SAFETY EVALUATION FOR OPERATION OF

COMANCHE PEAK UNIT 1

WITH A POSITIVE MODERATOR TEMPERATURE COEFFICIENT

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SECTION 1
INTRODUCTION

Safety analyses and evaluations have been performed to support the proposed Technical Specification change for Comanche Peak Unit 1 which would allow a more positive moderator temperature coefficient (MTC) to exist during power operation. The evaluations and results of the analyses, which are presented in the following sections, demonstrate that the proposed change can be accommodated with margin to applicable FSAR safety limits.

The present Comanche Peak Technical Specifications require the MTC to be +0 pcm/°F* or less at all times while the reactor is critical. A positive coefficient at reduced power levels results in a significant increase in fuel cycle flexibility, but has only a minor affect on the safety analyses of the events presented in the FSAR.

The proposed Technical Specification change would allow a +5 pcm/°F MTC below 70 percent of rated power, ramping down to pcm/°F at 100 percent power. This MTC is diagrammed in gure 1. A power-level dependent MTC Technical Specification was chosen to minimize the affect on postulated accidents at high power levels without affecting the fuel cycle flexibility.

As the power level is raised, the average core water temperature increases as allowed by the programmed average temperature for the plant, tending to bring the MTC more negative. Also, as xenon builds into the core and as fuel burnup is achieved, the required boron concentration is reduced and the MTC will become negative over the entire operating power range.

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* 1 pcm = 10^{-5} $\Delta k/k$

