September 16, 2008

Mr. Ross T. Ridenoure
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SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2 AND 3 – REQUEST TO REVISE SAFETY EVALUATION TO LICENSE AMENDMENT NOS. 210 AND 202 (TAC NOS. MD6759 AND MD6760)

Dear Mr. Ridenoure:

By application dated September 12, 2007 to the U.S. Nuclear Regulatory Commission (NRC), Southern California Edison (SCE, the licensee) requested removal of the computer code restriction on the use of the Departure from Nucleate Boiling (DNB) statistical convolution methodology of their SCE Alternate Source Term (AST) application. The computer code restriction is contained in the staff AST Safety Evaluation (SE) dated December 29, 2006, which states, *"However, the use of any combination of computer code, critical heat flux correlation, or fuel design, other than that explicitly approved by CENPD-183-A, will require submittal of revised probability distributions for NRC staff review and approval."* SCE agrees to the restrictions on critical heat flux correlation and fuel design; but not to the restriction on computer code.

The NRC staff finds that the information submitted in SCE's response to the NRC staff request for additional information, justifies removal of the staff computer code restriction. This revision does not change the staff's conclusions regarding Amendment Nos. 210 and 202. Results of the NRC staff review is discussed in the attached safety evaluation (SE). Also enclosed is revised Page 30 of the SE dated December 29, 2006. Additionally, revised Page 12 and Table 4 of the SE dated December 29, 2006, reflecting clarifications, are enclosed.

Sincerely,

/RA/

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosures: 1. Revised Pages 30 and 12, and Table 4 of SE dated December 29, 2006
2. Safety Evaluation

cc w/encls: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO FACILITY OPERATING LICENSE NOS. NPF-10 AND NPF-15

SOUTHERN CALIFORNIA EDISON

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By application dated September 12, 2007 to the U.S. Nuclear Regulatory Commission (NRC), (Agencywide Document Access and Management System (ADAMS) Accession No. ML072600158) Southern California Edison (SCE, the licensee), requested removal of the computer code restriction on the use of the Departure from Nucleate Boiling (DNB) statistical convolution methodology of their SCE Alternate Source Term (AST) application. Said computer code restriction is contained in the NRC staff AST Safety Evaluation (Reference 1) which states, *"However, the use of any combination of computer code, critical heat flux correlation, or fuel design, other than that explicitly approved by CENPD-183-A, will require submittal of revised probability distributions for NRC staff review and approval."* SCE agrees to the restrictions on critical heat flux correlation and fuel design; but not to the restriction on computer code.

DNB is a phenomenon that is characterized by a rapid significant drop in heat transfer, from fuel rod to coolant, due to the formation of an insulating steam blanket along the fuel rod's surface. In accident analyses, a fuel rod is conservatively assumed to fail when it experiences DNB. A fuel rod is considered to be in DNB if its DNB ratio (DNBR) value is less than or equal to the predetermined DNBR specified acceptable fuel design limit (SAFDL). DNBR is defined as the ratio of the heat flux required to attain DNB to the actual local heat flux. The DNBR SAFDL is defined such that there is a 95-percent probability with a 95-percent confidence level that the fuel rod will not experience DNB whenever its DNBR value is greater than the DNBR SAFDL.

2.0 EVALUATION

Regulatory Evaluation

Certain accidents which cause, for example, a decrease in reactor coolant flow while the plant is at power, could result in a degradation of core heat transfer, and a reduction in core thermal margin. If the applicable SAFDLs are exceeded, then DNB could occur and fuel rods could be damaged. Automatically actuated reactor protection and safety systems are designed to mitigate such accidents.

The NRC staff review considers (1) the postulated initial core and reactor conditions; (2) the methods of thermal and hydraulic analyses; (3) the sequence of events; (4) assumed reactions of reactor system components; (5) the functional and operational characteristics of the reactor protection system; (6) operator actions; and (7) the results of the transient analyses.

The NRC acceptance criteria are based on General Design Criterion (GDC) 10 “Reactor design,” of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including anticipated operational occurrences (AOOs); GDC 15, “Reactor coolant system design,” insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation; and GDC 26, “”Reactivity control system redundancy and capability,” insofar as it requires that a reactivity control system be provided and be capable of reliably controlling the rate of reactivity changes to ensure that, under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Technical Evaluation

SCE’s statistical convolution methodology (Reference 2), estimates the number of failed fuel rods by applying the probabilistic definition of DNBR SAFDL (i.e., the SAFDL represents DNBR value, above which there is a 95-percent probability, with a 95-percent confidence level, that the fuel rod will not experience DNB). The probability of DNB decreases as DNBR increases (e.g., above the DNBR SAFDL), and increases as DNBR decreases (e.g., below the DNBR SAFDL).

