UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DESIGNATED ORIGINAL

Certified By Patricia J Noonan

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-1

PORTLAND GENERAL ELECTRIC COMPANY

THE CITY OF EUGENE, OREGON

PACIFIC POWER AND LIGHT COMPANY

TROJAN NUCLEAR PLANT

DOCKET NO. 50-344

Introduction

By letter dated May 10, 1982, Portland General Electric Company requested changes to the Trojan Technical Specifications on F_{xy}, $F_{\Delta H}$ and moderator temperature coefficient. Since that time the staff has had several conversations with the licensee and additional material was submitted by letters dated July 7 and July 29, 1982.

$F_{\Delta H}$

The licensee requested that the partial power multiplier in the Technical Specification be changed from 0.2 to 0.3. The July 7, 1982 letter supplied additional material to support this change.

This change was requested to allow optimization of the core loading pattern by minimizing restrictions on the $F_{\Delta H}$ at low power.

Trojan core limits and axial offset limits for an increased allowable $F_{\Delta H}$ at reduced power levels were determined. The core limits at 1775, 2000 and 2250 psia remain unchanged from the current limits. At 2400 psia the proposed core limits are slightly more limiting below 100 percent power. Results of the Trojan $F_{\Delta H}$ Technical Specification limit analysis indicate this change may be made without changing any other Technical Specification limits.

We have previously reviewed and approved a similar change in WCAP-9500 and find this change for Trojan acceptable.

$\underline{F_{xy}}$

A change to the Technical Specifications on F_{xy} was requested to remove the cycle dependent values of F_{xy} as a function of core height and provide these F_{xy} values by means of a Peaking Factor Limit Report. It is anticipated that F_{xy} will change from cycle to cycle and this change would eliminate the necessity of making a technical specification change for each reload. A similar change has been approved for Farley Units 1 and 2. The wording of the Technical Specification has been changed from the original submittal and is included in this SER as an enclosure. A Radial Peaking Factor Limit Report was submitted with the May 10, 1982 letter for Cycle 5. We find this change acceptable.

The licensee requested that the partial power multiplier be changed from 0.2 to 0.3 for F_{xy} also. We do not think that this change is appropriate and the licensee has not been able to submit further justification so this change cannot be approved at this time.

Moderator Temperature Coefficient

The licensee requested that the moderator temperature coefficient (MTC) limits be revised to $\leq 0.5 \times 10^{-4}$ $^{\Delta k}/K/°F$ below 70 percent power and MTC of ≤ 0 at or above 70 percent power. This will allow operation with a positive MTC at power levels up to 70 percent rated thermal power.

By way of background, cores of this type have a slightly positive MTC at beginning of life, hot zero power, all control rods out. The coefficient becomes zero if the boron concentration is reduced by swapping with control rods, going to part power, or building in Xenon. What this means is, in order to get the reactor on line after a refueling outage, the reactor operator must observe temporary control rod insertion limits as a function of power level to keep the MTC zero. Performing calculations to generate the insertion limits and the extra care required to observe the limit delays the startup sequence. We have approved operation with slightly positive MTCs on a number of occasions in the past.

The licensee has assessed the impact of a positive MTC on the accident analysis presented in Chapter 15 of the Trojan FSAR. Those incidents which were found to be sensitive to positive or near-zero moderator coefficients were reanalyzed. The licensee's conclusions about which transients did and did not require reanalysis were the same as those made by previous licensees requesting a positive MTC.

The incidents reanalyzed, with one exception, used a +5 pcm/°F moderator temperature coefficent, assumed to remain constant for variation in temperature. (Note that a pcm is equal to a reactivity of $10^{-5}\Delta$ K/K.) This is conservative since the proposed change will require a zero coefficient at power levels above 70%. The exception is the rod ejection accident, for which the computer model cannot accept a constant coefficient. The control rod ejection analysis was based on a coefficient which was at least +5pcm/°F at zero power nominal average temperature, and which became less positive for higher temperatures. The assumption of a positive MTC at full power is conservative since the proposed Technical Specification requires the MTC be zero or negative at or above 70% power.

The transients and accidents that were reviewed and their results are:

<u>Control Rod Withdrawal From A Subcritical Condition</u>. The results of the reanalysis of this transient produced values for peak heat flux, peak coolant temperature and thermal power which do not exceed nominal full power values. Therefore, the conclusions in the FSAR are still valid.

<u>Uncontrolled Control Rod Assembly Withdrawal at Power</u>. The results of this reanalysis show that the core and reactor coolant system are not adversely affected since nuclear flux and overtemperature ΔT trips prevent the core minimum DNB ratio from falling below 1.73 for this incident. Thus the conclusions of the Cycle 2 Reload Safety Evaluation remain valid.

Control Rod Ejection

The results of reanalysis showed that the limited peak hot spot clad temperature was 2675°F. Peak hot spot fuel centerline temperature exceeded the melting point, but melting was restricted to less than the innermost 10 percent of the pellet. All fuel and clad temperatures are within the limits specified in the FSAR.

Since the reanalysis of these transients did not result in exceeding any of the limits specified in the existing analyses for Trojan, we conclude the proposed MTC Technical Specification change will not result in any significant loss of safety margins and is therefore acceptable.

