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Ref. # 10CFR50.90

May 14, 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, MD6616, MD8417 AND MD8418)

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 dated November 14, 2007 from Balwant Singal of the NRC to Mike Blevins of Luminant Power regarding License Amendment Request 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.

The NRC provided Luminant Power with additional RAIs from the following branches regarding the proposed changes to rated thermal power:

Electrical Engineering Branch
Mechanical Engineering Branch
Materials Branch
Reactor Systems Branch

Luminant Power has provided the requested information in the Attachment to this letter.

In Reference 2, the NRC informed Luminant Power, that they intended to complete their review of the subject License Amendment Request application in two parts (the proposed SPU and the new SFP criticality analysis) in order to meet the proposed SPU implementation schedule. Luminant Power agrees with the two part review of the License Amendment Request and proposes that the NRC complete the second part of the License Amendment Request for the SFP criticality analysis by the end of 2008 in order to support Spent Fuel Pool storage requirements.



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 contains no new commitment regarding Comanche Peak 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 May 14, 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

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
1100 West 49th Street
Austin, Texas 78756-3189

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
COMANCHE PEAK STEAM ELECTRIC STATION STRETCH POWER UPRATE

**Electrical Engineering Branch**

**Supplemental Information Provided by CPNPP:**
Safety Related Class 1E equipment and other equipment important to safety identified in the CPNPP electrical equipment qualification program were screened to ensure that they would continue to meet the requirements of 10CFR50.49, operate satisfactorily, and continue to perform their intended safety functions at the power uprated conditions.

The existing environmental parameters were compared to the post SPU conditions using the EQML in a spreadsheet format to screen for deviations in temperature, pressure and radiation. The review included normal and accident conditions both inside and outside containment. The review also addressed humidity and caustic spray.

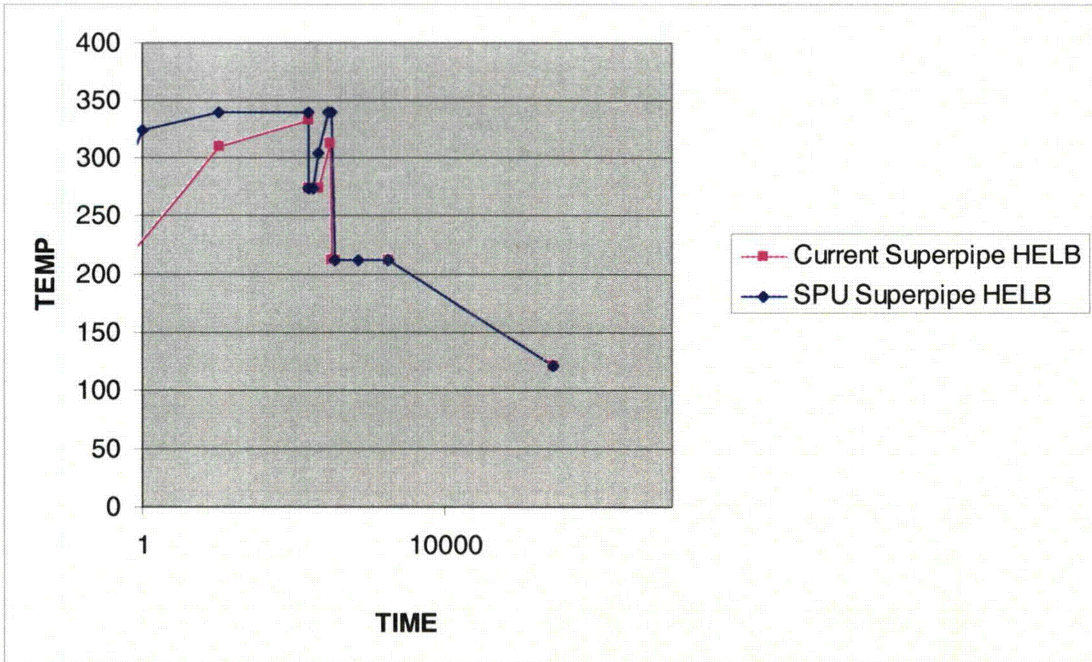
Normal pressure, temperature and humidity are unaffected by the SPU changes. There are also no changes to pH levels of the caustic spray. SPU does have an affect on the normal radiation doses.

Harsh accident parameters are affected by SPU changes. Specifically, SPU does have an affect on the pressure and temperature LOCA profiles, outside containment HELB profile and accident radiation doses.

The review found that the inside and outside containment equipment remains qualified for pressure, temperature and radiation conditions following a LOCA, MSLB and HELB. Within the mainsteam penetration areas additional evaluations required to address reduced margins for the mainsteam bypass isolation valves, ASCO solenoid valves and Namco Quick connectors are addressed as part of the design modification process at CPNPP.

**HELB COMPARISON**

The following is a comparison of the current HELB profile and the SPU HELB profile for superpipe areas.



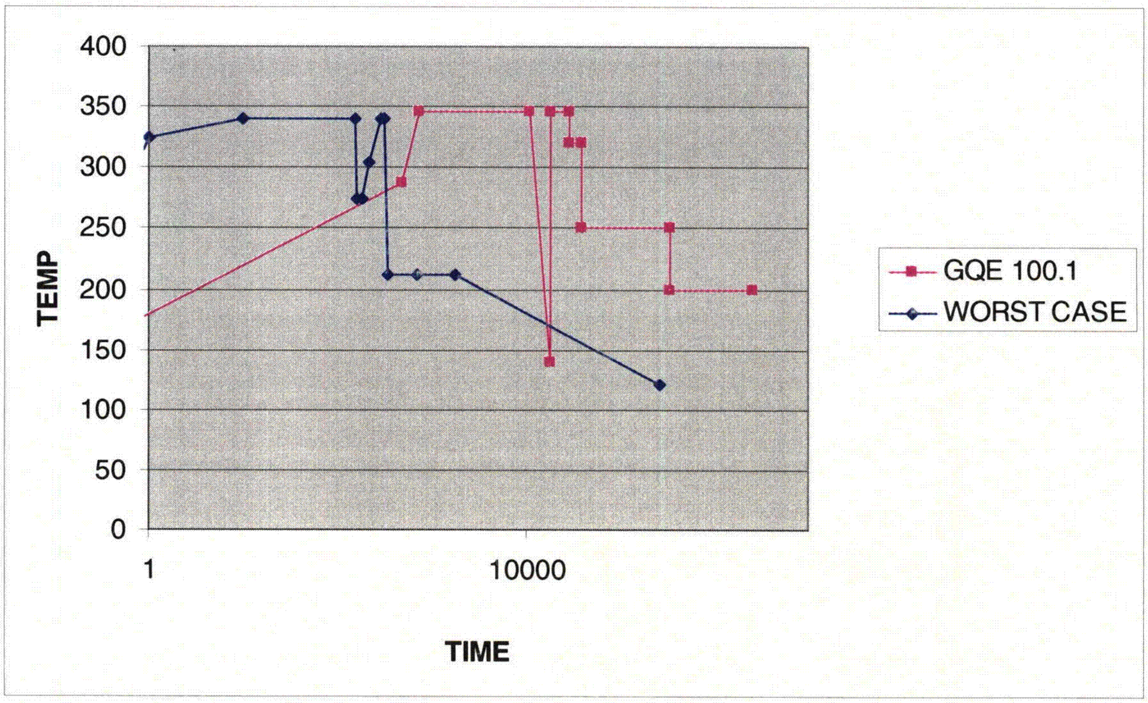

The new SPU superpipe HELB has a faster rise time than the previous temperature profile. The peak temperature of the SPU profile is 340°F. The profile is at the peak temperature for a total of 175 seconds, when the time at this temperature for both peaks is considered. The peak temperature in the previous profile was approximately 334°F, and this peak temperature was only achieved during the first temperature spike.