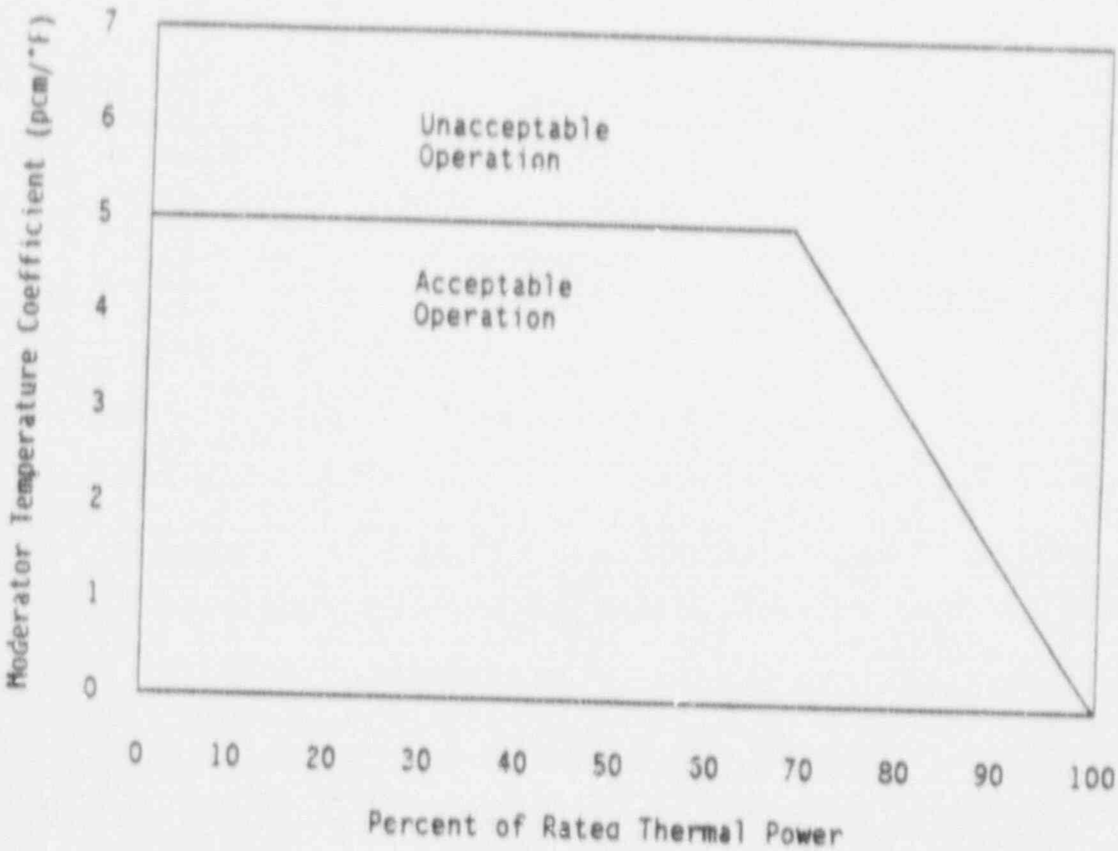

Figure 1

MODERATOR TEMPERATURE COEFFICIENT VERSUS POWER

SECTION 2
ACCIDENT EVALUATIONS

I. Introduction

The impact of a positive MTC on the LOCA and non-LOCA accident analyses presented in the Comanche Peak Unit 1 Final Safety Analysis Report (FSAR) has been assessed. Those events that were found to be sensitive to a positive MTC were reanalyzed. In general, these events are limited to those transients which cause the reactor coolant temperature to increase prior to reactor trip.

For the majority of the events analyzed and presented herein, it was assumed that a +5 pcm/°F MTC existed at full power which is conservative with respect to the proposed Technical Specification diagrammed in Figure 1. This Technical Specification requires that the coefficient be linearly ramped to zero above 70 percent power. For some events, it was necessary to credit the fact that the MTC is zero at full power in order to obtain acceptable results. For those events, it was noted that analyses for other similar Westinghouse plants have demonstrated that a full power - zero MTC combination bounds all lower power - allowable positive MTC combinations.

The remaining analysis assumptions were based on the identical analysis methods, computer codes, and assumptions employed in the FSAR. Any exceptions are noted in the discussion of each event. Accidents not analyzed included those resulting in excessive heat removal from the reactor coolant system for which the most negative MTC is more conservative. For some of the events presented in the FSAR, both beginning of life (minimum reactivity feedback) and end of life (maximum reactivity feedback) assumptions are analyzed. Although the maximum reactivity feedback cases are unaffected by a positive MTC, they were analyzed for completeness.

The following sections provide discussions for each of the FSAR events.

II. Events Not Affected by a Positive Moderator Temperature Coefficient

The following events were not analyzed because they either result in a reduction in reactor coolant system temperature, and are therefore sensitive to a negative MTC, or are otherwise not affected by a positive MTC.

A. Feedwater System Malfunctions that Result in an Increase in Feedwater Flow and a Decrease in Feedwater Temperature

The addition of excessive feedwater or the reduction of feedwater temperature are excessive heat removal events, and are consequently most sensitive to a negative MTC. Results presented in Section 15.1.1 and 15.1.2 of the FSAR, based on a negative coefficient, represent the limiting cases. Therefore, these events were not analyzed and the conclusions of the FSAR remain valid.

B. Excessive Increase in Secondary Steam Flow

An excessive load increase event, in which the steam load exceeds the core power, results in a decrease in reactor coolant system temperature. For those cases analyzed with the reactor in manual control, the analyses presented in Section 15.1.3 of the FSAR show that the limiting case is that with a negative MTC. If the reactor is in automatic control, the control rods are withdrawn to restore the average temperature to the programmed value and thus increase the nuclear power. The analyses of these cases presented in the FSAR show that the minimum DNBR is not sensitive to a positive MTC. Therefore, the results presented in the FSAR are applicable to this event and the conclusions of the FSAR remain valid.

C. Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a steam generator relief or safety valve (steam line depressurization) results in a decrease in the RCS temperature. The analysis thus assumes the most negative MTC. The worst conditions for a steam line depressurization are therefore those analyzed in the FSAR (Section 15.1.4). The conclusions of the FSAR remain valid.

