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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 47 TO NPF-10

AND AMENDMENT NO. 36 TO NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

Southern California Edison Company (SCE), on behalf of itself and the other licensees, San Diego Gas and Electric Company, The city of Riverside, California, and The City of Anaheim, California, has submitted several applications for license amendments for San Onofre Nuclear Generating Station, Units 2 and 3. Five such requests, designated PCN-201, PCN-202, PCN-203, PCN-204, and PCN-206 are evaluated herein.

By letter dated September 5, 1985, SCE submitted a request to reload and operate Unit 2 of the San Onofre Nuclear Generating Station (SONGS) for Cycle 3 (Ref. 1). In support of the request, the licensee submitted a reload safety analysis report (Ref. 2). By letter dated August 30, 1985 (Ref. 3), the licensee also submitted a report on Core Protection Calculator (CPC) and Control Element Assembly Calculator (CEAC) software modifications (Ref. 4).

Although the reload analysis report was prepared specifically for SONGS Unit 2, SCE and Combustion Engineering (CE) have stated that all pertinent technical specification changes and CPC software modifications are also applicable to SONGS Unit 3 Cycle 3. The licensee has also stated that the reload analysis report for SONGS Unit 3 Cycle 3 will be confirmatory relative to the reload analysis report for SONGS Unit 2 Cycle 3. Therefore, the technical specification changes and CPC software modifications were written accordingly to apply for SONGS Unit 3 Cycle 3 as well.

The NRC staff has reviewed the application and the supporting documents and has prepared the following evaluation of the fuel design, nuclear design, and thermal-hydraulic design of the core as well as an evaluation of those plant transients and accidents which were reanalyzed for Cycle 3.

2.0 FUEL DESIGN

2.1 Mechanical Design

The Cycle 3 core consists of 217 fuel assemblies. Eighty-eight fresh (unirradiated) Batch E assemblies will replace 80 Batch B assemblies and eight Batch C assemblies. The remaining 56 Batch C assemblies and all Batch D assemblies in the core during Cycle 2 will be retained. One Batch A assembly in place during Cycle 2 will be replaced with one Batch A assembly which was discharged after Cycle 1. In addition, a reload batch E consisting of 88 zone-enriched assemblies of four different types will be inserted. These will consist of 40 type E0 assemblies with 4.05 weight percent (w/o) and 3.40 w/o U-235 enriched fuel rods, eight type E1 assemblies with 4.05 w/o and 3.40 w/o U-235 enriched rods and four burnable poison shims per assembly, 28 type E2 assemblies with 3.40 w/o and 2.78 w/o U-235 enriched rods and eight burnable poison shims per assembly, and 12 type E3 assemblies with 3.40 w/o and 2.78 w/o U-235 enriched rods and 16 burnable poison shims per assembly.

The CEA guide tube wear sleeve modification made to the Batch E reload fuel has been reviewed and approved by the NRC (Ref. 5). In addition, SCE has evaluated the criticality effects of storage of the higher enriched Batch E fuel assemblies in the SONGS-2 fuel storage facilities and has shown that the acceptance criterion of K_{eff} less than or equal to 0.95 is met for all normal and abnormal conditions (Ref. 6, 7). We, therefore, conclude that the Batch E fuel assemblies are acceptable for use during Cycle 3.

The licensee has stated that the cladding creep collapse time for any fuel that will be irradiated during Cycle 3 was conservatively determined to be greater than its maximum projected residence time. The creep collapse analysis was performed by CE using the CEPAN computer code (Ref. 8) which has been approved by the NRC for licensing applications. We conclude that the cladding collapse has been appropriately considered and will not occur for Cycle 3 operation.

During the next (Cycle 2/3) refueling outage, fuel will be inspected to provide verification of adequate shoulder gap on fuel which will be reinserted in Cycle 3. In addition, a report will be provided to the NRC by SCE (Ref. 9) to demonstrate that the Cycle 3 fuel will have sufficient available shoulder gap clearance for its total planned exposure.

2.2 Thermal Design

The thermal performance of Cycle 3 fuel was performed by analyzing a composite fuel pin that envelopes the peak pins of the various fuel assemblies (fuel Batches A, C, D and E) in the Cycle 3 core using the NRC approved fuel performance code FATES3A. The NRC imposed grain size restriction (Ref. 10) was included and a power history that envelopes the power and burnup levels representative of the peak pin at each burnup interval from beginning-of-cycle (BOC) to end-of-cycle (EOC) was used. The maximum peak pin burnup analyzed for Cycle 3 bounds the expected EOC maximum fuel rod burnup. License Condition 6 in the SONGS 2 operating license requires that SCE provide analyses using fission gas release models acceptable to the NRC prior to the cycle of operation that will result in peak burnups greater than 20,000 MWD/MTU. Therefore, we find the above presented analysis meets the requirements of this license condition and demonstrates acceptable operating conditions for the design lifetime of the fuel. Based on this analysis, the internal pressure in the most limiting hot rod will not reach the nominal RCS pressure of 2250 psia. Since this satisfies the fuel rod internal gas pressure requirement of Standard Review Plan (SRP) 4.2, Section II.S.1(f), we find it acceptable and conclude that the fuel rod internal pressure limits have been adequately considered for Cycle 3 operation.

