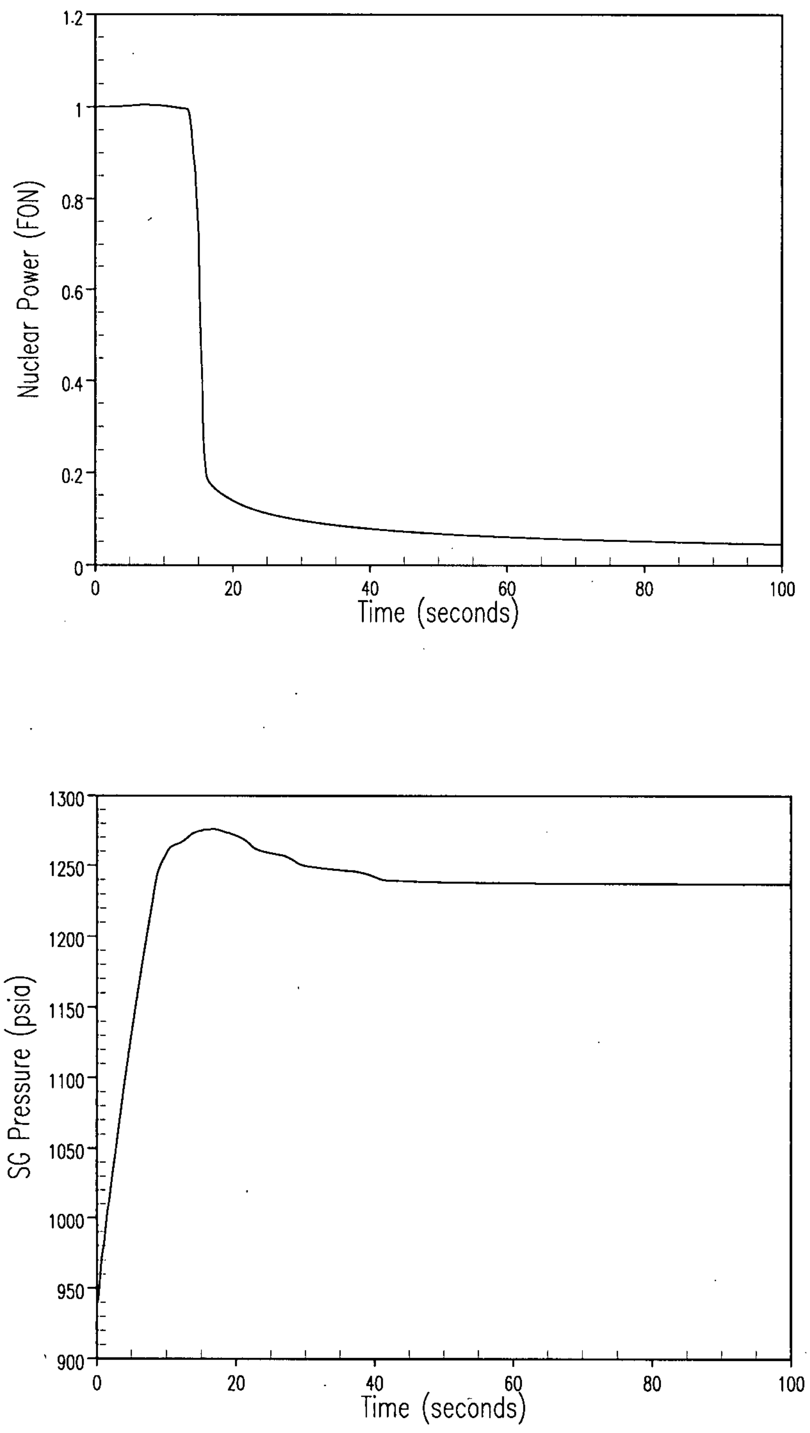
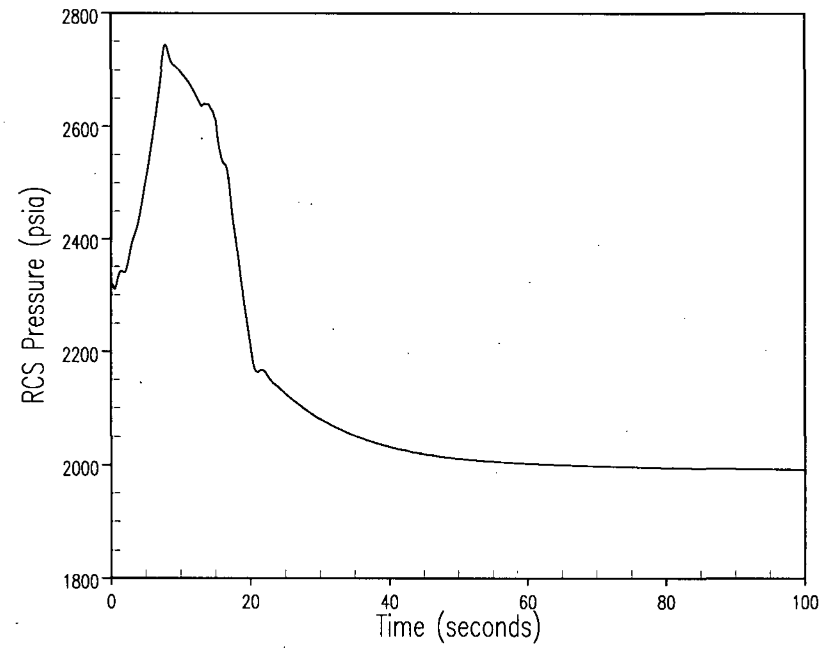