After 350 seconds the profiles are the same. The profiles are shown for 72.5 hours.

**Evaluation for ASCO Solenoid Valves**

ASCO solenoid valves are qualified using two primary test reports at CPNPP, AQS21678/TR and AQR-67368. The package used to qualify valves with EPDM elastomers is based on AQS21678, "Qualification Tests of Solenoid Valves by Environmental Exposure to Elevated Temperature, Radiation, Wear Aging, Seismic Simulation, Vibration Endurance, Accident Radiation and Loss-of-Coolant (LOCA) Simulation." The package used to qualify valves with Viton Elastomers is based on AQR-67368, "Report on Qualification of Automatice Switch Co. (ASCO) Catalog NP-1 Solenoid Valves for Safety-Related Applications in Nuclear Power Generating Stations".

The following is a comparison of the SPU HELB profile to the profile used to qualify ASCO valves using EPDM elastomers in AQS21678/TR:




While the ramp time of test report number AQS21678/TR is longer than that of the required profile, the duration of the peak temperature was much longer than required. Also the test from test report number AQR67368 contained both Viton and EPDM elastomers and showed that there is no failure due to high temperature or a steep ramp. Therefore, the long ramp of test report number AQS21678/TR will have no adverse impact on the qualification of the valves.

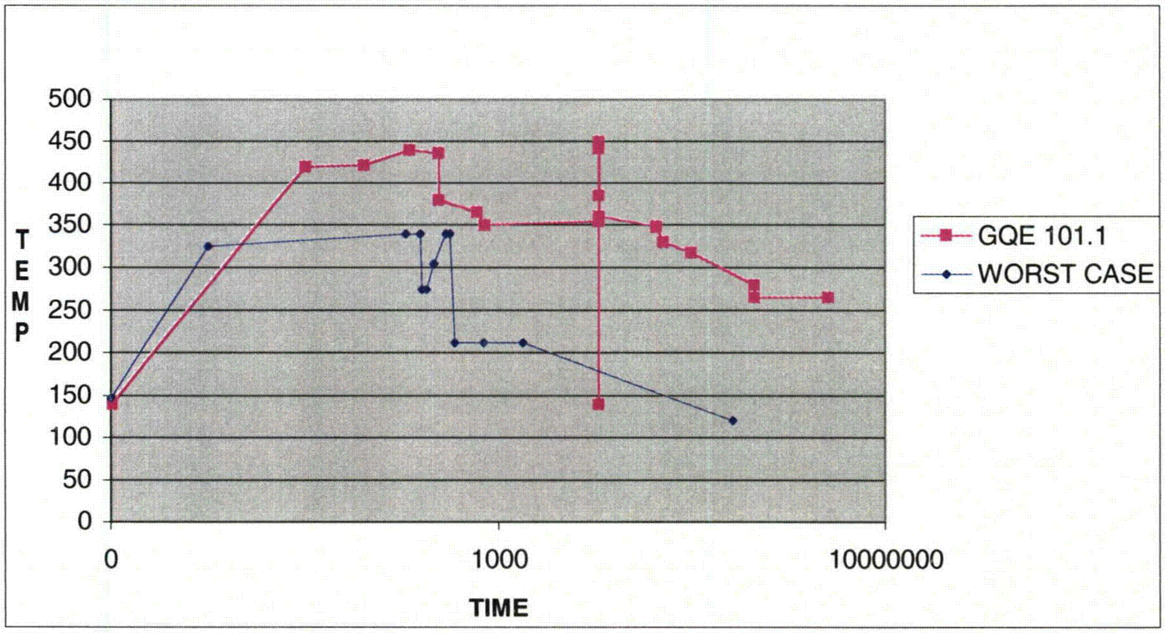
The test report number AQS21678/TR did not attain the recommended 15°F margin on temperature. This condition is acceptable based on the following:

1) The test attained a temperature of 346°F for three hours, while the requirement is 340°F for 175 seconds.

2) The test included two transients to 346°F each of which maintained 346°F for 10,000 seconds.

3) Test report number AQR67368 demonstrated a maximum temperature of 450°F. This test included EPDM and Viton elastomers.

The following is a comparison of the SPU HELB profile used to qualify ASCO valves using Viton elastomers in AQR-67368:

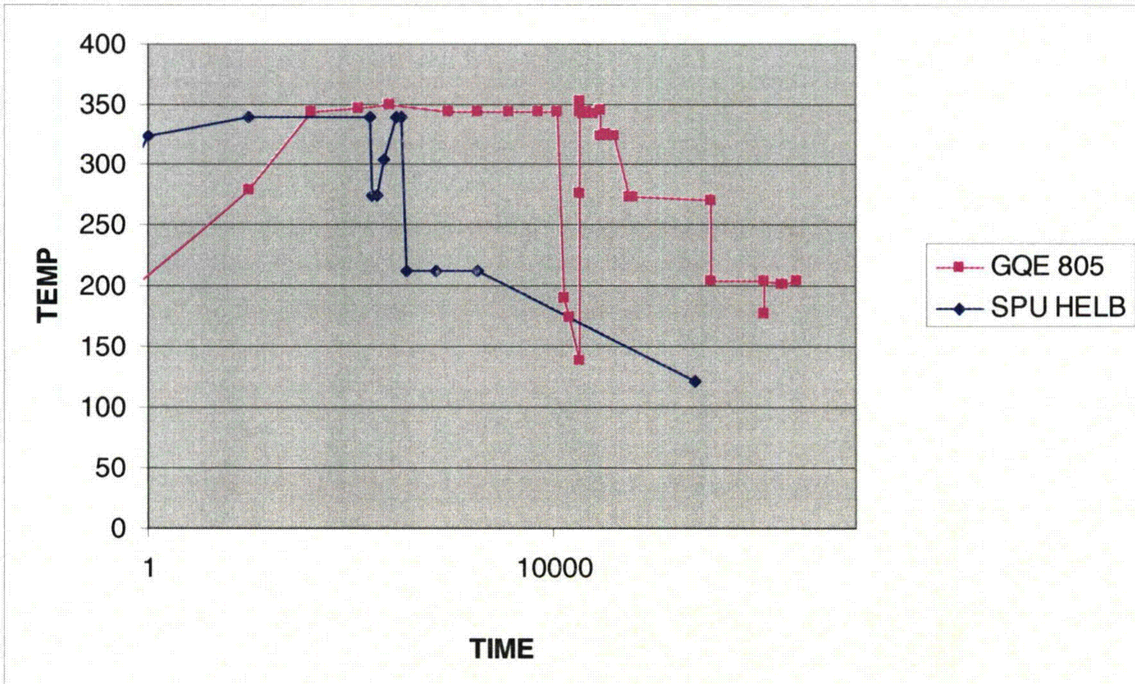



The ramp time of the required profile is steeper than the test profile. However, the peak temperature exceeds the required temperature by more than 110°F. The required profile is bounded by the test profile. The required 15°F margin is satisfied.