3.0 NUCLEAR DESIGN

3.1 Fuel Management

The SONGS Unit 2 Cycle 3 core consists of 217 fuel assemblies, each having a 16 by 16 fuel rod array. A general description of the core loading is given in Section 2.1 of this SER. The highest U-235 enrichment occurs in the Batch E fuel assemblies which contain fuel rods with 4.05 weight percent (w/o) U-235. The SONGS Units 2 and 3 fuel storage facilities have been approved for storage of fuel of maximum U-235 enrichment of 4.1 w/o (Ref. 7).

The Cycle 3 core will minimize power peaking by loading approximately half of the fresh fuel assemblies (Batch E) on the core periphery and shuffling the Cycle 2 peripheral assemblies to the interior of the core. Forty of the lower enriched Batch E assemblies will be mixed with the previously irradiated fuel in the central region of the core. With this loading and a Cycle 2 endpoint of 10,000 MWD/MTU, the Cycle 3 reactivity lifetime for full power operation is expected to be 14,500 MWD/MTU. The analyses presented by the licensee will accommodate a Cycle 3 length up to 16,000 MWD/MTU and is applicable for Cycle 2 termination burnups of between 9,800 and 10,200 MWD/MTU.

3.2 Power Distributions

Hot full power (HFP) fuel assembly relative power densities are given in Reference 2 for beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) unrodded configurations. Radial power distributions at BOC and EOC are also given for rodded configurations allowed by the power dependent insertion limit (PDIL) at full power. These rodded configurations consist of part length CEAs (PLCEAs), Bank 6, and Bank 6 plus the PLCEAs. The largest radial power peak occurs at BOC for both the rodded and unrodded configurations. These expected values are based on ROCS code calculations with neutron cross sections generated by the DIT code (Ref. 11). Also, the use of ROCS and DIT with the MC fine-mesh module explicitly accounts for the higher power peaking which is characteristic of fuel rods adjacent to water holes. These methods have been approved by the NRC and, therefore, the calculated power distributions are acceptable.

3.3 Control Requirements

The value of the required shutdown margin varies throughout core life with the most restrictive value occurring at EOC hot zero power (HZP) conditions. This minimum shutdown margin of 5.15% delta k/k is required to control the reactivity transient resulting from the reactor coolant system (RCS) cooldown associated with a steam line break accident at these conditions. For operating temperatures below 200°F, the reactivity transients resulting from inadvertent boron dilution events have established a 3.0% delta k/k shutdown margin requirement. Sufficient boration capability and net available CEA worth, including a maximum worth stuck CEA and appropriate calculational uncertainties, exist to meet these shutdown margin requirements. These results were derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable.

3.4 Augmentation Factors

A report entitled "Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods" (Ref. 12) was submitted to the staff during the review of the Calvert Cliffs Unit 1 Cycle 8 license application. The report presented an analysis performed by CE for Electric Power Research Institute (EPRI) and gave the results of a review of interpellet gap formation, ovality, creepdown and clad collapse data in modern PWR fuel rods (non-densifying fuel in pre-pressurized tubes). The report concluded that since the increased power peaking associated with the small interpellet gaps found in these rods is insignificant compared to other power distribution uncertainties used in the safety analyses, augmentation factors can be removed from the reload of any reactor loaded exclusively with this type of fuel. The staff accepted this conclusion for the Cycle 8 reload review of Calvert Cliffs Unit 1 and agrees that the conclusion is also valid for SONGS Unit 2 Cycle 3 since the same manufacturing process is used in the Calvert Cliffs and the SONGS fuel. The densification augmentation factors can, therefore, be eliminated for SONGS Unit 2 Cycle 3.

4.0 THERMAL-HYDRAULIC DESIGN

Steady-state thermal-hydraulic analysis for Cycle 3 is performed using the approved thermal-hydraulic code TORC (Ref. 13) and the CE-1 critical heat flux (CHF) correlation (Ref. 14). The core and hot channel are modeled with the approved method described in Ref. 15. The design thermal margin analysis is performed with the fast running variation of the TORC code, CETOP-D (Ref. 16). The licensee has shown that the CETOP-D model predicts minimum departure from nucleate boiling ratio (DNBR) conservatively relative to TORC (Ref. 16).

The uncertainties associated with the system parameters are combined statistically using the approved statistical combination of uncertainties (SCU) methodology described in Refs. 17, 18, and 19. Using this SCU methodology, the engineering hot channel factors for heat flux, heat input, fuel rod pitch, and cladding diameter are combined statistically with other uncertainty factors to arrive at an equivalent DNBR limit of 1.31 at a 95/95 probability/confidence level. The fuel rod bow penalty is incorporated directly in the DNBR limit. It has been calculated using the approved method described in Ref. 20. The value used for this analysis, 1.75% DNBR, is valid for bundle burnups up to 30,000 MWD/MTU. For those assemblies with average burnup in excess of 30,000 MWD/MTU, sufficient margin exists to offset rod bow penalties.