D. Steam System Piping Failure

The failure or rupture of a main steam pipe results in a rapid decrease in the RCS temperature. The analysis thus assumes a strong negative MTC. The worst conditions for a steam line break are therefore those analyzed in the FSAR (Section 15.1.5). The conclusions of the FSAR remain valid.

E. Loss of Nonemergency AC Power to the Station Auxiliaries / Loss of Normal Feedwater

The loss of nonemergency AC power and loss of normal feedwater events (presented in FSAR Sections 15.2.6 and 15.2.7) are analyzed to demonstrate the ability of the secondary system auxiliary feedwater to remove decay heat from the reactor coolant system. Following initiation of the event the reactor coolant temperature and pressure rise prior to reactor trip due to the reduced heat transfer in the steam generators. Thus, the assumption of a positive MTC could potentially result in a more limiting event.

These events are analyzed at Engineered Safty Features (ESF) design rating and a zero MTC. Based on analyses conducted for other similar Westinghouse plants, this combination is more limiting than any part power analysis with a positive MTC. Therefore, the analyses presented in the FSAR (Sections 15.2.6 and 15.2.7) remain bounding and the conclusions remain valid.

F. Feedwater System Pipe Break

The main feedwater pipe rupture accident (Section 15.2.8 of the FSAR) is analyzed to demonstrate the ability of the auxiliary feedwater system to remove decay heat from the RCS and to prevent primary and secondary system overpressurization. This ensures that the core remains geometrically intact and in place. This event is analyzed at Engineered Safty Features (ESF) design rating and a zero MTC.

Following initiation of the event the reactor coolant temperature rises prior to reactor trip due to the reduced heat transfer in the steam generators. However, following reactor trip a significant cooldown of the RCS results due to the depressurization of the faulted steam generator. During this portion of the transient there is the potential for a return to power. Therefore, the event is analyzed with a most negative MTC. The analysis presented in the FSAR (Section 15.2.8) remains bounding and the conclusions remain valid.

G. Rod Cluster Control Assembly Misalignment

Of the RCCA misalignment accidents presented in FSAR Section 15.4.3, only the dropped RCCA event is potentially affected by a positive MTC. Use of a positive MTC in the analysis would result in a larger reduction in core power level following the RCCA drop, thereby increasing the probability of a reactor trip. For the automatic rod control case, a positive MTC would result in a small increase or decrease in the power overshoot, depending upon the dropped rod worth. However, analyses at different times in cycle life demonstrate that the most limiting results are obtained with middle to end of life core characteristics. In addition, the limiting power level for this accident is at or near 100 percent power where the MTC must be close to zero or negative. Based on the above, this accident is unaffected by the proposed Technical Specification and thus the analysis was not repeated. Therefore, the conclusions of the FSAR remain valid.

H. Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

As stated in Section 15.4.6 of the FSAR, a boron dilution event cannot occur during refueling due to administrative controls which isolate the RCS from potential sources of diluted water. If a boron dilution event occurs during cold shutdown, hot standby, or startup, the automatic mitigation system terminates the dilution before the core returns critical. The assumptions used in the analyses of the above discussed modes are not adversely affected by the value of the MTC. Therefore, the positive MTC does not affect the boron dilution event in these operating modes.

The reactivity addition due to a boron dilution "at power" will cause an increase in power and reactor coolant system temperature. The "at power" boron dilution events are analyzed to determine the amount of time available for the operator to terminate the dilution prior to the loss of shutdown margin. The amount of time available is based on the initial and critical boron concentrations. The implementation of a positive MTC potentally increases the initial and/or critical boron concentration. However, it is estimated that the current analysis assumptions for the boron concentrations remain bounding for a positive MTC. This assumption will be confirmed as part of the Cycle 2 Reload Safety Evaluation.

For the manual rod control case, the time to alert the operator that a dilution is in progress is based on the time to reactor trip as determined from the results of the rod withdrawal at power event. Specifically, the equivalent reactivity insertion, resulting from the dilution, is used to determine the time to reactor trip. Based on the analyses presented in Section III.F, the current times presented in the FSAR remains bounding. Therefore, the conclusions of the FSAR remain valid.