**Evaluation for NAMCO Quick Connectors EC210 Series**

The NAMCO Quick Connectors are qualified to QTR-142 "Generic Qualification of EC210-Series (1.0 inch) Receptacle and Connector/Cable Assemblies for Use in Nuclear Power Plants in Compliance with IEEE Standards 323-1974, 382-1972, and 344-1975".

The following is a comparison of the tested and required SPU profiles:




The ramp time of test report number QTR-142 is longer than that of the required profile, and the test profile did not attain the recommended 15°F margin on temperature. This condition is acceptable based on the following:

1) The tested profile was a two transient test. The first transient reached a peak temperature of 350°F, while the second transient reached a peak temperature of 353°F.

2) Each of the two transients was maintained for 10,760 seconds.

3) During the first 3 hour transient, the temperature did not drop below 344°F, and during the second transient the temperature did not drop below 343°F.

Margin is demonstrated by the time at temperature of the test specimens. The required peak temperature is 340°F for 175 seconds. The test specimens were exposed to over 21,500 seconds at temperatures that exceeded 343°F.

**Evaluation for MSIV Bypass Valves**

The MSIV Bypass Valves are qualified in MQESP-MS-076-01. The only non-metallic part within the valve is the packing material, which is Grafoil, grade GTN. The qualification package states that the Grafoil has a maximum service temperature of 2500°F. Additionally, the Bypass Valve does not have non-metallic parts which maintain the integrity of pressure boundary when the valve is closed. The Bypass Valves are locked closed except for plant start up.

There are no qualification issues with the MSIV Bypass Valves. The increase in accident temperature, due to SPU, will not degrade the Grafoil packing material to the point it is unable to perform its intended safety function.

Additional analyses performed to qualify specific component types for radiation are tabulated below. This included location specific radiation calculations.

Radiation Components that required Location Specific analysis

| Component Type | Action Taken |
|---|---|
| Target Rock isolation valves for venting of the pressurizer | Location Specific calculation reduced radiation dose |
| Reactor coolant wide range pressure – Rosemount 1154 (Post Accident Monitoring) | Location Specific calculation reduced radiation dose |

CPNPP processes and procedures ensure that Environmental Qualification Report documentation (EEQSPs) are updated to reflect the changes to the environmental parameters due to SPU. Also, the design modification process at CPNPP ensures that the qualification packages are updated to reflect the environmental conditions that are the result of SPU.

**Description of the location specific radiation analysis performed for the SPU**

Location-specific environmental radiation levels were developed for five (5) components since the SPU environmental radiation levels in the Environmental Zones in which the above components are located exceed the qualification levels of the referenced components.

The 5 components are located inside containment and in the following locations:

- Unit 1 pressure transmitter 1-PT-3616 – Unit 1 Room 154A
- Unit 1 pressurizer vent valves, 1-HV-3609 and 1-HV-3610 – Unit 1Room 161E
- Unit 2 pressurizer vent valves, 2-HV-3609 and 2-HV-3610 – Unit 2 Room 161E

The bounding SPU environmental radiation level in each of the listed locations slightly exceed the qualification level of the components. Consequently, radiation levels at the specific location of each of the components was determined to support component qualification.

Unit 1 pressure transmitter 1-PT-3616

It has been determined that Unit 2 pressure transmitter 2-PT-3616 is located in Unit 2 Room 154A. An existing location-specific calculation done for the Unit 2 pressure transmitter (which had been scaled up to reflect the impact of the SPU) is utilized to evaluate the radiation environment expected at the specific location of Unit 1 pressure transmitter 1-PT-3616.

Units 1 and 2 are essentially identical units; therefore, the radiation sources in the areas around the transmitters are also similar. From comparison, it is seen that 1-PT-3616 and 2-PT-3616 are similarly located in their respective units. The two transmitters are also surrounded by similar shield slabs. Consequently, the calculated radiation level at the specific location of 2-PT-3616 is also applicable to 1-PT-3616. It is determined that this reduced location specific environmental radiation level is within the qualification level of Unit 1 pressure transmitter 1-PT-3616.

Unit 1 pressurizer vent valves, 1-HV-3609 and 1-HV-3610 & Unit 2 pressurizer vent valves, 2-HV-3609 and 2-HV-3610

The Unit 1 pressurizer vent valves, 1-HV-3609 and 1-HV-3610, are located above El. 905'-6" in the Unit 1 pressurizer room, i.e., Unit 1 Room 161E. The Unit 2 pressurizer vent valves, 2-HV-3609 and 2-HV-3610, are similarly located in the Unit 2 pressurizer room, i.e., Unit 2 Room 161E.

Currently, at each Unit, the radiation levels conservatively assigned to the pressurizer room, is the same as that calculated for the steam generator compartments.

The pressurizer vent valves are above the top of the pressurizer in room 161E for both Unit 1 and Unit 2. The only significant contained source in the area is the pressurizer. The only contained radioactive sources affecting the pressurizer vent valves are pressurizer piping and the pressurizer. The vent valves are connected to 1-inch pipes and are a short distance away from 6-inch pipes that are connected to the pressurizer. The major sources of radiation affecting the valves are the 1-inch piping, 6-inch piping, and the pressurizer. In addition, in the containment building, for a depressurized loss-of-coolant accident (LOCA) scenario, the airborne radiation also contributes to the environmental radiation levels.

Computer program SW-QADCGGP was used to calculate the dose rate from the various contained sources based on source models for the pipes and the pressurizer. The contribution to the doses at the specific locations of the valves from normal operations and accident sources were calculated. For accident conditions, both the depressurized LOCA and pressurized LOCA scenarios were evaluated. The larger dose values were taken as the applicable environmental radiation levels at the specific location of these components. The depressurized-LOCA scenario resulted in slightly higher radiation levels.

The total integrated environmental radiation dose at the component location was ultimately determined based on the following contributions:

- 40-years of normal operations from contained sources in the pressurizer cubicle
- 40-years of normal operations airborne source
- 1-year depressurized LOCA contained sources in the pressurizer cubicle
- 1-year depressurized-LOCA airborne source.

The total integrated dose at the specific-locations of the pressurizer vent valves was determined to be within the qualification level of the valves.