5.0 SAFETY ANALYSES

The design basis events (DBEs) considered in the safety analyses are categorized into two groups: anticipated operational occurrences (AOOs) and postulated accidents. All events were reviewed by the licensee to assess the need for reanalysis as a result of the new core configuration for Cycle 3. Those events for which results were not bounded by the FSAR were reanalyzed by the licensee to assure that the applicable criteria are met. The AOOs were analyzed to assure that specified acceptable fuel design limits (SAFDLs) on DNBR and fuel centerline to melt (CTM) are not exceeded. This may require reactor protection system (RPS) trips and/or sufficient initial steady state margin to prevent exceeding the SAFDLs.

Unless otherwise stated, the plant response to the DBEs was simulated using the same methods and computer programs which were used and approved for the reference cycle analyses. These include the CESEC III, STRIKIN II, TORC and HERMITE computer programs. For some of the reanalyzed DBEs, certain initial core parameters such as CEA trip worth and moderator temperature coefficient (MTC) were assumed to be more limiting than the actual calculated Cycle 3 values in order to bound future cycles. All of the events reanalyzed have results which are within NRC acceptance criteria and, therefore, are acceptable.

5.1 Increased Main Steam Flow

The increased main steam flow event was reanalyzed due to the increased Doppler coefficient multiplier, the availability of the variable overpower trip (VOPT), and a more adverse fuel pin power peaking census for Cycle 3 compared to the reference cycle. For the increased main steam flow event without a single failure, the DNBR and CTM limits are not exceeded. For transients coupled with a concurrent single failure, the most limiting event with respect to DNBR is the increase in main steam flow with loss of AC power. This event resulted in a coincident CPC VOPT/low flow trip and a minimum DNBR of 1.21 compared to the design limit of 1.31. For this event, we consider any pin which has a DNBR below 1.31 to have failed. Based on our criterion, the licensee has determined that less than 1.5% of the fuel pins would fail. For a conservative estimate of the radiological consequences, a failed fuel percentage of 5% was used, resulting in offsite doses of approximately 7 rem thyroid and a whole body dose of less than 2 rem (Ref. 21). These are well within 10 CFR 100 values and are, therefore, acceptable. A maximum allowable linear heat rate (LHR) of 16.0 kW/ft could exist before the transient begins without causing the CTM of 21.0 kW/ft to be exceeded. This amount of margin is assured by setting the LHR limiting condition for operation (LCO) based on the more limiting loss of coolant accident (LOCA) limit of 13.9 kW.ft. We, therefore, find the results of the licensee's analysis to be acceptable.

5.2 Steam System Piping Failures Inside and Outside of Containment

Steam line breaks (SLBs) inside containment may have break areas up to the cross section of the largest main steam pipe (7.41 ft^2). In the reference cycle, the licensee performed a parametric analysis in both MTC and break area to determine the limiting inside containment SLB event in terms of fuel pin failure caused by the pre-trip power excursion. The inside containment SLB event was reanalyzed for Cycle 3 using the more restrictive Cycle 2 key parameters used in the parametric analysis combined with the more adverse Cycle 3 fuel pin census. Since inside containment SLBs may cause environmental degradation of sensor input to the CPCs and pressure measurement systems, the only credit taken for CPC action during this event is the CPC VOPT. The required input to the VOPT includes output from the resistance temperature detectors (RTDs) and the excore neutron flux detectors. These sensors have been qualified in degraded environmental conditions for a sufficient length of time to allow their use in providing input for VOPT action for this event (Ref. 22). Because of this credit for CPC VOPT action, the amount of calculated fuel failure for the Cycle 3 inside containment SLB is bounded by the reference cycle analysis. Break areas for outside containment SLBs are limited to the area of the flow restrictors (4.13 ft^2) located upstream of the containment penetrations. The outside containment SLBs, however, are not subject to the same environmental effects on the RPS as the inside containment breaks and the full array of RPS trips, including the CPC low DNBR trip, can be credited. Although the reference cycle identified the limiting break location to be inside containment, the reanalysis for Cycle 3 indicated that the limiting break in terms of radiological consequences is located outside of the containment building

(Ref. 21). This is because crediting the action of the CPC VOPT reduced the amount of calculated fuel failure for the inside containment SLB relative to the reference cycle. The results of outside containment SLB reanalyses indicate that a coolable geometry is maintained during the event since the number of calculated fuel pin failures is less then 1.5%. In addition, the site boundary doses are a small fraction of 10 CFR 100 limits for a coincident iodine spike. For a pre-existing iodine spike or for the predicted fuel failure of less than 1.5%, the resultant doses are within the 10 CFR 100 limits.

The licensee has demonstrated conformance with the acceptance criteria stipulated SRP Section 15.1.5. As such, we conclude that Cycle 3 operation is acceptable with respect to accidents resulting in breaks in the steam line.

5.3 Feedwater System Pipe Break Event

The feedwater system pipe break event with a loss of AC power at time of trip was analyzed to demonstrate that the assumed increase in the number of plugged steam generator tubes in Cycle 3 will not cause violation of the RCS pressure criterion. The initial RCS pressure and initial steam generator inventory were selected such that the low steam generator water level trip and the high pressurizer pressure trip occur simultaneously, resulting in the maximum peak RCS pressure after trip. The RCS pressure increases to 2943 psia compared to the reference cycle value of 2930 psia. Since the staff considers a feedwater line break with a concurrent loss of non-emergency AC power to be a very low probability event, SRP 15.2.8 requires that the RCS pressure should be maintained below 120% of the design pressure (3000 psia). This criterion is met and the feedwater line break, which is the limiting event with respect to RCS pressure, results in acceptable consequences during Cycle 3.

