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October 17, 1984

Director, Office of Nuclear Reactor Regulation
Attention: W. A. Paulson, Acting Chief
Operating Reactors Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Increased Power Peakings
San Onofre Nuclear Generating Station
Unit 1

Provided below is a proposed change to the basis for Technical Specification 3.5.2 "Control Group Insertion Limits." This change is proposed in connection with the safety evaluation for the Cycle 8 restart of San Onofre Unit 1. During the reload design and evaluation for each cycle, all appropriate safety parameters are evaluated. Due to the extended Cycle 8 shutdown, these safety parameters have been reevaluated. This reevaluation demonstrated that core design and safety limits for the remainder of Cycle 8 will be satisfied with the present technical specifications. However, the proposed change to the basis of Specification 3.5.2 should be made for consistency with the safety evaluation. The change submitted herein is deemed not to require submittal of a License Amendment based on the provisions of 10 CFR 50.36(a), which requires inclusion of bases for technical specifications with license applications, but indicates such bases shall not become part of the technical specifications. Therefore, your concurrence is requested to incorporate the basis change described herein.

As a result of the reevaluation discussed above, it has been determined that the $F^N_{\Delta H}$ design maximum value (1.55) in the basis of Technical Specification 3.5.2 may be exceeded and should be increased to 1.57 to accommodate calculated increased power peakings. In addition, this higher value of $F^N_{\Delta H}$ should be increased as a function of power levels less than the 1347 MWt rated core power, to be consistent with current safety analyses of the plant.

The basis for Technical Specification 3.5.2 currently reads, in part:

"1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_{\Delta H}$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

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A more restrictive limit on the design maximum value of specific power, $F^N_{\Delta H}$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. The values of the specific power, $F^N_{\Delta H}$ and F_Q are 13.7 kW/ft., 1.55 and 2.89, respectively. The control group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution."

The basis for Technical Specification 3.5.2. would be revised to read, in part:

"1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_{\Delta H}$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

A more restrictive limit on the design maximum value of specific power, $F^N_{\Delta H}$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. At 1347 MWt rated core power, the maximum values of specific power, $F^N_{\Delta H}$ and F_Q are 13.7 Kw/ft., 1.57 and 2.89, respectively. At partial power the $F^N_{\Delta H}$ maximum values (limits) increase according to the following equation, $F^N_{\Delta H}(P) = 1.57\ [1 + 0.2(1-P)]$, where P is the fraction of rated power. The control group insertion limits in conjunction with Specification B prevent exceeding these values, even assuming the most adverse Xe distribution."

The balance of the basis for Technical Specification 3.5.2 would remain as now presented in Appendix A to Provisional Operating License No. DPR-13.

A safety evaluation report for the $F^N_{\Delta H}$ increase is enclosed as Attachment A. As concluded in this evaluation, there is sufficient DNBR margin to accommodate the increased $F^N_{\Delta H}$ limit. The remaining design bases, technical specification limits, and the current accident analyses remain valid.

If you have any questions or desire additional information, please call me.

Very truly yours,

M. O. Medford

cc: E. McKenna, NRC Project Manager
J. O. Ward, Chief, Radiological Health Branch,
State Department of Health Services

ATTACHMENT A

SAFETY EVALUATION FOR A $F^N_{\Delta H}$ INCREASE

for

Southern California Edison Company

San Onofre Unit 1

Docket No. 50-206

September 1984

Editor: J. Skaritka
W-NFD

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1.0 INTRODUCTION AND SUMMARY

This report presents a safety evaluation for San Onofre Unit 1 operation with an increased $F^N_{\Delta H}$ limit as a Basis for the Technical Specification 3.5.2, control Rod Insertion Limits.

The increase in the 100% rated power $F^N_{\Delta H}$ limit from 1.55 to 1.57 is needed to accommodate calculated increased power peakings due to a 32 month Cycle 8 shutdown for plant modifications and two separated RCCA rodlets stuck in fuel assembly guide thimbles.

The incidents analyzed and reported in the FSA(1) which could potentially be affected by changes described in this report have been evaluated and are discussed in later sections.

The result of the evaluation/analysis described herein lead to the following conclusions:

1. The proposed changes to the Section 3.5.2 Basis do not impact the other design/safety bases used for the Cycle 8 RSEs(2)(3) which were submitted for NRC review(4).

2. There is sufficient DNBR margin to accommodate the reduction in margin resulting from the increased $F^N_{\Delta H}$ limit. The remaining design bases, technical specification limits and the current non-LOCA and LOCA analyses remain valid.

2.0 NUCLEAR DESIGN EVALUATION

The proposed technical specification basis changes do not impact the other nuclear design bases used to evaluate the Cycle 8 RSEs[2][3]. The standard calculational methods described in the "Westinghouse Reload Safety Evaluation Methodology"[5] continue to apply. As is current practice, each reload core design is evaluated to assure that design and safety limits are satisfied according to this reload methodology.

3.0 THERMAL-HYDRAULIC DESIGN EVALUATION

3.1 Introduction

The proposed San Onofre Unit 1 technical specification basis changes which impact DNBR evaluation is the value of $F^N_{\Delta H}$ determined from the following equation:

$$F^N_{\Delta H} \leq F^L_{\Delta H} (1 + 0.2 (1-P))$$

where $F^N_{\Delta H}$ = measured radial peaking factor with appropriate uncertainties.

$F^L_{\Delta H}$ = peaking factor limit at 100% Rated Power

P = fractional core power level at less than 100% Rated Power or,

= 1.0 at greater than or equal to 100% Rated Power.