I. <u>Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position</u>

Fuel and core loading errors (e.g. due to the inadvertent loading of one or more assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or loading a full fuel assembly during manufacture with pellets of the wrong enrichment) will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

The power distribution due to any combination of misplaced assemblies would significantly raise peaking factors and would be readily observable with the in-core flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. In-core flux measurements are taken during the startup subsequent to every refueling operation.

Analyses were performed to demonstrate the adequacy of the in-core instrumentation to detect misloaded assemblies. Since this is a static event and thus no temperature rise occurs, the positive MTC has no effect on the analysis. The conclusions of the FSAR remain valid.

J. Inadvertent Operation of the Emergency Core Cooling System During Power Operation

Analysis of a spurious actuation of safety injection at power is presented in Section 15.5.1 of the FSAR. This transient results in a decrease in RCS temperature and core power and the results are not adversly affected by a positive MTC. Therefore, this event was not analyzed with a positive moderator coefficient. The conclusions of the FSAR remain valid.

K. Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures (Inside Containment)

A secondary system pipe rupture (inside containment), from which the mass and energy releases are generated, results in a decrease in the RCS temperature. The event is analyzed with a most negative MTC. Therefore, the mass and energy releases presented in the FSAR (Section 6.2.1.4) remain bounding and valid.

L. Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures (Outside Containment)

A secondary system pipe rupture (outside containment), from which the mass and energy releases are generated, results in a decrease in the RCS temperature. The event is analyzed with a most negative MTC. Therefore, the mass and energy releases presented in References 1 and 2 remain bounding and valid.

M. Loss of Coolant Accident (LOCA)

1. Large Break LOCA

The large break LOCA analyses (FSAR Section 15.6.5) are performed at 102% of the rated core thermal power which has been demonstrated to be the limiting condition for an "at power" large break LOCA. At this power level, the MTC must be zero or negative. Therefore, there is no effect of a positive MTC on the large break LOCA results. Further, the large break LOCAs calculate that the reactor cora is shutdown rapidly by the voids generated in the reactor core which will limit the effect of the MTC on core power production. Therefore, the Comanche Peak large break LOCA analyses are unaffected by the assumptions of a positive MTC provided that the MTC is zero or negative at 100% rated thermal power. The conclusions of the FSAR remain valid.

2. Small Break LOCA

Small break LOCA analyses (FSAR Section 15.6.5) are performed at 102% of the rated core thermal power which has been demonstrated to be the limiting condition for an "at power" small break LOCA. Because the reactor core design is required to have a zero or negative MTC at 100% power, there is no effect of a positive MTC on the small break LOCA analyses. Additionally, small break LOCA

analyses credit the RCCAs shortly after the initiation of the event to shutdown the reactor. During a small break LOCA, the reactor power and temperature drop rapidly. Thus, the presence of a positive MTC at lower power levels will tend to shutdown the reactor faster than in the presence of a negative MTC. Therefore, the proposed positive MTC Technical Specification will not change the results of the small break LOCA analyses. The conclusions of the FSAR remain valid.

3. <u>LOCA Blowdown Reactor Vessel and RCS Loop Forces</u>

The LOCA analyses performed to determine the hydraulic forcing functions (FSAR Section 15.6.5) are performed at 100% of the rated core thermal power. The analysis calculates the maximum consequences within 500 milliseconds of break which results in very little change to the system parameters and core power. Therefore, the implementation of proposed positive MTC Technical Specification will not affect the calculated consequences for the LOCA hydraulic forcing function. The conclusions of the FSAR remain valid.

4. <u>LOCA Hot Leg Switchover of the ECCS to Prevent Potential Boron Precipitation</u>

The calculations performed to determine the time necessary to switch the ECCS to hot leg recirculation assume that the reactor has been shutdown and the only power production is from decay heat. The decay heat model assumed in the analyses (1971 ANS decay heat model) assumes an infinite burnup with a 20% uncertainty factor and is unaffected by the proposed positive MTC Technical Specification. Therefore, the time at which swithcover of the ECCS is required is unaffected.