Figure 2.8.4.2-4 Unit 2 Loss of Load/Turbine Trip RCS Overpressurization Analysis Nuclear Power and Steam Generator Pressure vs. Time



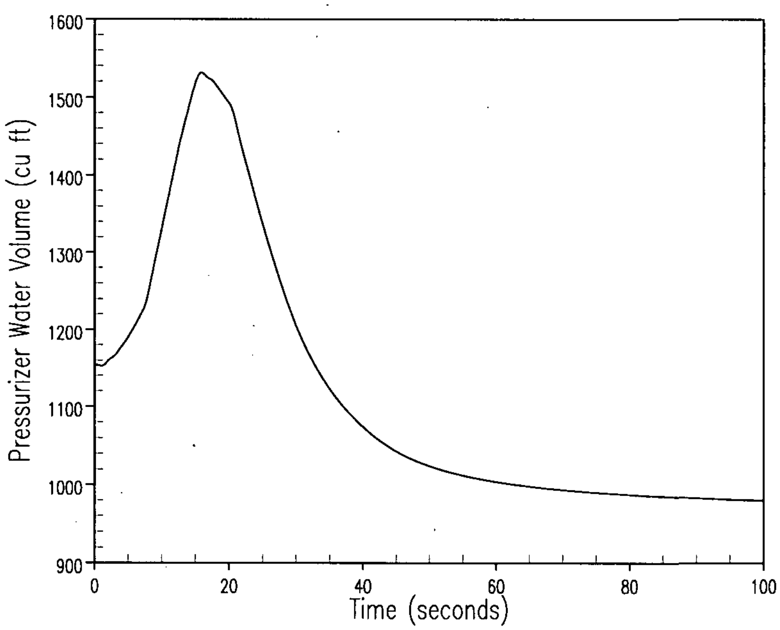


Figure 2.8.4.2-5 Unit 2 Loss of Load/Turbine Trip RCS Overpressurization Analysis RCS Pressure and Pressurizer Water Volume vs. Time

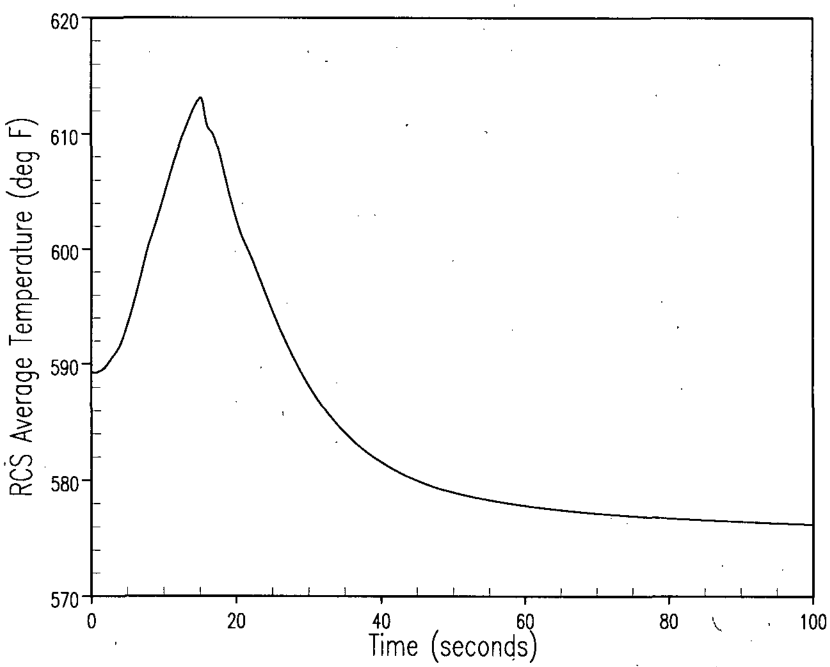


Figure 2.8.4.2-6 Unit 2 Loss of Load/Turbine Trip RCS Overpressurization Analysis RCS Average Temperature vs. Time