The radial peaking factor limit ($F^L_{\Delta H}$) increase from 1.55 to 1.57 has a direct impact on DNBR calculations.

The core limits of the Technical Specification Figure 2.1.1 include a restriction that the average enthalpy at the vessel exit must be less than the enthalpy of saturated liquid to assure the proportionality between vessel ΔT and core power. The exit enthalpy restriction is more limiting than DNBR at low heat fluxes and is independent of radial peaking factor as shown in the following relation.

$$h_{out} = h_{in} + \frac{Q}{G} < h_{sat}$$

where h_{out} = average coolant enthalpy at vessel exit (BTU/lb_m)

h_{in} = vessel inlet coolant enthalpy (BTU/lb_m)

Q = total core power (BTU/hr)

G = total core coolant flow (lb_m/hr)

3.2 Justification for Increasing $F^N_{\Delta H}$

In Cycle 8, 4.4% in DNBR margin is available due to pitch reduction and adverse design axial power shape.(3) This available DNBR margin is sufficient to accommodate the reduction in DNBR margin resulting from increasing the 100% rated power $F^N_{\Delta H}$ from 1.55 to 1.57. The core limit curves (Technical Specification Figure 2.1.1) remain unchanged.

3.3 Conclusion

In summary, the effect of increasing the 100% rated power $F^N_{\Delta H}$ from 1.55 to 1.57 is offset by a reduction in DNBR margin such that the existing core DNB limits (Overtemperature ΔT) and non-Overtemperature ΔT trip accident analyses remain valid. The FSA DNBR design basis is met with the increased $F^N_{\Delta H}$.

4.0 ACCIDENT EVALUATION

4.1 Introduction

This section summarizes the effects of the increased $F^N_{\Delta H}$ limit for San Onofre Unit 1 on the FSA Chapter 14 non-LOCA and LOCA analyses.

4.2 Non-LOCA Evaluation

The methods used for accident evaluation are described in Reference 5 and are the same as those applied to previous San Onofre reload safety evaluations.

To ensure adequate core protection, the Reactor Core Thermal and Hydraulic Safety Limits were reevaluated due to the increased $F^N_{\Delta H}$ limit. Since this evaluation determined that the reactor core safety limits are not changed, the variable low pressure setpoint equation a given in the technical specification will not change. Therefore, no reanalysis is required for transients requiring the variable low pressure trip for protection.

The overall system transient response is not affected by the increased $F^N_{\Delta H}$ limit. Rather, the effect of the increased $F^N_{\Delta H}$ is accounted for in the DNBR calculations.

As noted in Section 3.0, the penalty incurred as a result of the increased $F^N_{\Delta H}$ is offset by a reduction in available DNBR margin. Therefore, the current non-LOCA accident analyses remain valid and no reanalysis is required.

4.3 LOCA Evaluation

The current large break LOCA FSAR analysis continues to establish conformity with the requirements of (Reference 6) for an increase in enthalpy rise peaking factor ($F^N_{\Delta H}$) from 1.55 to 1.57. The analysis was performed assuming a core average heat flux of 4.764 kw/ft based on a core licensed power of 1347 MW produced by 180 rods in each of 157 assemblies. The hot rod power distribution is modeled in LOCTA-R2 (Reference 7) as the limiting chopped cosine shape having a maximum heat flux of 13.74 kw/ft. The total hot rod peaking factor (F_Q^T), the ratio of the hot rod maximum heat flux to the core average heat flux, is therefore 2.89.

Keeping the core average coolant enthalpy rise constant while increasing $F^N_{\Delta H}$ represents an increase in the hot channel coolant enthalpy rise. This results in a change to the hot rod chopped cosine power distribution used in calculating the large break LOCA peak clad temperature. The change is an increase in the average heat flux of the hot rod.

The current large break LOCA analysis of record has assumed an enthalpy rise peaking factor of 1.75. Therefore, for an F_Q^T of 2.89, the analysis remains applicable and conservatively bounds any enthalpy rise peaking factor less than 1.75.

The hot rod axial power distribution assumed in the small break LOCA analysis of record continues to be applicable for an F_Q^T of 2.89 and an enthalpy rise peaking factor of 1.57.

In summary, the current large and small break LOCA analyses of record remain applicable for an enthalpy rise peaking factor of 1.57, and no further LOCA analyses or evaluations are required to support this technical specification change.

5.0 REFERENCES

1. Docket Number 50-206, "San Onofre Nuclear Generating Station,Unit 1, Part 2, Final Safety Analysis".

2. Skaritka, J., Editor, "Reload Safety Evaluation - San Onofre Unit 1, Cycle 8 - Revision 1," October 1980.

3. Skaritka, J., Editor, "Reload Safety Evaluation - San Onofre Unit 1 Cycle 8 - Revision 2", April, 1981.

4. Letter from K. P. Baskin (SCE) to D. N. Crutchfield (NRC); Subject: Amendment 95 to San Onofre Unit 1 Operating License; Docket No. 50-206; April 15, 1981.

5. Bordelon, F. M., et. al., "Westinghouse Reload Safety Evaluation Methodology", WCAP-9272 (Proprietary), March 1978.

6. Section IV.A and B, "Criteria for Emergency Core Cooling Systems for Light Water Power Reactors", Federal Register, Vol. 36, No. 125, June 29, 1971.

7. Bezella, W. A., Caso, C. L., and Spencer, A. C., "LOCTA-R2 Program: Loss of Coolant Transient Analysis", WCAP-7437-L, January, 1970.