The DNB statistical convolution groups fuel rods with respect to their radial peaking factors; calculates the minimum DNBR in each radial peaking factor group, and determines the probability of DNB as a function of DNBR value. For any given value of DNBR, the number of fuel rods, within a radial peaking factor group, that are predicted to experience DNB and fail, is the product of the number of fuel rods in the radial peaking factor group and the probability of DNB. The total number of predicted fuel rod failures is the summation of each group’s failed fuel rods. The DNB statistical convolution methodology relies upon a probability distribution function (i.e., the probability of DNB as a function of DNBR).

CENPD-183-A (Reference 3) contains probability distribution functions for the Combustion Engineering (CE) 14x14 and 16x16 rod assemblies. These probability distribution functions were established using the TORC computer code and the CE-1 heat flux correlation.

In response to a request for additional information (RAI) from the NRC staff, SCE has provided the probability distribution function that is used for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. This probability distribution function is also based upon the TORC computer code and the CE-1 heat flux correlation. According to this probability distribution function, the DNBR SAFDL that corresponds to the 95-percent probability of no occurrence of DNB, with a 95-percent confidence level, is 1.31.

The SONGS Units 2 and 3 analyses of accidents and transients that could degrade DNB margin are evaluated with the CETOP-D code (Reference 4). CETOP-D (also known as CETOP), is based on the TORC code, and used as a design code for thermal margin analyses. CETOP-D is a simplified, faster-running version of TORC. Since CETOP-D will not generally produce the same results as TORC, CETOP-D is benchmarked against TORC DNBR data to ensure that CETOP-D DNBR results are always accurate or conservative relative to TORC DNBR results (i.e., CETOP-D DNBR values are always less than or equal to corresponding TORC DNBR values). Consequently, application of the statistical convolution methodology would be expected

to yield a higher estimate of the number of failed fuel rods using the CETOP-D minimum DNBR than when using the TORC minimum DNBR, assuming that both convolutions are based upon the same probability distribution function.

Since (1) CETOP-D is in the SCE licensing basis for thermal margin analyses; (2) CETOP-D is benchmarked against TORC DNBR data to ensure that CETOP-D DNBR results are always accurate or conservative relative to TORC DNBR results; and (3) estimates of the number of failed fuel rods are based upon the same TORC/CE-1 probability distribution function, the NRC staff agrees that removal of the computer code restriction on the use of the DNB statistical convolution methodology is justified.

Conclusion

For the reasons stated above, the NRC staff hereby removes the computer code restriction on the use of the DNB statistical convolution methodology. The affected statement, in the Safety Evaluation, is revised to read, “However, the use of any combination critical heat flux correlation and fuel design, other than that explicitly approved by CENPD-183-A, will require submittal of revised probability distributions for NRC staff review and approval."

9.0 REFERENCES

1. NRC letter to Richard M. Rosenblum, San Onofre Nuclear Generating Station, "SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS RE: FULL-SCOPE IMPLEMENTATION OF AN ALTERNATIVE SOURCE TERM ”, dated December 29, 2006 (ADAMS Accession No. ML0634003590) (TAC NOS. MC5495 AND MC5496)

2. CEN-356(V)-P-A, Revision 01-P-A, "Modified Statistical Combination of Uncertainties", May 1988 (Westinghouse/CE Proprietary information)

3. CENPD-183-A, "Loss of Flow - CE Methods for Loss of Flow Analysis," June 1984 (Westinghouse/CE Proprietary information)

4. “CETOP-D Code Structure and Modeling Methods for San Onofre Nuclear Generating Station Units 2 and 3, CEN-160(S)-P", May 1981

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Date: September 16, 2008

for DNBR calculations, the applicant is required to submit a fuel damage probability distribution for staff's approval.

In summary, the LOF analysis procedure using the static method of hot channel DNBR calculation is acceptable.

The LAR did not request the use of alternate computer codes or critical heat flux correlations, nor has the licensee provided sufficient information to warrant the approval to use alternate computer codes or critical heat flux correlations. A review of the licensee's UFSAR reveals that the CENPD-183-A listed computer codes are part of the licensee's current licensing basis and that TORC/CE-1 is being used for the DNB analysis for the single RCP sheared shaft transient.

The SER approving CENPD-183-A did not limit it to the sheared shaft event analysis. CENPD-183-A was intended for use with loss of forced flow (LOF) transients at CE plants. Further, the abstract for CENPD-183-A does not limit it to any particular LOF transient. In demonstrating the methodology, CENPD-183-A provided sample analysis for a four pump LOF coast down and a seized shaft LOF. Additionally, while the concept of a DNB statistical convolution methodology is approved in CENPD-183-A, the probability distributions provided therein are computer code, critical heat flux correlation, and CE fuel design 14x14 and 16x16 specific. Therefore, based on the review stated above, the NRC staff is approving the licensee's request to expand the use of fuel failure estimates by DNB statistical convolution methodology to all USFAR Chapter 15 non-LOCA events that assume a loss of flow. However, the use of any combination of critical heat flux correlation and fuel design, other than that explicitly approved by CENPD-183-A, will require submittal of revised probability distributions for NRC staff review and approval.