6. <u>Post-LOCA Longterm Core Cooling</u>

The requirement to keep the reactor core subcritical post-LOCA with All control Rods Out (ARO) is affected by the reactivity of the core under cold conditions. Because a positive MTC can increase the cold reactivity of a reactor core, the potential exists to require an increase in the Refueling Water Storage Tank (RWST) minimum boron concentration. However, the Cycle 2 core is scheduled to be a 12 month design and the expectation is that the current RWST minimum boron concentration will continue to provide the required negative reactivity. This evaluation will be provided in the Cycle 2 Reload Safety Evaluation.

III. Events Analyzed for a Positive Moderator Coefficient

The following transients were analyzed because they are sensitive to a positive MTC. As noted in the introduction, for some of the events analyzed at full power, a zero MTC was assumed to obtain acceptable results. Justification for this is based on analyses, performed for similar Westinghouse plants, which have demonstrated that a full power - zero MTC combination bounds all lower power - allowable positive MTC combinations. In addition, for some of the events presented in the FSAR, both minimum and maximum reactivity feedback cases are analyzed. Athough the maximum reactivity feedback cases are not affected by a positive MTC, they were analyzed for completeness.

A. Turbine Trip

Introduction

Two cases, analyzed for both minimum and maximum reactivity feedback, are presented in Sections 15.2.3 of the FSAR:

1. Full credit is taken for the effect of the pressurizer spray and the pressurizer power operated relief valves. Safety valves are also available.

2. No credit is taken for the effect of the pressurizer spray or power operated relief valves. Safety valves are operable.

Although the most negative MTC cases remain unaffected, all cases were repeated for completeness.

The result of a turbine trip or loss of load is a core power level which exceeds the secondary system heat removal causing an increase in core water temperature. The consequences of the reactivity addition due to a positive MTC are increases in both peak nuclear power and RCS pressure.

Method of Analysis

A constant MTC of +5 pcm/°F was assumed. The method of analysis and assumptions used were otherwise in accordance with those presented in the FSAR.

Results and Conclusions

The system transient response to a total loss of load or turbine trip from full power with minimum reactivity feedback and pressurizer pressure control was calculated. The peak RCS pressure reaches 2585 psia following a reactor trip on high pressurizer pressure. A minimum DNBR well above the limit value is reached shortly after reactor trip.

The transient response to a loss of load assuming no credit for pressure control was also calculated. Peak RCS pressure reaches 2626 psia following reactor trip on high pressurizer pressure. The minimum DNBR occurs at the initiation of the transient.

The system transient response to a total loss of load from full power with maximum reactivity feedback and pressurizer pressure control was calculated. The peak RCS pressure reaches 2464 psia following a reactor trip on steam generator low-low water level. The DNBR increases from the initiation of the event.

The transient response to a loss of load assuming no credit for pressure control was also calculated. Peak RCS pressure reaches 2597 psia following reactor trip on high pressurizer pressure. The DNBR increases throughout the transient.

The analyses of the cases discussed above demonstrates that the integrity of the core and the reactor coolant system pressure boundary during a turbine trip or loss of load transient will not be impacted by a +5 pcm/°F MTC since the minimum DNB ratio remains well above the limit value, and the peak reactor coolant system pressure is less than the event acceptance criterion of 110 percent of the RCS design value of 2500 psia. In addition, the steam generator safety valves limit the secondary steam pressure conditions to less than 110 percent of the design pressure. Therefore, the conclusions presented in the FSAR remain valid.

B. Partial and Complete Loss of Forced Reactor Coolant Flow

Introduction

The loss of flow events presented in FSAR Sections 15.3.1 and 15.3.2 were analyzed to determine the effect of a +5 pcm/°F MTC on the nuclear power transient and the resultant minimum DNBR reached during the events. The effect on the nuclear power transient would be limited to the initial stages of the event during which reactor coolant temperature increases; this increase is terminated shortly after reactor trip.