**Mechanical Engineering Branch**

**<u>Supplemental Information Regarding Feedwater Pump provided by CPNPP:</u>**

**Summary of Analysis of Feed Pump for Nozzle Loads due to SPU**
A finite element model based on the drawings listed below was prepared in ANSYS 10.0-SP1(<1>) for the purpose of a linear static finite element analysis (FEA) investigating nozzle loading due to Stretch Power Uprate of CPNPP. The calculated stresses and deflections are compared to material properties and are shown in the table below. The nozzle loads for SPU are also tabulated below. The evaluation of the results of this analysis shows adequate margin exists in the components to perform their intended design function

**Drawings**
The finite element model is based on the following drawings:
H-3985 Rev. L Foundation drawing (GA)
H-4503 Rev. M Volute case drawing
H-4544 Rev. D Baseplate drawing

**Comanche Peak Unit 1 and Unit 2 / Feedwater Pump Nozzle and Anchorage**

| Component | Load Condition | Stress Category | Location | Stress (psi) /Deflection (in) | Allowable (psi) / (in) | Interaction Ratio |
|---|---|---|---|---|---|---|
| Main Feedwater Pump Nozzle | Thermal Expansion, Deadweight, Fluid Transient | Normal, Upset | Shaft-end Deflections | .018 in | .038 in | .474 |
| Main Feedwater Pump Nozzle | Thermal Expansion, Deadweight, Fluid Transient | Normal, Upset | Concrete Anchor Bolt Stresses | 22,757 psi | 30,856 psi | 0.73 |
| | Thermal Expansion, Deadweight, Fluid Transient | Normal, Upset | Pump Hold-down bolt stresses | 24,159 psi | 55,860 psi | 0.432 |
| Main Feedwater Pump Nozzle | Thermal Expansion, Deadweight, Fluid Transient | Normal, Upset | Baseplate Stresses | 22,967 psi | 27,000 psi | 0.85 |
| Main Feedwater Pump Nozzle | Thermal Expansion, Deadweight, Fluid Transient, Pressure | Normal, Upset | Casing Stresses | 36,134 psi | 47,639 psi | 0.75 |

Allowable Stresses were computed utilizing DBD-CS-015, DBD-CS-85, DBD-CS-018 and AISC Manual 8th Ed. for guidance.

**Table 01 Nozzle Loads for SPU Conditions**

| Location | Condition | Fz | Fy | Fx | Mz | My | Mx |
|---|---|---|---|---|---|---|---|
| | | | [lbf] | | | [lb-ft] | |
| Suction | Dead Weight ("DWT") | ± 5000 | ± 1000 | ± 1,000 | ± 4,000 | ± 8,000 | ± 8,000 |
| | Thermal ("T") | ± 3,000 | ± 6,000 | ± 6,000 | ± 10,000 | ± 23,000 | ± 23,000 |
| | Occ Fluid Transient ("F") | 0 | 0 | 0 | 0 | 0 | 0 |
| | **Load-Case-A** | -8,000 | -7,000 | -7,000 | -14,000 | -31,000 | -31,000 |
| | **Load-Case-B** | 8,000 | 7,000 | 7,000 | 14,000 | 31,000 | 31,000 |
| | | | | | | | |
| Discharge | Dead Weight ("DWT") | ± 5,000 | ± 1,000 | ± 1,000 | ± 3,000 | ± 8000 | ± 8,000 |
| | Thermal ("T") | ± 3,000 | ± 6,000 | ± 6,000 | ± 14,000 | ± 23,000 | ± 23,000 |
| | Occ Fluid Transient ("F") | ± 12,000 | ± 11,000 | ± 11,000 | ± 9,000 | ± 35,000 | ± 35,000 |
| | **Load-Case-A:** | -20,000 | -18,000 | -18,000 | -26,000 | -66,000 | -66,000 |
| | **Load-Case-B:** | 20,000 | 18,000 | 18,000 | 26,000 | 66,000 | 66,000 |

Note 1: Coordinate system:

x↔ parallel to pump shaft axis, pointing away from driver
y ↔ horizontal;
z ↔ vertical, upward

These nozzle loads are acceptable based on the casing stresses of 36,134 psi as compared to an allowable stress of 47,639 psi for a casing material of A296 CA 6NM.

**Materials Branch**

**NRC REQUEST:**
The question we had pertains to the reactor internal (RI) components that are fabricated from cast austenitic stainless steel (CASS), which may be potentially susceptible to thermal aging. We need to know several things about the CASS material used in these RI components in order to determine whether the licensee has adequately addressed the susceptibility of these components to thermal aging:

1. What are the delta-ferrite levels in these materials?
2. What was the casting process that was used for manufacturing these components? Specifically, were the components manufactured by centrifugal casting or static casting?

This question applies only to Unit 1 because the licensee stated that the Unit 2 RI components do not contain any CASS materials.

**Supplemental Information provided by CPNPP:**

As reported in the licensing report supporting the power uprating of Comanche Peak Unit 1 the reactor vessel internals of Unit 1 contains cast austenitic stainless steel (CASS). The Nuclear Regulatory Commission, in reviewing the licensing report, is interested in specifics of the CASS material such as molybdenum content, ferrite content, and method of casting (static versus centrifugal). After an exhaustive search of microfilmed records a set of certifications was located for the 52 upper support column base castings. This document presents the pertinent information found in the certifications.

All 52 of the certifications were made on identical forms with hand-written entries. Although the certifications are in relatively poor condition, it is possible to piece together individual certifications to result in the ability to read all areas. Figure 1 shows a representative certification which happens to be the best available. The ASME SA-351, Grade CF 8 castings (Reference 1) were made by Kearsarge Metallurgical Company of Conway, New Hampshire, in early 1975. Each casting was identified by a unique casting serial number, heat number, and heat treat number; much of this information is unreadable on the forms. As was required for certification to SA-351 in 1975 the following elements were reported: carbon, manganese, silicon, chromium, nickel, phosphorus, and sulfur; cobalt was also reported, although it was not required. Tensile requirements consisted of yield strength, ultimate tensile strength, and % elongation. Each certification was signed and dated. Each certification was further stamped with a Westinghouse heat code number and a Westinghouse inspection mark. Missing from the certifications, because it was not required by SA-351, is the information requested by the NRC.

It is very likely that the static casting method was used due to the shape of the castings; centrifugal casting is more associated with piping.