The transients that were reanalyzed were limited to those which cause the reactor coolant system (RCS) temperature to increase. Transients that result in a reduction in RCS temperature for which a negative MTC is more limiting; and those for which heatup prior to reactor trip is small were not reanalyzed.

The transients not reanalyzed are:

A. RCCA Misalignment/Drop
B. Startup of an Inactive RCS Loop
C. Excessive Heat Removal Due to Feedwater System Malfunctions
D. Excessive Load Increase
E. Loss of Normal Feedwater, Loss of Offsite Power
F. Accidental Depressurization of the Reactor Coolant System
G. Spurious Actuation of Safety Injection
H. Rupture of a Main Steam Pipe
I. Rupture of a Main Feedwater Pipe (Feedwater Line Break)
J. Loss of Coolant Accident (LOCA)

Heat-up transients from this list include Loss of Normal Feedwater, Loss of Offsite Power and Feedwater Line Break. Heat-up prior to trip in the Loss of Normal Feedwater and Loss of Offsite Power transients is indeed small as indicated by FSAR analyses. FSAR analyses of the Feedwater line break accident from full power show about a $40^{\circ}F$ RCS temperature rise prior to reactor trip at 20 seconds. This translates to a moderator induced reactivity addition rate of no greater than $+1.0 \times 10^{-4}$ $\Delta K/K/sec$ (i.e., $40^{\circ}F/20$ sec. x $.5 \times 10^{-4}$ $\Delta K/K/^{\circ}F$). The resulting power transient and degradation in DNB margin would be terminated by the overtemperature ΔT trip.

The reactor system transients analyzed and their results are:

Boron Dilution

The reactivity addition due to a boron dilution at power will cause an increase in power and RCS temperature. Due to the temperature increase a positive MTC would add additional reactivity and increase the severity of the transient. With the reactor in automatic control, the rod insertion alarms provide the operator with adequate time to terminate the dilution before shutdown margin is lost. With the reactor in manual control the boron dilution incident is no more severe than a rod withdrawal at power.

Loss of Coolant Flow

The most severe loss of flow transient is caused by the simultaneous loss of power to all four reactor coolant pumps (RCPs). This case was reanalyzed to determine the effect of a positive MTC on the nuclear power transient and the resultant effect on the minimum DNBR reached during the transient. The RCS temperature increases $2^{\circ}F$ above the initial value and the minimum DNBR remains above the limit value of 1.73. Since this is the limiting loss of flow transient and since the DNBR ratio remains above 1.73, the results from the cycle 2 Reload Safety Evaluation are still valid and acceptable.

Locked Rotor

The Locked Rotor event was reanalyzed because of the potential effect of positive MTC on the nuclear power transient and, thus, on RCS pressure and fuel temperature. A positive MTC will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the incident. Analyses were performed for the case of four pump operation at 102% power and three pump operation at 72% power. An MTC of +5 pcm/$^{\circ}F$ was assumed in both cases. Results show that the full power four pump case is the worst. For this case, the peak RCS pressure remains below that which would cause stresses to exceed the faulted condition stress limits; and the peak clad temperature for the hot spot remains much less than $2700^{\circ}F$. Thus, the conclusions presented in the FSAR remain valid.

Loss of External Electric Load

The loss of external electric load transient was reanalyzed for beginning of cycle (BOC) since the MTC will be negative at end of cycle (EOC) and will give the same results as in the FSAR. Two cases were analyzed: (1) reactor in the automatic rod control mode with operation of the pressurizer spray and pressurizer power operated relief valves (PORV); and (2) reactor in the manual control mode with no credit for pressurizer spray or PORV's. The result of a loss of load is a core power that momentarily exceeds the secondary system power removal, causing an increase in RCS coolant temperature. The reactivity addition due to a positive MTC, causes an increase in both nuclear power and RCS pressure. The result for the control rods in the automatic control and assuming pressurizer spray and relief is an RCS pressure of 2497 psia following a reactor trip on overtemperature ΔT. A minimum DNBR of 2.15 is reached shortly after reactor trip. The result for the case of rods in manual control with no credit for pressure control is a peak RCS pressure of 2542 psia following a reactor trip on high pressure. The minimum DNBR is initially 2.26 and increases throughout the transient. Since the DNB ratio remains above 1.73 and the peak RCS pressure is less than 110% of design the conclusions presented in the cycle 2 Reload Safety Evaluation are still applicable.

Reanalysis of the affected reactor system transients shows that none will exceed any of the fuel limits or safety limits specified in the Trojan FSAR or Cycle 2 Reload Safety Evaluation. We conclude that the proposed Technical Specification, as it relates to the systems and analyses discussed in this evaluation, will not pose an undue risk to the health and safety of the public, and is therefore acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: August 13, 1982

Principal Contributors:
M. Chatterton
M. Caruso

Enclosure: Radial Peaking
Factor Limit Report

ENCLOSURE

Radial Peaking Factor Limit Report

The F_{xy} limits for Rated Thermal Power (F_{xy}^{RTP}) for all core planes containing bank "D" control rods and all unrodded core planes and the plot of predicted $(F_q^T \cdot P_{Rel})$ vs Axial Core Height with the limit envelope shall be provided to the NRC Regional Administrator with a copy to:

Director of Nuclear Reactor Regulation
ATTENTION: Chief, Core Performance Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter.

In addition, in the event that the limit should change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.