Method of Analysis

Analysis methods and assumptions used in the analysis were consistent with those employed in the FSAR. The digital computer codes used to calculate the flow coastdown and resulting system

transient were the same as those used to perform the analysis described in the FSAR. The analysis of the partial loss of flow assumed a constant MTC of +5 pcm/°F. For the analysis of the complete loss of flow event, a 0 pcm/°F MTC was assumed. Analyses have been performed which demonstrate that the combination of a 0 pcm/°F MTC at full power bounds any lower power - positive MTC combination.

Analyses were also performed for the underfrequency event assuming a frequency decay of 5 hz/sec. The analysis assumed a 0 pcm/°F MTC. Analyses have been performed which demonstrate that the combination of a 0 pcm/°F MTC at full power bounds any lower power - positive MTC combination. The analysis assumptions for this event were otherwise consistent with those made for the complete loss of flow event.

Results and Conclusions

Results of the analysis show that the minimum DNBR remains above the limit value for these transients. Therefore, the conclusions of the FSAR analyses remain valid.

C. Reactor Coolant Pump Shaft Seizure (Locked Rotor)/Shaft Break

Introduction

The case presented in the FSAR (Section 15.3.3) for this transient was analyzed. Following a shaft seizure/locked rotor event, the reactor coolant system temperature rises until shortly after reactor trip. Because DNB is conservatively assumed to occur at the beginning of the event, a positive MTC will not affect the time to DNB. The transient was analyzed, however, to assess the effect on the nuclear power transient and thus on the peak reactor coolant system pressure and fuel and clad temperatures.

Method of Analysis

The digital computer codes used in the analysis to evaluate the pressure transient and thermal transient were the same as those used in the FSAR. The analysis employed a constant MTC of +5 pcm/°F. Other assumptions used were consistent with those employed in the FSAR.

Results and Conclusions

Analysis of the locked rotor event with a +5 pcm/°F MTC shows that the peak reactor coolant system pressure remains below that which would cause stresses to exceed the faulted condition stress limits. The peak clad average temperature at the "hot spot" was determined to be 1778°F which is much less than the 2700°F limit. The amount of zirconium - water reaction at the "hot spot" was calculated to be less than 1% which is less than the limit of 16%. Also, the peak RCS pressure was calculated to be 2578 psia which is less than 110% of the design RCS pressure. Therefore, the conclusions presented in the FSAR remain valid.

D. Locked Rotor - Rods-in-DNB

Introduction

Although this event is not specifically discussed in the FSAR Section 15.3.3, in response to Question Q212.66 it is noted that for the locked rotor events the maximum number of rods experiencing DNB is conservatively calculated to be less than 7%. To confirm that this value remains valid with the implemantation of a positive MTC of 5 pcm/°F, the event was analyzed.

Method of Analysis

The digital computer codes used in the analysis to determine the percentage of rods-in-DNB were the same as those discussed in FSAR Section 15.3.3. The analysis employed a 0 pcm/°F MTC. Analysis have been performed which demonstrate that the combination of a 0 pcm/°F MTC at full power bounds any lower power - positive MTC combination. Analysis assumptions were made to maximize the heat flux and thus minimize the DNB, consistent with the current FSAR.

Results and Conclusions

Analysis of the locked rotor rods-in-DNB event shows that the percentage of rods experiencing DNB is less than 7%. Therefore, the conclusions of the FSAR remain valid.

E. Uncontrolled Rod Cluster Control Assembly (Bank) Withdrawal from a Subcritical Condition or Low Power Startup Condition

Introduction

An uncontrolled RCCA (Bank) withdrawal from subcritical event results in an uncontrolled addition of reactivity leading to a power excursion (Section 15.4.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the negative reactivity feedback of the Doppler coefficient. The power excursion causes a heatup of the moderator and fuel. The reactivity addition due to a positive MTC results in increases in the peak heat flux and peak fuel and clad temperatures.

