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Task 170 - N. I. Calibration Error
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INTRODUCTION

A concern raised during certain overcooling transients and rod ejection transients is that a transient-induced neutron flux error greater than that assumed in the FSAR could exist. This concern was presented to the Owners in a group meeting in Lynchburg on October 23 and 24, 1980. The purpose of Master Services Task 170 - NI Calibration Error is to identify plant safety margins available to offset the additional NI errors. thereby providing justification for full power operation of Rancho Seco. Cycle 5.

SUMMARY AND CONCLUSIONS

A concern was raised that overcooling events could result in an induced neutron flux measurement error which might increase as reactor coolant temperature decreases. This error could result in the actual core power level exceeding the assumed 112% overpower condition before a reactor trip occurs. However, an evaluation has shown that the increase in core power is offset by the beneficial effect of the temperature decrease on core thermal margins. DNBR analyses performed for the most limiting condition (indicated power at high flux trip limit of 105.5%, RC pressure at low pressure trip limit of 1900 psig) demonstrate that the minimum DNBR will be greater than 1.30 for conditions under which a reactor trip would be initiated at core power levels up to 135% of rated power. In addition, cycle 5 calculated core power distributions at all allowable rod index and Axial Power Shaping Rod (ASPR) positions for normal full power operation were examined and margin existed for both DNB and CFM assuming an actual core power of 125%. It is therefore concluded that the induced flux measurement error does not compromise the safe operation of Rancho Seco, Cycle 5 during overcooling events initiated from anywhere within the allowable operating range.

The concern on the rod ejection transient is that the high flux trip may not be activated for an ejected rod of small worth less than .2% k/k Under these conditions. current models would show unacceptable results (peak fuel enthalpy > 280 cal/gram). Although no reanalysis has been performed. an engineering evaluation of the conservatisms in the original analysis (such as adiabatic heatup) has led to the conclusion that a reanalysis using realistic assumptions, will show that the peak fuel enthalpy will not exceed 280 cal/gram. Therefore. this concern is not considered to compromise the safe operation of Rancho Seco, Cycle 5. A further evaluation of the small worth ejected rod accident is provided in Attachment 1.

OVERCOOLING ACCIDENT ANALYSIS

The current flux error assumptions in the FSAR are:

2.0% Heat Balance

2.0% Steady State Neutron Measurement

2.0% Transient-Induced Neutron Measurement

<u>0.5%</u> Instrumentation

6.5% Total

The 2.0% transient induced neutron measurement error is sufficient to accomodate all transient induced errors excluding those which are the topic of this report. The additional transient induced error was discovered during a study for the WPPSS - WP 1/4 FSAR. a 205 plant. Attachment 2 provides an evaluation of the induced NI error during certain overcooling events.

Although no transient analysis has been performed specifically for Rancho Seco. an engineering evaluation based on the WPPSS analysis has shown that the maximum transient induced error for moderate frequency overcooling transients will be approximately 13%. Therefore. for moderate frequency overcooling transients only. the instrumentation error that should be considered is·

2.0%	Heat Balance
2.0%	Steady State Neutron Measurement
0 to 13%	Transient Neutron Measurement Dependent on Coolant Temperatures
0.5%	Instrumentation
17.5%	Total

To justify full power operation. one must demonstrate that operation up to 123% full power is acceptable during these overcooling transients. This power level is based on a high flux trip setpoint of 105.5% full power plus a total error of 17.5%. It should be remembered that this power level could only be reached during certain overcooling transients that provided specific core conditions.

The analysis of induced flux errors during overcooling transients led to the quantification of the ratios of indicated power to actual core power as a function of downcomer fluid temperature and core average coolant temperature. The primary concern is to determine the conditions that would permit the actual core power to exceed 112% without a reactor trip occurring. The error calculations were used to determine the maximum actual core power as a function of temperature for the case where the indicated power would be 105.5% which is the high flux trip setpoint (Figure 1). A series of heat balance calculations were then performed. using the minimum licensed RCS flowrate (387, 710 GPM). to determine the corresponding core operating conditions. When the heat balance is superimposed on the curves of Figure 1. the result is a single line. for any given pressure. which defines the actual core power as a function of coolant inlet and average temperature. consistent with the assumed constant indicated power level of 105.5%. This line is shown (dashed) on Figure 1 plotted against downcomer temperature. It can be seen by examination of Figure 1 that core operation under conditions below or to the left of the power vs. temperature line would be less restrictive (lower power and temperature). Also. operation above and to the right of this line would be prevented because the indicated power level would be greater than 105.5%. thus resulting in a reactor trip.