5.4 Total Loss of Forced Reactor Coolant Flow

The loss of coolant flow (LOF) event was reanalyzed by the licensee due to the reduction in CEA worth at trip. As for Cycle 2, the LOF event for Cycle 3 was analyzed with a CPC trip based on low reactor coolant pump (RCP) shaft speed, initiated when the shaft speed drops to 95% of its initial speed. For conservatism, the analysis actually assumes that the trip is initiated when the reactor coolant flow reaches 95% of its initial value since the reduction in core flow lags the decrease in RCP shaft speed. The results show that this event initiated from the Technical Specification LCOs in conjunction with the low RCP shaft speed trip will not exceed the DNBR limit and is, therefore, acceptable.

5.5 Single Reactor Coolant Pump Sheared Shaft

The single reactor coolant pump sheared shaft was reanalyzed due to a change in the fuel failure pin census. Reactor trip was assumed to occur when the rapid flow reduction across the steam generator in the affected loop decreases the delta-pressure below the trip setpoint. The minimum DNBR was evaluated at the asymptotic flow of 75% of initial flow with no credit for heat flux decay on reactor trip. The analysis used core parameters which bound Cycle 3 values.

The amount of fuel failure calculated for Cycle 3 was 4.1% (Ref 23). However, in order to bound possible future conditions, the radiological consequences were calculated based on 9.0% fuel failure. The resultant doses were 14 REM thyroid and 1.0 REM whole body which are a small fraction of 10 CFR 100 guidelines and, therefore, acceptable.

A turbine trip and coincident loss of offsite power and coastdown of undamaged pumps was not considered in the analysis since it was not assumed in the FSAR analysis which provides the licensing basis for this event. However, the steam releases used to determine the doses assumed that the steam bypass was unavailable and these steam releases bound those for a loss of offsite power. The assumption of a loss of offsite power would not impact the amount of predicted fuel failure since CE has previously demonstrated that a minimum of 3 seconds exists from the time of turbine trip to the time of loss of offsite power. This delay places the time of the coast-down of the remaining pumps well past the time of minimum DNBR for the Cycle 3 analysis.

5.6 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition

The uncontrolled CEA withdrawal event from a subcritical or low power condition was reanalyzed due to an increase in the maximum reactivity insertion rate, a decrease in the Doppler coefficient multiplier, a change in minimum CEA trip worth, and, for the event initiated from low power, the addition of the CPC VOPT. The events are analyzed to ensure that the DNBR and the CTM SAFDLs are not violated and to verify that the peak RCS pressure is less than the design limit of 2750 psia.

The CEA withdrawal from subcritical conditions resulted in a reactor trip on high logarithmic power with a minimum DNBR greater than the design limit of 1.31. The peak linear heat generation rate (PLHGR) was predicted to be 26 kW/ft which is in excess of the steady state centerline melt limit of 21 kW/ft. Since this transient value of PLHGR exceeded the steady state limit, an assessment of the resultant fuel centerline temperature was performed by the licensee based on the maximum centerline enthalpy of the fuel. The calculation assumed that no heat is transferred away from the centerline during the transient (i.e., adiabatic conditions). The total enthalpy was calculated to be 75.8 cal/gm. The temperature corresponding to this enthalpy is less than 2000°F, which is well below the UO_2 melting point of 4900°F. Additionally, the peak RCS pressure is less than the design limit of 2750 psia.

For the CEA withdrawal from low power, a parametric analysis on reactivity addition rate was performed to yield a coincident VOPT/high pressurizer pressure trip in order to maximize the peak RCS pressure. The results indicate that the DNBR, CTM, and RCS pressure limits will not be exceeded during the event.

We, therefore, conclude that Cycle 3 meets the requirements of SRP Section 15.4.1 and 15.4.2 governing CEA withdrawal events.

5.7 Inadvertent Boron Dilution

This event was reanalyzed due to the Cycle 3 increase in critical boron concentrations. For power operation (Modes 1 and 2), an inadvertent boron dilution event will be terminated by the CPC trip system. For subcritical modes (Modes 3 through 6), the time required to achieve criticality due to boron dilution depends on the initial and critical boron concentrations as well as the inverse boron worth and the rate of dilution. The analysis for Mode 5 (cold shutdown) with the RCS partially drained assumes that only one charging pump is operable. The results show that, with the alarms which were installed before Cycle 1 startup, sufficient time exists to alert the operator of a boron dilution event at least 15 minutes before criticality (30 minutes during refueling) during all modes of Cycle 3 operation. We conclude that SONGS Unit 2 Cycle 3 meets the requirements of SRP Section 15.4.6 and is acceptable with respect to inadvertent boron dilution events.