Ferrite can be calculated using ASTM A-800 (Reference 2) and the following equations of chromium and nickel equivalents:

$$\mathrm{Cr(equiv)} = \mathrm{Cr} + 1.5 \times \mathrm{Si} + 1.4 \times \mathrm{Mo} + \mathrm{Nb} - 4.99$$
$$\mathrm{Ni(equiv)} = \mathrm{Ni} + 30 \times \mathrm{C} + 0.5 \times \mathrm{Mn} + 26 \times (\mathrm{N}-0.02) + 2.77$$

Ferrite is determined by plotting the ratio of chromium and nickel equivalents on the Schoefer diagram presented in A-800. There exists, however, a problem with using A-800 (or any other method which relies on an equation based on elements): molybdenum, niobium, and nitrogen are required for the equation and all three elements are missing from the certifications. In order to estimate the relative amounts of molybdenum, niobium, and nitrogen, we note that WCAP-10456 (Reference 3) provides a precedence to estimate quantities of these elements. This technique was previously used for determining the ferrite in cast reactor coolant piping as reported in WCAP-10456. In WCAP-10456 molybdenum was estimated to be 0.20, niobium was estimated as 0, and nitrogen was estimated at 0.05. Use of these estimates is appropriate for this evaluation because these values are typical amounts of these elements that can be expected in similar castings. Although WCAP-10456 was making ferrite calculations for CF-8A material the estimates would also be applicable to CF-8 materials. Using a representative chemistry of 19.2% chromium, 0.7% silicon, 10% nickel, 0.06% carbon, and 1.15% manganese the chromium equivalent is calculated to be 15.54 and the nickel equivalent is calculated to be 15.93. With a resulting ratio of chromium to nickel equivalents of 0.98 the ferrite is shown to be approximately 3% in the Schoefer diagram. Respecting the potential for error as shown in the Schoefer diagram the ferrite percentage could range from about 0 to 6.

Archive certifications for cast austenitic stainless steels have been reviewed for Comanche Peak Unit 1. Using estimated amounts for missing elements, a representative ferrite amount can be calculated in accordance with ASTM A-800 to be 3% with a +/- 3% range of error.

**REFERENCES:**

1. ASME Section II, Part A, 1975 Edition, SA-351, Specification for Castings, Austenitic, Austenitic-Ferritic (Duplex), for Pressure Containing Parts.

2. ASTM A-800-01, Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof.

3. WCAP-10456, November 1983, The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems, Westinghouse Electric Co.

**Reactor Systems Branch**

**NRC REQUEST:**
The Comanche Peak Steam Electric Station (CPSES) Stretch Power Uprate Licensing Report (SPULR) contained a special, overpressure evaluation for a rod withdrawal at a low-power condition (RWAP).

The licensee stated that Westinghouse performed a "conservative generic evaluation" showing adequate protection would be provided through the use of the high positive flux rate and high pressurizer pressure reactor trip functions to prevent overpressurization for this event. The licensee stated that this evaluation is applicable to CPSES. The details of this evaluation are not readily available on the CPSES Dockets; therefore, the Nuclear Regulatory Commission (NRC) staff obtained pertinent information from a request for additional information (RAI) response provided by the Diablo Canyon Power Plant (DCPP) licensee containing details of the generic overpressure RWAP evaluation performed by Westinghouse.

The parameters used in the evaluation were not bounding at DCPP, and the licensee justified the continued conservatism.

By comparison, CPSES has a higher uprate power level than the generic analysis.

You are requested to provide the following additional information.

1. A copy of the generic evaluation performed for CPSES.

2. Transient plots for the limiting pressurization case resulting in excessive reactor coolant system (RCS) pressurization.

3. Compare CPSES-specific parameters to demonstrate the applicability of the generic evaluation to CPSES.

4. Explain how the CPSES Licensing Basis accounts for crediting the Positive Flux Rate Trip (PFRT) in transient analyses.

REFERENCE:

Oatley, David H., Pacific Gas and Electric Company, Letter to USNRC, "Response to NRC Request for Additional Information Regarding License Amendment Request 03-02, 'Response Time Testing Elimination and Revision to Technical Specification 3.3.1, 'Reactor Trip System Instrumentation,''" December 2, 2003.

**Supplemental Information provided by CPNPP:**

The plant response during an uncontrolled rod control cluster assembly (RCCA) bank withdrawal at power (RWAP) event can vary significantly depending on the reactivity insertion rate, initial power level, and core reactivity feedback effects. The RWAP event results in an addition of positive reactivity into the core and an increase in the core heat flux. Since the heat extraction from the steam generator (SG) lags behind the core power generation, 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 a departure from nucleate boiling (DNB) or reactor coolant system (RCS) overpressure condition. With respect to the DNB concern, the RWAP analysis described in detail in Section 2.8.5.4.2 of the Comanche Peak Stretch Power Uprate (SPU) Licensing Report demonstrates the adequacy of the plant protection features by confirming that the departure from nucleate boiling ratio (DNBR) never falls below the safety analysis limit. With respect to the RCS pressure concern, a Westinghouse generic RWAP analysis was determined to be applicable to Comanche Peak for demonstrating the adequacy of the plant protection features.

The following information describes the Westinghouse generic RWAP analysis that was used to verify that the Comanche Peak plant protection features are adequate for preventing the RCS pressure from exceeding 110 percent of the design value (2748.2 psia). The generic RWAP RCS pressure analysis was performed with the LOFTRAN computer code, which is an NRC-approved code for RWAP analyses (see Reference 1). As in other peak RCS pressure analyses, e.g. loss of load/turbine trip, conservative initial condition uncertainties and modeling features are applied in the generic RWAP analysis so as to maximize the resultant peak RCS pressure.

The magnitude of the RCS pressure increase resulting from the RWAP event is a function of the reactivity insertion rate, the initial power level, and the amount of reactivity feedback. The potential for RCS overpressure increases as the time between the start of the reactivity insertion (RCCA bank withdrawal) and a reactor trip increases due to the time lag associated with the transfer of heat generated in the core to the steam generator secondary-side. For small positive reactivity insertion rates at any time in core life, the nuclear power and RCS temperature increase relatively slowly and in equilibrium such that the thermal lag effect is minimal, and thus the peak RCS pressure is not a concern. For large reactivity insertion rates at the end of core life conditions, the nuclear power and RCS temperature increase in relative equilibrium because of the effect of the end of core life reactivity feedback characteristics, and thus the peak RCS pressure is not a concern. However, for large reactivity insertion rates at the beginning of core life with minimum reactivity feedback characteristics, the nuclear power increases much faster than the rate at which the energy can be transferred from the core to the steam generator secondary-side. It was determined that a low initial power level is conservative for maximizing the peak RCS pressure following an RCCA bank withdrawal because the resultant net power mismatch between the primary and secondary systems is maximized before a reactor trip is actuated. The power mismatch that exists when the reactor trip occurs is a significant factor in determining the magnitude of the insurge into the pressurizer and therefore the magnitude of the RCS pressure rise.

In order to obtain conservative RCS pressure results that can be applied to multiple plants, the following assumptions were made in the generic RWAP analysis:

(1) The initial NSSS power level is 8 percent of 3608 MWt. 8 percent corresponds to the minimum power level at which the high neutron flux low setting reactor trip can be blocked (10 percent) minus 2 percent uncertainty. Starting at a low power allows the core power to increase and the maximum power mismatch to occur before a reactor protection setpoint is reached.

(2) Minimum reactivity feedback, including a +7 pcm/°F moderator temperature coefficient, was assumed in the generic study to allow the core power to increase more rapidly, which results in a greater power mismatch between the primary and secondary systems.