In order to quantify core thermal margin for the conditions corresponding to operation at an indicated power level of 105.5%. DNBR calculations were performed using the CHATA and TEMP codes. All 2200 psia RCS pressure points allowed by the RPS and corresponding to operation with indicated power equal to 105.5%. lie well above the Tech Spec minimum DNBR of 1.30 for the B&W-2 correlation. For low pressures corresponding to the RPS low pressure setpoint (1900 psig). the variable low pressure trip provides a trip if the RCS outlet temperature exceeds $597.8^{o}F$. This trip function then provides protection to a minimum DNBR of 1.3 for core power levels up to $\approx$ 123% FP (Figure 2). In addition. the high flux trip provides protection to the minimum DNBR of 1.3 for power levels up to 135% FP. Any operation at the right of these limits as plotted is prevented by the Reactor Protection System. In order to quantify DNBR margin along this line. a parameter study was performed to determine the effect of coolant temperature variation on DNBR at 1900 psig and constant power

The calculations described above are based upon reference design peaking conditions (1.714 radial x local peak and a 1.5 core mid-plane axial peak). The applicability of the design peaking to core power shapes is ensured by the use of Maximum Allowable Peaking (MAP) curves The MAP limits define the maximum peak allowed over a range of core elevations and axial peaks. These maximum allowable peak combinations represent radial and axial peaking for which the calculated MDNBR is the same as for the design peaking conditions. The MAP limits are used in the maneuvering analyses as an acceptance criterion for the development of power imbalance limits (normal operating or RPS limits).

Three-dimensional power distribution calculations were performed to assess the core power distribution perturbation at 125% FP due to an overcooling transient, and to determine the margins to centerline fuel melt (CFM) and departure from nucleate boiling (DNB) limits. Identical calculations were generated from normal steady- state operation at 100% FP and from operation at 125% FP with a 16^{o}F inlet temperature reduction. All power distribution calculations were initiated from within or near the normal rod index, APSR and axial imbalance limits of operation, such that the core behavior over the entire allowable operating range was examined.

CFM and DNB margins were computed for the 125% FP cases to determine if core safety limits would be preserved during an overcooling transient. Since all calculations were performed from near steady-state conditions, appropriate peaking factors were included in the 125% FP margins calculations to account for potential peaking increases due to transient xenon and quadrant tilt. Maximum allowable peaking curves for Rancho Seco, Cycle 5 were used to evaluate the DNB margins. The applicability of these curves at 125% FP was verified by DNBR analyses performed for the limiting cases.

RESULTS

The increased power resulting from certain overcooling transients is accommodat.d within the cycle 5 Normal Operating Limits. The analysis of the design peaking distribution yielded DNBR margins in excess of 6.7% DNBR relative to the 1.30 DNBR limit. Additionally, analysis of the most limiting peaking distribution identified by maneuvering analyses yielded DNBR peaking margins as shown in Table 1. The high flux trip provides DNBR protection to the minimum DNBR limit up to core power levels of 135%FP.

TABLE 1

RANCHO SECO Cycle 5

Minimum CFM DNB Peaking Margins

(Most Limiting Conditions)

Cycle Burnup (EFPD)	Rod Index (%WD)	APSR Index (%WD)	CFM Margin* (%)	DNB Margin* (%)
125	300	12	19.93	4.68
125	287	12	17.39	7.05
244	300	100	15.17	8.55
244	273	100	.54	10.23

* Margin is quoted in terms of relative peaking compared to the limit value.

ATTACHMENT 1

Since the ejected-rod accident analyzed in the FSAR for .65% Δ k/k worth rod is terminated by a high flux trip, concern has been expressed that if no high flux trip occurred during ejection of rods with worths less than .2% Δ k/k, that fuel pin failure might occur. Current models used for FSAR analysis do not consider energy transfer from the local pin to the reactor coolant, therefore, if used for analysis of the small worth rod ejection accident could show unacceptable results (peak fuel enthalpy > 280 cal/gm).