5.8 Asymmetric Steam Generator Events

The four events which affect a single steam generator are:

- (A) loss of load to one steam generator (LL/1SG)
- (B) excess load to one steam generator (EL/1SG)
- (C) loss of feedwater to one steam generator (LF/1SG)
- (D) excess feedwater to one steam generator (EF/1SG)

Of these, the LL/1SG event is the limiting asymmetric event. This event is initiated by the inadvertent closure of a single main steam isolation valve (MSIV), which results in a loss of load to the affected steam generator. The CPC high differential cold leg temperature trip serves as the primary means of mitigating this transient with the steam generator low level trip providing additional protection. The minimum transient DNBR calculated was greater than the DNBR SAFDL limit of 1.31. A maximum allowable LHR of

17.0 kW/ft could exist as an initial condition without exceeding the fuel CTM SAFDL of 21.0 kW/ft during the transient. This amount of margin is assured by setting the LHR LCO based on the more limiting allowable LHR for LOCA of 13.9 kW/ft. The staff concludes that the calculations contain sufficient conservatism to assure that fuel damage will not result from any asymmetric steam generator event during Cycle 3 operation.

A methodology change from the reference cycle analysis of this event is the application of the HERMITE computer code to model both the effects of the temperature tilt on radial power distribution and the space-time impact of the CEA scram. HERMITE has been approved for licensing applications (Ref. 24) and uses the core parameters generated by the CESEC code (core flow, RCS inlet temperature, RCS pressure, and reactor trip time) as input to simulate the core in two dimensions. We find this improved modeling technique acceptable.

5.9 Loss of Coolant Accident (LOCA)

The ECCS performance evaluation for both the large break and the small break LOCA must show conformance with the acceptance criteria required by 10 CFR 50.46. A SONGS specific analysis was performed for Cycle 3, primarily to account for an increase in the assumed number of steam generator tubes plugged from 100 to 1000 per steam generator. Also, the minimum containment pressure assumed was lowered from 14.4 psia to 13.7 psia in order to provide operational flexibility. Since a comparison of the two limiting LOCA events for Cycle 1 had previously demonstrated that the large break LOCA ECCS performance was more limiting than the small break LOCA performance results, only the large break LOCA was reanalyzed for Cycle 3. The analysis was performed for Cycle 3 using approved computer programs and models which meet the requirements of Appendix K to 10 CFR 50.

The 1.0 double-ended guillotine at pump discharge (DEG/PD) break results in the highest peak clad temperature (2116°F) and the highest core wide clad oxidation percentage (0.68%). Previous analyses have shown that the local clad oxidation percentage for both the 1.0 and the 0.8 DEG/PD break were essentially equivalent. For the 1.0 DEG/PD break the peak local oxidation was calculated to be 10.08%. Since the results meet the acceptance criteria for peak clad temperature (2200°F), peak local clad oxidation percentage (17.0%), and core wide clad oxidation percentage (1.0%), we conclude that operation of SONGS 2 with a peak linear heat generation rate (PLHGR) of 13.9 kW/ft is acceptable for Cycle 3.

6.0 CPC/CEAC SOFTWARE MODIFICATIONS

The SONGS Units 2 and 3 Core Protection Calculator/CEA Calculator system is provided by the reactor vendor Combustion Engineering. The system is designed to provide the necessary reactor trips (low DNBR and high local power density) to ensure that the specified acceptable fuel design limits (SAFDLs) on DNB and centerline fuel melting are not exceeded during AOOs. The CPC system is also designed to aid in limiting the consequences of certain postulated accidents.

The CPC/CEAC software for Cycle 3 operation is an updated version of the CE CPC/CEAC software which has been previously approved for use in CESSAR 80 plants. By letters dated August 30, 1985 (Ref. 3) and October 18, 1985 (Ref. 25), the licensee submitted CEN-308-P, "CPC/CEAC Software Modifications for the CPC Improvement Program" (Ref. 4) and CEN-310-P, "CPC and Methodology Changes for the CPC Improvement Program" (Ref. 26), which describe additional CPC/CEAC software modifications to be applied to the SONGS Cycle 3 operation. These modifications will also apply to Arkansas Nuclear One Unit 2 (ANO-2), Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, and Waterford Unit 3 and are intended to be implemented at each plant at the appropriate time. These modifications have been reviewed and approved (Ref. 27) and are summarized in Table 6-1 below.

As a result of these CPC software modifications, changes have been made to some addressable constants. The power synthesis algorithm changes in the POWER program allow the addressable constants ARM6, ARM7, EOL, ASM6 and ASM7 to be deleted. The combination of the penalty factor multipliers for DNBR and LPD into a single multiplier results in the deletion of addressable constant PFMLTL. Also, as a result of the simplification of the flow calculations, the core coolant mass flow rate calibration constant FC2 will be deleted. These addressable locations will now contain the following new addressable constants:

1. ARM6 will contain the maximum value of Variable Over Power Trip (VOPT) setpoint.
2. ARM7 will contain the offset between VOPT setpoint and Follow.
3. EOL will contain the DNBR trip setpoint.
4. ASM6 will contain the ASGT WT trip setpoint.
5. ASM7 will remain vacant.
6. PFMLTL will contain the CEAC penalty factor time delay as a result of the CEAC desensitization changes.
7. FC2 will contain the pump speed trip setpoint.