(3) The range of positive reactivity insertion rates considered is consistent with the bounding range that was examined in the RWAP DNB analysis. A sensitivity study showed that insertion rates

less than 20 pcm/sec are non-limiting with respect to RCS pressure. The maximum reactivity insertion rate analyzed was 110 pcm/sec, which exceeds the maximum possible reactivity insertion rate associated with the simultaneous withdrawal of the two control rod banks having the maximum combined worth at the maximum speed.

(4) The initial reactor vessel average temperature ($T_{avg}$) is 586.5°F, which is very conservative (high) for an initial power level of 8 percent; a high initial $T_{avg}$ is conservative because the rate of liquid expansion becomes more severe with increased temperature.

(5) The initial pressurizer water level, which corresponds to 10% power plus uncertainty, is 35.1% of span. Maximizing the initial pressurizer water level minimizes the available pressurizer vapor volume space and maximizes the net pressurization effect for a given pressurizer liquid insurge.

(6) Accounting for an uncertainty of ±50 psi, cases were evaluated at initial pressurizer pressure values of 2200 psia and 2300 psia. A sensitivity study showed that the direction of conservatism is dependent on the reactivity insertion rate, and thus a range of initial pressurizer pressure values was considered.

(7) There was no credit taken for the pressurizer power-operated relief valves' (PORVs) relief capacity.

(8) There was no credit taken for the pressurizer spray system to control RCS pressure.

(9) There was no credit taken for the steam dump control system.

(10) The pressurizer safety valve (PSV) lift setpoints were assumed to be at a maximum value of 2600 psia, which accounts for 3 percent setpoint tolerance plus 1 percent setpoint shift. The setpoint shift is modeled along with a purge delay time of 1.5 seconds to account for water-filled PSV loop seals as discussed in WCAP-12910 (Reference 2).

(11) A maximum (bounding for all 4-loop plants) pressurizer surge line friction factor was applied to maximize the pressure drop between the RCS and pressurizer, and thereby maximize the peak RCS pressure during PSV relief conditions.

(12) Maximum (bounding for all 4-loop plants) Main Steam Safety Valve (MSSV) setpoints were applied to delay the secondary-side steam relief.

(13) The generic RWAP analysis showed that the following two reactor trip functions were sufficient in helping (along with the PSVs) provide the protection required to limit the peak RCS pressure to an acceptable level: high pressurizer pressure (HPPT = High Pressurizer Pressure Trip) and high positive neutron flux rate (PFRT = Positive Flux Rate Trip). For the HPPT, a setpoint of 2440 psia and a signal delay time of 2 seconds were applied. For the PFRT, a setpoint of 9 percent/second with a time constant of 2 seconds and a signal delay time of 3 seconds were applied.

(14) The RCCA trip insertion characteristics were based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

Figure 1 shows trends of the calculated peak RCS pressure results versus reactivity insertion rate for the two generic cases that resulted in the maximum RCS pressures without taking credit for the PFRT. Case 1 assumes the minimum initial pressurizer pressure of 2200 psia, while Case 2 assumes the maximum initial pressurizer pressure of 2300 psia. What is important to observe in Figure 1 is that although the high pressurizer pressure reactor trip can be actuated in time to protect against exceeding the RCS pressure limit (2748.2 psia) for lower reactivity insertion rates, it is insufficient for higher reactivity insertion rates. For low reactivity insertion rates, the nuclear power and RCS temperature increase relatively slowly, such that the main steam safety valves (MSSVs) are able to open and limit the primary-to-secondary power mismatch until the reactor trip is actuated. At greater reactivity insertion rates, the reactor power begins increasing faster than the primary system can transfer the heat to the secondary, and the power mismatch continues to increase until the reactor trip occurs and the control rods insert into the core. Without the PFRT, the maximum reactivity insertion rate would need to be limited to a fairly low value.

Figure 2 shows trends of the calculated peak RCS pressure results versus reactivity insertion rate for the same two limiting RCS pressure cases with credit taken for the PFRT. Table 1 lists the time sequence of events for the Case 2 (2300 psia initial pressurizer pressure) characteristic small, intermediate, and large reactivity insertion rate results, with the PFRT function credited for mitigation. For the small reactivity insertion rate, the power increases slowly enough such that the PFRT is not generated. Similar to the results shown in Figure 1, Figure 2 shows that for lower reactivity insertion rates, the HPPT is actuated in time to limit the power mismatch and to keep the RCS pressure below the limit. As the reactivity insertion rate increases to 25-35 pcm/second (depending on the Case), the resultant increase in reactor power becomes rapid enough to actuate the PFRT. As the reactivity insertion rate becomes larger, the PFRT is actuated earlier in the RWAP event. Up to a point after the PFRT comes into play (~60 pcm/second), the effect of tripping the reactor earlier outweighs the effect of an increase in the reactivity insertion rate, and the RCS pressure response becomes less limiting. Beyond this point, the peak RCS pressure becomes more limiting with each increase in reactivity insertion rate, despite an earlier reactor trip. Based on the results presented in Figure 2, the combination of HPPT and PFRT provides adequate protection for a bounding range of reactivity insertion rates to ensure that the RWAP event does not result in an RCS pressure that exceeds the limit.

Figures 3 and 4 provide comparisons of Case 2 results for the maximum reactivity insertion rate (110 pcm/second) with and without crediting the PFRT. Figure 3 compares the RCS pressure transients and Figure 4 compares the pressurizer water volume transients.

Table 2 provides comparisons between the critical parameter values applied in the generic RWAP analysis and the corresponding Comanche Peak SPU safety analysis values. As shown in the table, most of the critical parameter values applied in the generic RWAP analysis bound the Comanche Peak SPU safety analysis values. The only two exceptions are the nominal NSSS power level and the HPPT setpoint for Unit 1.

It is determined that the combination of the conservatisms in the other parameters significantly offset the impact of the non-conservative parameters. In particular, the values for the following parameters are very conservative in the generic analysis: moderator temperature coefficient, initial vessel average temperature, RCS pressure uncertainty, PSV setpoint, HPPT delay, and PFRT delay. Therefore, it is concluded that the Westinghouse generic RWAP analysis is valid for Comanche Peak Units 1 and 2 in support of the SPU.

Note that although it was not identified in Table 2.8.5.0-4 (*Summary of RTS and ESFAS Functions Actuated*) of the Licensing Report (WCAP-16840), the High Pressurizer Pressure and High Positive Neutron Flux Rate reactor trip functions were credited in the analysis of the RWAP event (in addition to the Power-Range High Neutron Flux and Overtemperature N-16 reactor trips).

**References**

1. **WCAP-7907-A**, "LOFTRAN Code Description," T. W. T. Burnett, April 1984.

2. **WCAP-12910 Revision 1-A**, "Pressurizer Safety Valve Set Pressure Shift, WOG Project MUHP 2351/2352," G.O. Barrett, June 1993.

3. **NSAL-02-11**, "Reactor Protection System Response Time Requirements," July 29, 2002.