For feed and bleed operation of a B&W core, the highest worth ejected rod at power is extremely low < .1% Δ k/k, this is because the transient bank of rods (Group 7) is normally operated > 80% wd from the core. Therefore, for an evaluation of this low worth rod ejection accident, the most important parameters for consideration are the subchannel rate of power increase and the total local power generation. The combination of these two parameters will determine the heat transfer and therefore, fuel-rod enthalpy. Contrary to the ejected rod analysis in the FSAR, where the local peaking factor doubles or triples in less than .3 seconds, the small rod ejection causes a very small local peaking change. This results because the power is depressed to the bottom of the core in the ejected-rod location before the rod ejection: after rod ejection, the axial flux shape actually flattens resulting in minimal percentage changes in both radial and total power peaking.

For the average-channel power response, the small ejected rod accident without trip on high flux provides a comparison with the large ejected rod analyzed in the FSAR. First, the peak power is considerably lower than the FSAR ejected-rod case and secondly, the local power rate of change is relatively small.

ATTACHMENT 1

An illustration for the above justification considers the power rate of change for two ejected rod cases .2 and .45% k/k and the resulting energy balance as a function of the time the power is above 100% FP. For a .2% k/k ejected rod case. the peak power is 135% FP and decreases to 112% FP in 11 seconds whereas for a .45% k/k ejected rod the peak power is 245% FP at .15 seconds and decreases to 100% FP at 1.4 seconds. Therefore. the total stored energy in the fuel pin is the difference between thermal power generation and heat removal rate. Since the power response after a small rod-ejection is slow. and the average-channel power is relatively low with an insignificant total peaking change. the heat transferred out of the pin to the reactor coolant within the 11 seconds should be significant and therefore. the peak pin fuel enthalpy will be less than 280 cal/gms.

Using the above observation and maintaining normal RCS flowrate. it should be conservative to compare the amount of power excursion between small worths and FSAR ejected-rod transients. The existing data indicates the stored energy should be between 200 to 250 cal/gm which is highly conservative yet still below the limiting value of 280 cal/gm.

ATTACHMENT 2

Since the concern of induced neutron flux error during overcooling events was raised. the important question has been to quantify the magnitude of the induced error and define the transient(s) that result in such an error. To this end. a task was undertaken to review current available data. determining the bounding moderate frequency overcooling transient and estimating the resulting induced neutron flux error for that case.

The most complete data available to evaluate was from WPSS 205 FSAR analysis. Based on the WPSS analysis, the most severe moderate frequency overcooling transient which is terminated by a high flux trip signal is a failure of the turbine bypass system (atmospheric on/off valves). This failure consists of a single atmospheric on/off valve on each of the four steam lines opening at power. The induced "break" results in an increased steam flow of $\approx 4.6 \times 10^6$ lbs./Hr. The real flux (thermal power) increases to $\approx$122% FP with an induced flux error of $\approx$13% FP. Since the indicated power does not exceed 105.5% FP. the high flux trip will not terminate the transient. This transient causes a reduction in the cold leg or downcomer temp. by $\approx 16^oF$. a reduction in core average temperature by $\approx 10^oF$ and very little change in RCS pressure.

This transient has been analyzed on Power Train IV with moderator coefficients for BOC and EOC covering a range of $0.0 \times 10^{-4}\ \Delta k/k/^oF$ to $-3.37 \times 10^{-4}\ \Delta k/k/^oF$ and with varying feedwater temperatures

The change in flux at the out-of-core detector locations due to changes in cold leg temperature were determined using the ANISN transport theory program. The induced neutron flux error was obtained by using the Power Train IV output (real power, T_{hot}, T_{ave}) and the results of the ANISN analysis.

ATTACHMENT 2

Since the Atmospheric Valve failure is considered to be the most probable and restrictive overcooling transient of concern. it is realistic to believe the 13% FP induced neutron flux error is valid for the small overcooling. moderate frequency event of interest.

Using the WPPSS Atmospheric Valve failure for determination of 177 FA plants induced neutron flux error results, involves the following assumptions

1) The specific transient response for 177 FA plants including the event timing is similar to WPPSS 205 FA plant
2) The failure of all bypass valves on 177 FA plants is similar to the failure of the Atmospheric valves event on WPPSS.
3) The magnitude of the induced neutron flux error for 177 FA plants should be similar or conservative because
   - o The total bypass flow in 177 FA plants is similar or less than the 205 plant. Also a specific single failure can be identified to cause all TBV's to open.
   - o The failure of a single steam generator bypass valve can also occur in a a 177 FA plant: however. the resulting steam flow increase will be 50 to 70% less than the WPPSS steam flow increase.