Table 6-1
CPC System Software Algorithm Changes for Cycle 3

A. FLOW Program

1. Simplification of flow calculations.*
2. Removal of the DNBR flow projection modules.

B. UPDATE Program

1. Addition of variable overpower trip.*
2. Removal of redundant thermal power compensation filters.
3. Enhancement of ASGT delta-T compensation filter.*
4. Changes for CEAC desensitization.*
5. Removal of pressure projection.
6. Combination of PFMLTD and PFMLTL into a single penalty factor multiplier.*

C. POWER Program

1. Base low power ASI calculation on actual axial shape.
2. Revise power synthesis calculations.*
3. Removal of flow projection calculations and DNBR operating limit.
4. Incorporation of an ASI dependent power peaking adjustment.
5. Changes for CEAC desensitization - CEA Withdrawal Prohibit (CWP) flag for misoperation.

D. TRIPSEQ Program

1. Removal of comparison to flow projected DNBR and pressure projected DNBR.
2. Redefinition of J_{trip}.
3. Changes for CEAC desensitization.
4. Addition of DNBR trip setpoint to addressable constants.*

E. CEAC Program

1. Changes for CEAC desensitization - Set flag to initiate CWP.

*Require additions to or modification of Addressable Constants.

7.0 TECHNICAL SPECIFICATION CHANGES

The staff has reviewed the proposed modifications to the Technical Specifications for Cycle 3 submitted by letter from K. P. Baskin (SCE) to H. R. Denton (NRC) on October 9, 1985 (Ref. 28). The change numbers as given in Reference 28 are given as the heading for each evaluation.

Proposed Change No. PCN-201

The proposed change revises Technical Specifications 3/4.2.4, "DNBR Margin," and 3/4.3.1, "Reactor Protective Instrumentation." The proposed change consists of the following four parts:

1) Figures 3.2-1, 3.2-2, and 3.2-3 are revised and the existing LCO is replaced with four parts, i.e., Sections 3.2.4.a through 3.2.4.d.

2) The rod bow penalty factors on DNBR as a function of fuel burnup are removed from Surveillance Requirement 4.2.4.4.

3) ACTION statement 6 in Table 3.3-1 is revised to combine ACTION 6.b and 6.c and the reference to Figure 3.2-1 is replaced by reference to Specification 3.2.4.b. Also, the reference to the penalty factor on the BERR1 constant is replaced by reference to Specification 3.2.4.d.

4) The requirement of Surveillance Requirement 4.2.4.2 to verify that the appropriate penalty factors have been implemented on all CPC system channels is removed and replaced with a requirement to verify on any CPC system channel.

Our evaluation of these proposed changes is given below.

The first set of changes are largely administrative in that the six existing figures used to maintain an adequate DNBR margin under various operating states are replaced by four administrative control statements and two new figures. The first two new administrative statements, 3.2.4.a and 3.2.4.b, merely replace existing Figures 3.2-1 (one for Cycle 1 and one for Cycle 2) with words to the same effect when COLSS is in service. Also, both new figures, 3.2-1 and 3.2-2, supplant the existing Figures 3.2-2 and 3.2-3 in compliance with 3.2.4.c and 3.2.4.d when COLSS is out of service. The new 3.2.4.c. and 3.2.4.d specifications, however, state that when COLSS is out of service, the CPC calculated DNBR on any operable CPC system channel must be kept within the limits of either Figure 3.2-1 or 3.2-2. The staff requested additional clarification from the licensee as to the acceptability of monitoring any one operable CPC channel for control purposes rather than the most limiting channel. In their response, the licensee stated that each CPC channel is analyzed independently and is guaranteed (by means of the CPC uncertainty analysis) to be always conservative. Therefore, it does not matter which of the four CPC channels is chosen to monitor DNBR when COLSS is out of service. The four redundant CPC channels are still required for protection purposes to ensure appropriate protective action during DBEs. We find this acceptable.

The second change relates to Surveillance Requirement 4.2.4.4 which requires that the rod bow penalty on DNBR as a function of fuel exposure should be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days. The rod bow penalty, as shown in Section 4.0 of this SER, has been found to be 1.75% at a fuel exposure of 30,000 MWD/MTU. Because of the physical burndown effect, a fuel assembly with burnup exceeding 30,000 MWD/MTU would not produce sufficient power to be subject to a limiting DNBR condition. Therefore, 30,000 MWD/MTU can be considered the cutoff point for a rod bow penalty calculation. As a result of the application of statistical combination of uncertainties (SCU) (Ref. 17), the rod bow penalty of 1.75% at 30,000 MWD/MTU has been incorporated in the minimum DNBR limit of the CPC and Surveillance Requirement 4.2.4.4 can be deleted. The rod bow penalty factor will be verified by the licensee for each future cycle by design analysis.

The proposed change to ACTION 6 of Table 3.3-1 addresses operation with COLSS out of service and none, one, or both CEACs operable. Existing ACTION 6.b addresses operation with COLSS out of service and either one or both CEACs operable. Except for whether or not the CEACs are operable, these two ACTION statements are essentially identical and, therefore, the modification which combines them is acceptable. For one or both CEACs operable, reference to the previous Figure 3.2-1 is replaced by reference to the new Specification 3.2.4.b. When neither CEAC is operable, reference to the penalty factor on the CPC addressable constant BERR1 is replaced by reference to the new Specification 3.2.4.d. This is acceptable since the same penalty is applied by SCU as described above.