**Table 1**

Westinghouse Generic RWAP Analysis
Sequence of Events for Case 2 Results
With Credit for the PFRT

| Event Description | Reactivity Insertion Rate | | |
|---|---|---|---|
| | **20 pcm/sec** | **60 pcm/sec** | **110 pcm/sec** |
| Initiation of Uncontrolled RCCA Withdrawal | 0 sec | 0 sec | 0 sec |
| Reactor Trip Setpoint Reached | 20.2 sec (HPPT) | 7.1 sec (PFRT) | 3.7 sec (PFRT) |
| RCCAs Begin to Fall into Core | 22.2 sec | 10.1 sec | 6.7 sec |
| Peak RCS Pressure Occurs | 24.9 sec (2592 psia) | 13.1 sec (2617 psia) | 9.5 sec (2708 psia) |

**Figure 1**

Westinghouse Generic RWAP Analysis
Peak RCS Pressure Results Without PFRT

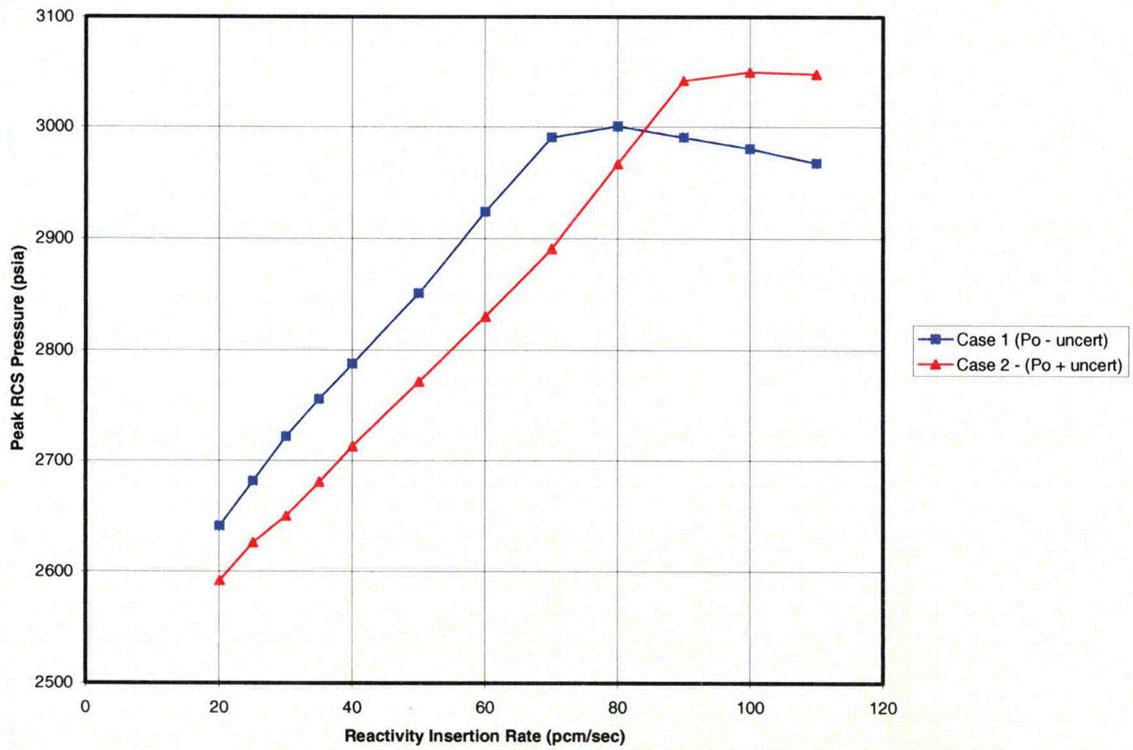


**Figure 2**

Westinghouse Generic RWAP Analysis
Peak RCS Pressure Results with PFRT

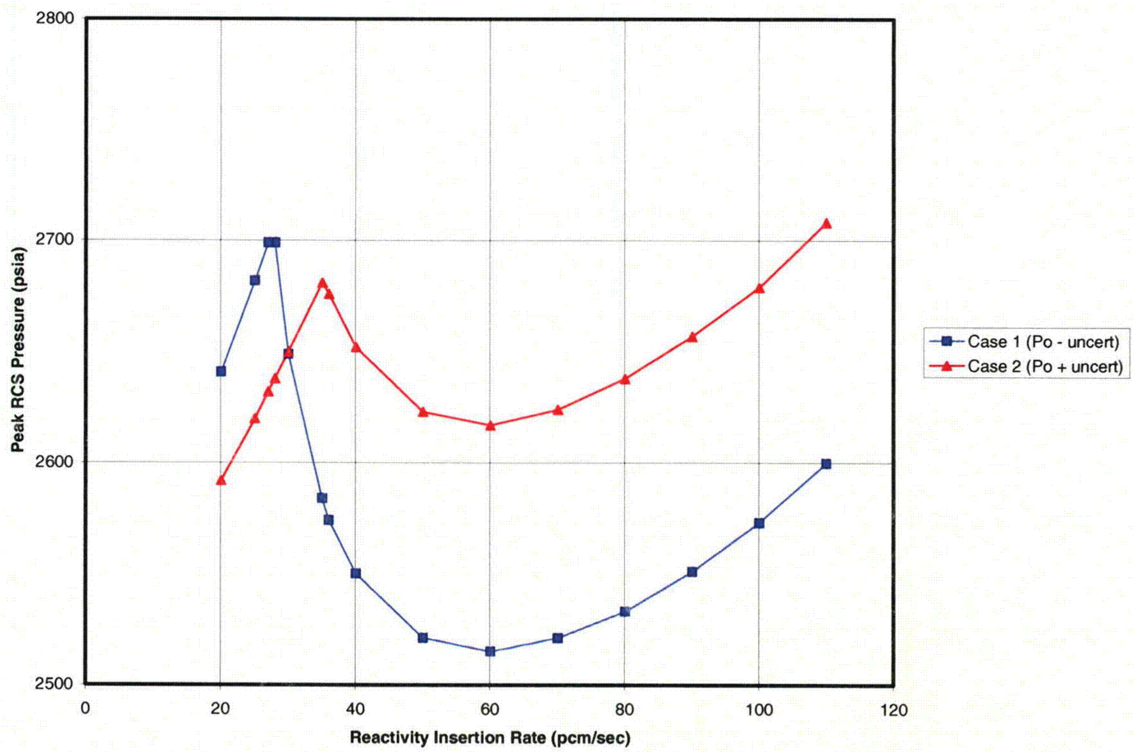


**Figure 3**

Westinghouse Generic RWAP Analysis
RCS Pressure Transient
for 110 pcm/sec Reactivity Insertion Rate