4.0 REGULATORY COMMITMENTS

In its letter dated December 27, 2004, the licensee has made the following regulatory commitments:

1. Following approval of this license amendment request, future revisions to UFSAR Chapter 15 design basis accident control room and offsite radiological consequence analyses will be performed using AST methodology.

2. Following approval of this license amendment request, the manual dose calculation methodology as described in Emergency Planning Implementation Procedures (EPIPs) and other Emergency Planning guidance documents will be revised to reflect AST methodology.

3. Raddose V dose assessment software will be evaluated by June 30, 2005, to determine what specific changes may be warranted in order to maintain consistency with the manual dose assessment calculation methodology.

The licensee, in an e-mail dated December 27, 2006 (to be added to ADAMS), stated that it completed an evaluation of Raddose V assessment software in April, 2006, to determine the changes necessary to implement the AST PCN. That evaluation identified 6 changes, listed below, needed prior to implementing the AST LAR.

radiological consequence analysis, this rate is reduced to 0.05 percent per day after 24 hours following a LOCA for the remaining duration of the accident (30 days), consistent with the guidance provided in RG 1.183. The licensee has not proposed to change the design basis containment leak rate.

3.2.1.1.1 Radioactivity Removal Inside the Containment

The fission products in the containment atmosphere following the postulated LOCA at SONGS 2 and 3 are mitigated by (1) natural deposition of fission products in aerosol form, and (2) removal by the containment spray system (CSS). SCE's analysis assumed removal of fission products in aerosol form by natural deposition in the containment following the postulated LOCA using Powers simplified natural deposition model in the RADTRAD dose consequences computer code described in NUREG/CR-6604 and its supplements. The Powers simplified natural deposition model is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," The licensee used the 10 percentile confidence interval (90 percent probability) removal values implemented in the RADTRAD code. The Powers natural deposition model was derived by correlation of results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The NRC staff finds that the use of this model in NRC computer code, RADTRAD, is acceptable, as discussed in RG 1.183. Aerosol removal rates by natural deposition developed and used by the licensee are shown in Table 4.5-3 of the licensee's March 10, 2006, submittal.

The licensee's analysis also assumed removal of elemental iodine by wall deposition using the methodology provided in SRP Section 6.5.2. Inputs to this methodology include a mass transfer coefficient, the wetted surface area inside containment, and the containment building net-free volume. The licensee used the mass transfer coefficient value of 4.9 meters per hour recommended by SRP Section 6.5.2. The NRC staff finds that the use of this methodology according to the guidance in SRP Section 6.5.2 is acceptable. The elemental iodine deposition removal rate value calculated and used by the licensee is 4.26 per hour.

The CSS at SONGS 2 and 3 is an ESF system. When used in conjunction with two containment emergency cooling units (ECUs), each rated at 31,000 cfm, and one dome air circulator unit (DACU), rated at 37,000 cfm (conservatively modeled as 0 cfm), the CSS is designed to ensure that containment pressure does not exceed the design-basis value of 60 psig and also to remove fission products in the containment atmosphere following the postulated LOCA. To meet the single failure criterion, only one of the two ECU trains and one of the two DACU trains are assumed to be operational for mixing of air in the containment. The licensee assumes that the ECU and DACU start operation 1 minute after the start of the LOCA. The licensee has determined that 99 percent of the contaminated air in the containment unsprayed region will be replaced with air from the sprayed region within 28 minutes, which equates to approximately four change-outs of the containment unsprayed region prior to the end of the activity releases from the core at conclusion of the early in-vessel phase at 1.8 hours. RG 1.183 assumes that two turnovers of the unsprayed region per hour provide adequate mixing. Therefore, the NRC staff concludes that the licensee has shown that the ECUs and DACUs provide adequate mixing between sprayed and unsprayed regions of the containment atmosphere in accordance with the guidance provided in RG 1.183.

Table 4

Parameters and Assumptions Used in Radiological Consequence Calculations for LOCA

Parameter	Value
Reactor power, MWt	3,507
Containment volume, ft^3	
Total	2,284,000
Sprayed area	1,907,000
Unsprayed area	377,000
Containment leak rates, % per day	
0 to 24 hour	0.1
24 to 720 hours	0.05
Containment mechanical mixing rate, cfm	
Sprayed to unsprayed	62,000
Unsprayed to sprayed	62,000
Containment iodine and aerosol removal	Variable
Spray	Table 4a
Aerosol natural deposition	Powers 10th percentile
Elemental iodine deposition, per hr	4.26
ESF recirculation volume, ft^3	46,647
ESF leak rates, cfm	
0 to 20 minutes	0
20 minutes to 30 days	0.007
RWST volume, ft^3	
Air space	35,880
Liquid space	7,345
RWST flow rates, cfm	
Mini-flow into RWST, total after 20 min	0.4010
Discharge check valve into RWST	
0 - 1.08 hr	0
1.08 - 2 hr	1.2859
2 - 8 hr	1.2778
8 - 24 hr	0.9622
24 - 96 hr	0.5103
96 - 119.72 hr	0.1078
119.73 - 720 hr	0

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