Method of Analysis

The analysis was performed assuming a reactivity insertion rate of 75 pcm/sec. This reactivity insertion rate was used in this analysis and is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). The analysis used a MTC more conservative than a +5 pcm/°F for all appropriate temperature values. The computer codes, initial conditions, and other assumptions are consistent with those described in the FSAR.

Results and Conclusions

Analysis of this event assuming a 75 pcm/sec insertion rate, coupled with a positive MTC of +5 pcm/°F results in a rapid increase in the heat flux. The analysis of this event demonstrates that the minimum DNBR remains greater that the safety analysis limit value. Thus, no fuel or clad damage is predicted as a result of DNB. Therefore, the conclusions of the FSAR remain valid.

F. Uncontrolled Rod Cluster Control Assembly (Bank) Withdrawal at Power

Introduction

An uncontrolled bank withdrawal at power produces a mismatch in steam flow and core power, resulting in an increase in reactor coolant temperature. A positive MTC increases the power mismatch resulting in a faster heatup of the reactor coolant. However, this effect is offset by the fact that the faster heatup and reactivity addition result in an earlier reactor trip from either overtemperature N-16 or high neutron flux. A discussion of this event is presented in Section 15.4.2 of the FSAR.

Method of Analysis

The transient was analyzed employing the same digital computer code and assumptions discussed in the FSAR. This transient was analyzed at power levels of 100, 60 and 10 percent of rated thermal power with a +5 pcm/°F MTC.

Results and Conclusions

For each initial power level a full range of reactivity insertion rates from 110 pcm/sec to 1.0 pcm/sec was analyzed. In all cases, the DNBR remains above the safety analysis limit value. These results demonstrate that the conclusions presented in the FSAR remain valid. That is, the core and reactor coolant system are not adversely affected since the high neutron flux and overtemperature N-16 trips prevent the core minimum DNBR from falling below the safety analysis limit value for this event. In addition, the RCS pressures and secondary system pressures remain below 110% of the design pressures. The conclusions of the FSAR remain valid.

G. Spectrum of Rod Cluster Control Assembly Ejection Accidents

Introduction

The rod cluster control assembly ejection (rod ejection) transient is analyzed at full power and hot zero power for both minimum (beginning of life) and maximum (end of life) reactivity feedback conditions in the FSAR. The positive MTC only affects the minimum reactivity feedback (beginning of life) cases. However, both the minimum and maximum reactivity feedback cases were analyzed for completeness. The presence of a positive MTC results in increases in the nuclear power and thus adversely affects the hot spot fuel and clad temperatures resulting from a rod ejection. A detailed discussion of this transient is presented in Section 15.4.8 of the FSAR.

Method of Analysis

The digital computer codes for analysis of the nuclear power transient and hot spot heat transfer are the same as those used in the FSAR. The ejected rod worths and transient peaking factors assumed are conservative with respect to the actual calculated values for current fuel cycles. In all cases, the ejected rod worths and transient Fq values are consistent with the current FSAR with the the exception of the zero power - end of life case which assumed an ejected rod worth of 900 pcm and an Fq of 20. The analysis used a MTC more conservative than a +5 pcm/°F for all appropriate temperature values and power levels. This is a conservative assumption because the MTC actually decreases to zero from 70 percent to 100 percent power.

Results and Conclusions

The criteria for this event include 1) the average fuel pellet enthalpy at the hot spot is less than 200 cal/gm for irradiated fuel, 2) fuel melting will be limited to less than 10 percent of fuel at the hot spot and 3) the peak reactor coolant pressure is less than that which could cause stresses to exceed the faulted condition stress limits. The full power beginning of life case resulted in a peak fuel average temperature of 3383°F and a peak fuel average enthalpy of 143 cal/gm at the hot spot. The zero power beginning of life case resulted in a peak fuel average temperature of 3056°F and a peak fuel average enthalpy of 127 cal/gm at the hot spot. The full power end of life case resulted in a peak fuel average temperature of 3458°F and a peak fuel average enthalpy of 147 cal/gm at the hot spot. The zero power end of life case resulted in a peak fuel average temperature of 3296°F and a peak fuel average enthalpy of 139 cal/gm at the hot spot. In all the cases, no centerline fuel melt was predicted.