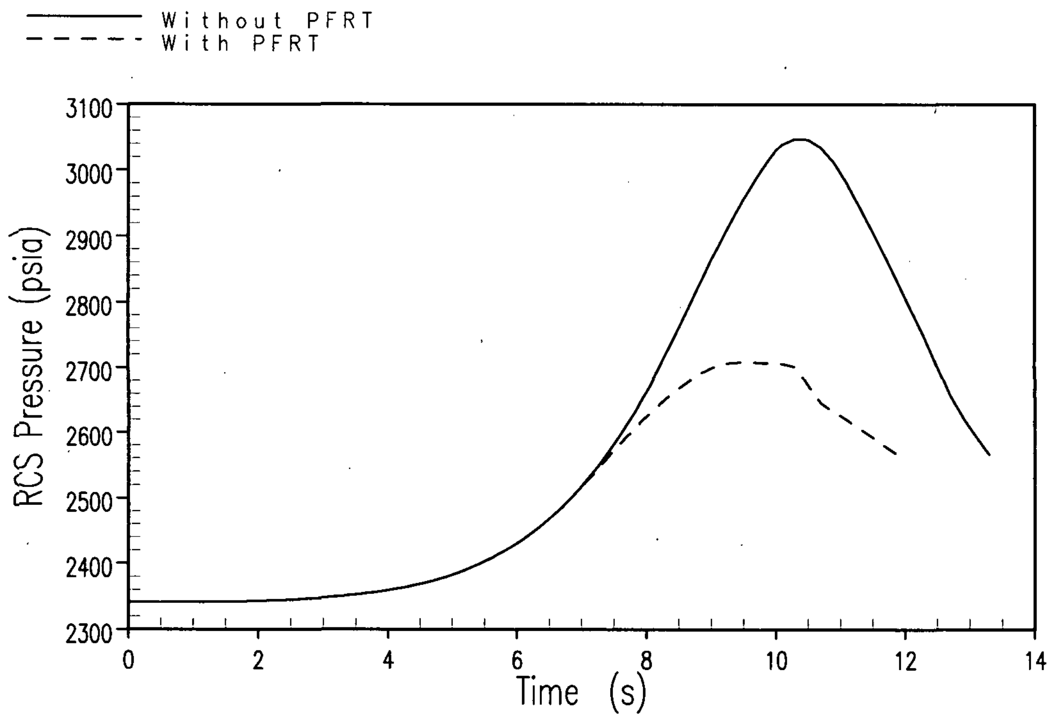
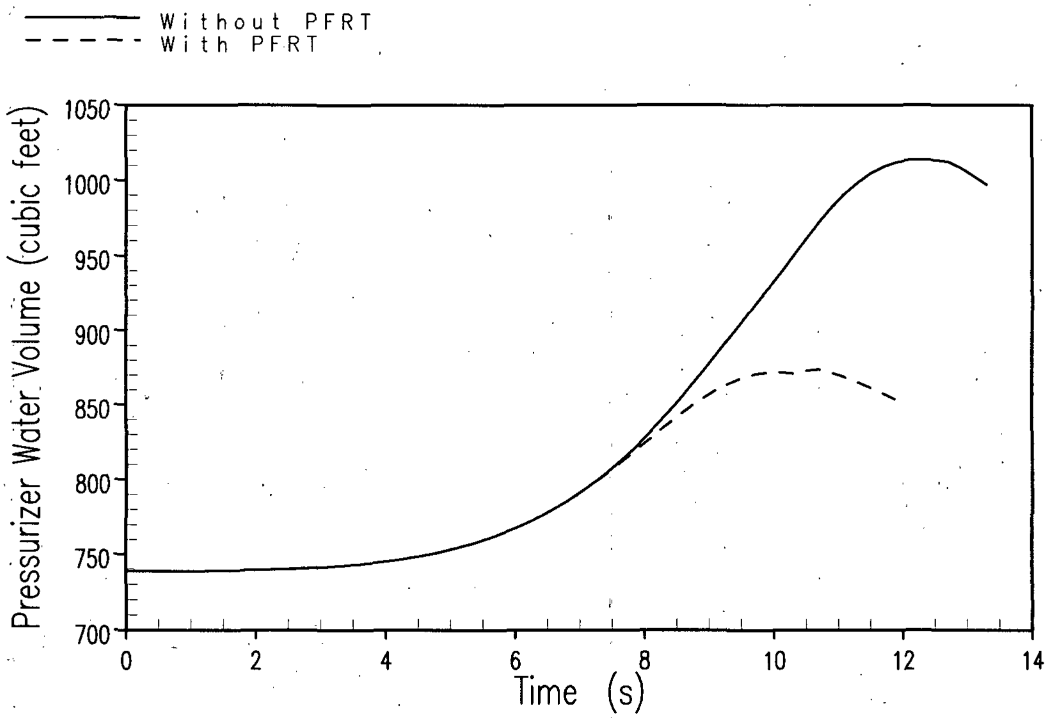


**Figure 4**

Westinghouse Generic RWAP Analysis
Pressurizer Water Volume Transient
for 110 pcm/sec Reactivity Insertion Rate



## Table 2

Comparison of Westinghouse Generic RWAP Analysis Critical Parameters to Comanche Peak SPU Parameters

| **Critical Parameter** | **Generic Analysis** | **Comanche Peak SPU** | **Is Generic Analysis Value Bounding?** |
|---|---|---|---|
| Nominal (100%) NSSS Power | 3608 MWt | 3628 MWt | No |
| Power Uncertainty | -2% | -0.6% | Yes |
| Moderator Temperature Coefficient | +7 pcm/°F | +5 pcm/°F | Yes |
| Maximum Reactivity Insertion Rate | 110 pcm/sec | 110 pcm/sec | Yes |
| Initial Vessel Average Temperature (at 10% power), Including Uncertainty | 586.5°F | 566.2°F | Yes |
| Initial Pressurizer Water Level | 35.1 % of span | 33.5 % of span | Yes |
| Nominal RCS Pressure | 2250 psia | 2250 psia | Yes |
| RCS Pressure Uncertainty | ±50 psi | ±30 psi | Yes |
| PSV Setpoint, Including Tolerance and Setpoint Shift | 2600 psia | 2521.4 psia | Yes |
| PSV Loop Seal Purge Delay | 1.50 sec | 1.05 sec | Yes |
| HPPT Setpoint | 2440 psia | 2460 psia (Unit 1)<br>2437 psia (Unit 2) | No (Unit 1)<br>Yes (Unit 2) |
| HPPT Delay | 2.0 sec | 1.25 sec | Yes |
| PFRT Setpoint / Rate Time Constant | 9.0% / 2 sec | 6.3%[1] / 2 sec | Yes |
| PFRT Delay | 3.0 sec[2] | 0.5 sec | Yes |

[1]The value of 6.3% of Rated Thermal Power (RTP) is the allowable value from the current Comanche Peak Technical Specifications. It is recognized that the safety analysis limit for this setpoint (with all uncertainties included) would be higher. However, it is not expected that the safety analysis limit for this setpoint would exceed the value of 9.0% assumed in the generic analysis.

[2]As discussed in Westinghouse Nuclear Safety Advisory Letter (NSAL) 02-11 (Reference 3), response time testing is not required to verify this value.