The remaining modification revises Surveillance Requirement 4.2.4.2 which requires that above 20% of rated thermal power, and with COLSS out of service, DNBR must be verified to be within its allowable limits (as per Figures 3.2-1 and 3.2-2) at least once per 2 hours as indicated on any operable DNBR channel. The existing surveillance requires verification on all operable DNBR channels. As mentioned above, the staff has found that monitoring DNBR when COLSS is out of service on any one of the operable CPC channels is acceptable.

Proposed Change No. PCN-202

The negative limit on moderator temperature coefficient (MTC) has been changed to -3.3×10^{-4} delta k/k/°F at rated thermal power in Technical Specification 3.1.1.3.b.

The revised MTC negative limit is consistent with the value used in the reanalysis of any Cycle 3 transient or accident which involves a decrease in primary coolant temperature. The results of these reanalyses remain within all acceptable criteria as specified in Section 15 of the Standard Review Plan (Ref. 29). These safety analyses and the calculation of MTC have been performed with approved methods. The change is, therefore, acceptable.

<u>Proposed Change No. PCN-203</u>

The proposed changes revise Technical Specification 3/4.3.1, "Reactor Protective Instrumentation," and Technical Specification 3/4.2.4, "DNBR Margin." Specifically, the resistance temperature detector (RTD) maximum response time has been revised from 13 seconds to 8 seconds and Tables 3.3-2a and 3.3-2b, which provided penalty factors to be applied to the CPC and COLSS calculations, have been removed. In addition, the proposed change also revises Note (#) appended to Item 10(e), "Primary Coolant Pump Shaft Speed," by specifying that the response time is measured using simulated pump coastdown rather than from the onset of a two out of four reactor coolant pump (RCP) coastdown.

The RTDs are used to measure the cold and hot leg temperatures which are used in the CPC and COLSS for core power and DNBR calculations. For the previous cycles, the accident analysis assumed an initial response time of 6 seconds but the Technical Specifications allowed continued operation with the RTD time constant possibly degraded beyond 6 seconds by adjusting the CPC and COLSS calculations with penalty factors given in Tables 3.3-2a and 3.3-2b. For Cycle 3, the algorithms for these power and DNBR calculations are being modified to have a built-in RTD response time constant of eight seconds, which is the RTD response time used in the Cycle 3 accident analysis. In addition, the licensee has stated that the actual measured response times at SONGS 2 and 3 have been significantly less than 6 seconds and have shown no evidence of degradation between measurements which are required for each reactor trip function every 18 months (Ref. 23). Therefore, revision of the RTD response time to 8 seconds is acceptable and Tables 3.3-2a and 3.3-2b may be removed.

The CPC counts the pulses generated by the fly-disk on the RCP. The pulse frequency is then used by the CPC to determine RCP speed. The pump coastdown is then simulated by changing the frequency that would represent the pump speed at the end of the transient marking the point at which the CPC generates a trip. Therefore since the response time measurement is done using a simulation rather than from the onset of a two out of four RCP coastdown, the revision to Note (#) of Item 10(e) is appropriate and acceptable.

<u>Proposed Change No. PCN-204</u>

The trip setpoint and allowable value for the local power density (LPD) trip has been changed to 21.0 kW/ft. The change revises Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," of Technical Specification 2.2.1 and its associated bases.

The LPD trip setpoint specifies the setpoint required to prevent the peak linear heat rate, in the limiting fuel pin in the core, from exceeding the value which corresponds to the centerline fuel melting temperature during anticipated operational occurrences. It also assists in mitigating the consequences of accidents. The modification increases the trip setpoint from 19.95 kW/ft to 21.0 kW/ft, which is the linear heat generation rate corresponding to fuel centerline melting as determined by approved methods. Previously, the trip setpoint incorporated an adjustment for dynamic effects that will now be accounted for elsewhere in the CPC algorithms. Also, the 21.0 kW/ft value has been previously approved for the Palo Verde Units 1 and 2 and the CESSAR System 80 Technical Specification LPD trip setpoint. Based on this, and on the fact that, effectively, the CPC LPD protection is not being changed, the modification is acceptable.

Proposed Change PCN-206

The CPC addressable constants have been removed from the Technical Specifications. The requirement for Onsite Review Committe (OSRC) review and approval of the entry of addressable constants outside the allowable range previously specified in the Technical Specifications has also been deleted.

The addressable constants of the CE designed CPCs provide a mechanism to incorporate reload dependent parameters and calibration constants to the CPC software so that the CPC core model is maintained current with changing core configurations and operating characteristics. There are two types of addressable constants. The first type, Type I, are the calibration constants, sensor operability status flag and pretrip alarm set points which are expected to change frequently during cycle operation. These constants are entered into the CPC via the CPC operator module. The second type, Type II, are related to measured physics test parameters, uncertainties, allowances and adjustments. Values are determined or confirmed during startup tests following each fuel loading and are not expected to change during cycle operation. These addressable constants are typically entered into the CPCs from diskettes.

The staff has previously approved the request by the Arizona Nuclear Power Project to delete the CPC addressable constants from the Palo Verde Unit 2 Technical Specifications. In addition, the bases are revised to identify that (1) the potential for inadvertent misloading of addressable constants is minimized by administrative controls, (2) modifications to CPC software will be made in accordance with an NRC approved procedure, and (3) CPC software modifications which involve either an unreviewed safety question, Technical Specification changes, or new methodology previously not reviewed by the NRC will require NRC approval prior to implementation. This includes additions or deletions to the addressable constants or changes to the software limits on the addressable constants. Therefore, the changes to Technical Specifications 2.2.2, 6.5.1.6. 6.8.1, and the Bases for Specifications 3/4.3.1 and 3/4.3.2 are acceptable. In addition, the reference to Specification 2.2.2 in Notation (11) to Table 4.3-1 should also be removed from the Technical Specifications.