Because fuel temperatures and enthalpies do not exceed the limits specified in the FSAR, there is no danger of sudden fuel dispersal into the coolant. The peak reactor coolant system pressure remains less than that which would cause stresses to exceed the faulted condition stress limits. In addition, because it has been demonstrated that the "hot spot" criteria are satisfied, the limit of less than 10 percent of the rods entering DNB is not violated. Therefore, the conclusions of the FSAR remain valid.

H. Inadvertent Opening of a Pressurizer Safety or Relief Valve

Introduction

The inadvertent opening of a pressurizer safety or relief valve results in the depressurization of the reactor coolant system, as discussed in FSAR Section 15.6.1. Because a safety valve is sized to relieve at a much greater flow rate than a relief valve and will therefore allow a much more rapid depressurization, the case of a safety valve opening is analyzed. This situation initially results in a rapidly decreasing reactor coolant system pressure until the hot leg saturation pressure is reached. With a positive MTC (negative density coefficient), the decrease in pressure results in an increase in core reactivity because the coolant density decreases as the pressure decreases. The most limiting case assumes the reactor is in manual control, such that the increase in core reactivity causes nuclear power and average coolant temperature to increase until the reactor trips.

Method of Analysis

The method of analysis and assumptions used were the same as those presented in the FSAR except for the following:

1. A constant MTC of +5 pcm/°F was assumed.

2. The reactor was assumed to operate in the manual mode of operation to prevent rod insertion prior to reactor trip.

3. A least negative Doppler-only power coefficient of reactivity was assumed to maximize any power increase caused by the positive MTC.

Results and Conclusions

The system transient response to the inadvertent opening of a pressurizer safety valve with the reactor in manual rod control in the presence of a positive MTC was analyzed. A reactor trip occurs on the overtemperature N-16 protection function. The minimum DNBR occurs shortly after the control rods begin to drop into the core.

The analysis demonstrates that the integrity of the core during a reactor coolant system depressurization transient is not adversely affected by a positive MTC because the minimum DNBR remains above the safety analysis limit value. Therefore, the conclusions presented in the FSAR remain valid.

SECTION 3

CONCLUSIONS

To assess the effect on the accident analyses of operation of Comanche Peak Unit 1 with a positive MTC of +5 pcm/°F, analyses of transients sensitive to a positive MTC were performed. These transients included the control rod assembly withdrawal from subcritical, control rod assembly withdrawal at power, loss of reactor coolant flow, locked rotor, turbine trip, control rod ejection, and RCS depressurization events. The remaining transients, i.e., LOCA events, excessive load increase, feedwater malfunction, steam system piping failure, inadvertent opening of a steam generator relief or safety valve, loss of normal feedwater, feedwater system pipe break, etc., as discussed within, are not affected.

The analyses employed a constant moderator coefficient of +5 pcm/°F, independent of power level, except as noted. For the complete loss of flow, underfrequency loss of flow and locked rotor rods-in-DNB analyses, a 0 pcm/°F MTC was assumed. For these events, analyses have been performed for other similar Westinghouse plants which demonstrate that a full power 0 pcm/°F MTC combination bounds any lower power - positive MTC combination.

The results of this study demonstrate that the proposed positive MTC Technical Specification diagrammed in Figure 1 is supported by the safety analyses. Therefore, the conclusions of the FSAR remain valid.

SECTION 4

TECHNICAL SPECIFICATIONS

This proposed mark up does not reflect Technical Specifications which employ COLR.