Figure 1. ACTUAL CORE POWER VERSUS INLET TEMPERATURE

RCS PRESSURE = 1900 PSIG

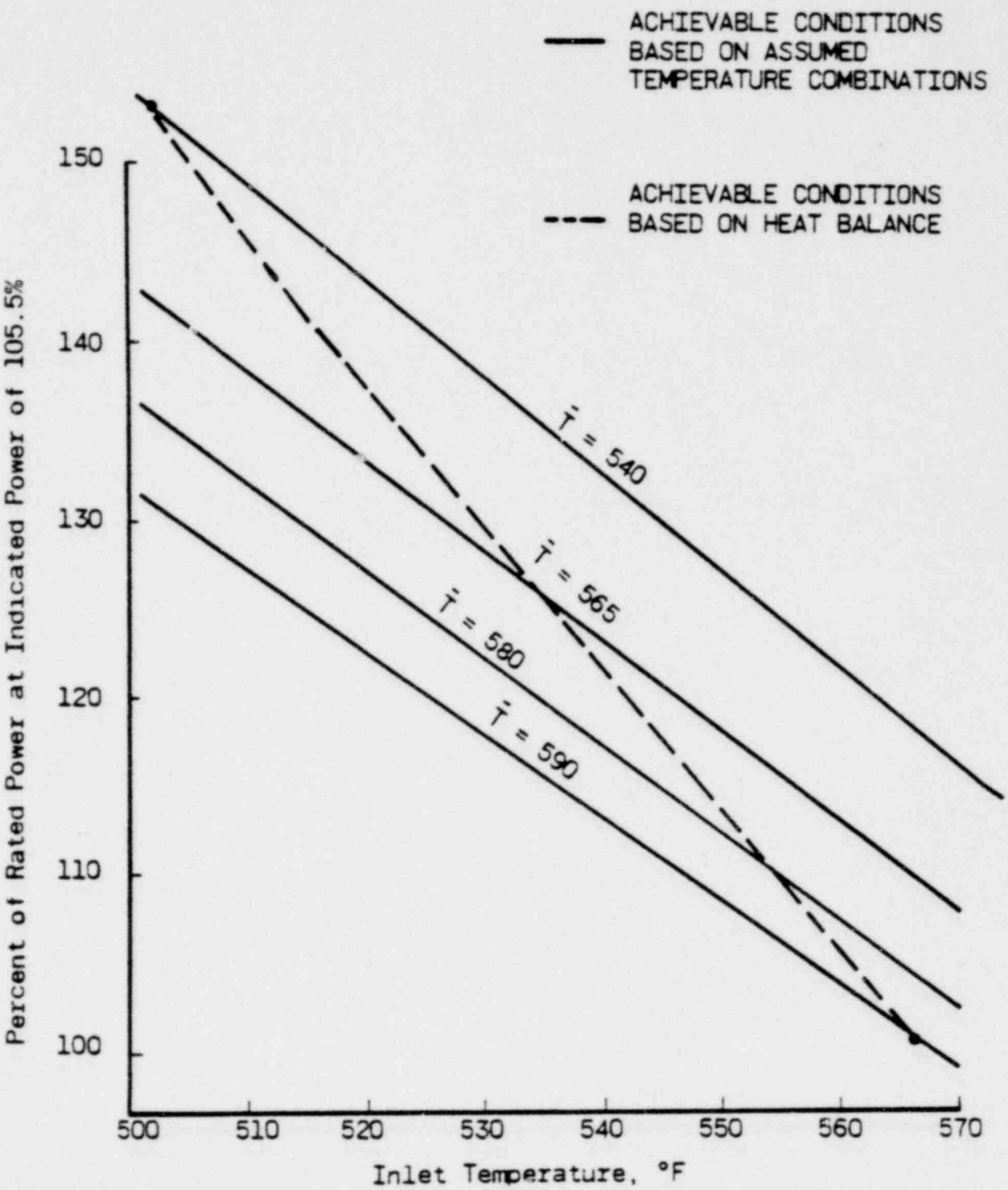

Figure 2. REACTOR PROTECTION SYSTEM AND DNBR LIMITS

RCS PRESSURE = 1900 PSIG
2772 MWt