8.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics and thermal-hydraulics information presented in the SONGS Units 2 and 3 Cycle 3 reload report. We have also reviewed the Technical Specification revisions, the CPC/CEAC modifications, and the safety reanalyses. Based on our evaluations given in the preceding sections, we find the proposed reload report acceptable.

9.0 CONTACT WITH STATE OFFICIAL

The NRC staff has advised the Chief of the Radiological Health Branch, State Department of Health Services, State of California, of the proposed determination of no significant hazards consideration. No comments were received.

10.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in the installation or use of facility components located within the restricted area. The staff has determined that the amendments involve no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cummulative occupation radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on such findings. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec. 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of these amendments.

11.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable, and are hereby incorporated into the San Onofre 2 and 3 Technical Specifications.

Dated: May 16, 1986

12.0 REFERENCES

1. Letter from M. O. Medford (SCE) to G. W. Knighton (NRC), Reload Analysis Report, San Onofre Nuclear Generating Station Units 2 and 3, September 5, 1985.

2. "San Onofre Nuclear Generating Station Units 2 and 3 Cycle 3 Reload Analysis Report," Combustion Engineering, September 1985.

3. Letter from A. E. Scherer (CE) to G. W. Knighton (NRC), "CPC/CEAC Software Modifications for the CPC Improvement Program," LD-85-044, August 30, 1985.

4. "CPC/CEAC Software Modifications for the CPC Improvement Program," CEN-308-P, Rev. 00-P, Combustion Engineering, August 1985.

5. Letter from C. O. Thomas (NRC) to A. E. Scherer (CE), "Acceptance for Referencing of Licensing Special Report LD-84-043, CEA Guide Tube Wear Sleeve Modification," September 7, 1984.

6. Letter from M. O. Medford (SCE) to G. W. Knighton (NRC), submitting proposed change NPF-10/15-199 revising Technical Specification 5.3.1, "Fuel Assemblies," August 23, 1985.

7. Letter from G. W. Knighton (NRC) to K. P. Baskin (SCE), Issuance of Amendment No. 39 to Facility Operating License NPF-10 and Amendment No. 28 to Facility Operating License NPF-15, San Onofre Nuclear Generating Station, Units 2 and 3, December 2, 1985.

8. "CEPAN Model of Analyzing Creep Collapse of Oval Cladding," CENPD-187-A, Combustion Engineering, March 1976.

9. Letter from K. Baskin (SCE) to G. W. Knighton (NRC), "Shoulder Gap Clearance San Onofre Units 2 and 3," July 25, 1983.

10 Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BG&E), "Safety Evaluation of CEN-161 (FATES3)," March 31, 1983.

11. "The ROCS and DIT Computer Code for Nuclear Design," CENPD-266-P-A, Combustion Engineering, April 1983.

12. "CEPAN Method of Analyzing Creep Collapse of Oval Cladding, Vol. 5: Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," EPRI NP-3966-CEM, Vol. 5, Project 2061-6, Computer Code Manual, April 1985.

13. "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P, Combustion Engineering, July 1975.

14. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," CENPD-162-P-A, Combustion Engineering, April 1975.

15. "TORC Code, Verification and Simplified Modeling Methods," CENPD-206-P, Combustion Engineering, January 1977.

16. "CETOP-D Code Structure and Modeling Methods for San Onofre Nuclear Generating Units 2 and 3," CEN-160(s)-P, Rev. 1-P, Combustion Engineering, September 1981.

17. "Statistical Combination of Uncertainties, Part 1, Combination of System Parameter Uncertainties in Thermal Margin Analyses for SONGS Units 2 and 3," CEN-283(s)-P, Combustion Engineering, June 1984.

18. "Statistical Combination of Uncertainties, Part 2, Uncertainty Analysis of Limiting Safety System Settings for SONGS Units 2 and 3," CEN-283(s)-P, Combustion Engineering, October 1984.

19. "Statistical Combination of Uncertainties, Part 3, Uncertainty Analysis of Limiting Conditions for Operation for SONGS Units 2 and 3," CEN-183(s)-P, Combustion Engineering, October 1984.

20. "Fuel and Poison Rod Bowing," CENPD-225-P-A, Combustion Engineering, June 1983.

21. Letter from M. O. Medford (SCE) to G. W. Knighton (NRC), Response to Initial Round of Questions Pertaining to the Cycle 3 Reload Analysis Report for SONGS Unit 2, December 13, 1985.

22. "Environmental Qualification Report Per Requirements of NUREG-0588, Revision 2," San Onofre Nuclear Generating Station Unit 2 and 3, Unit 2 Docket 50-361, Unit 3 Docket 50-362.

23. Letter from M. O. Medford (SCE) to G. W. Knighton (NRC), Response to Second Round of Questions on the Cycle 3 Reload Analysis Report for SONGS Unit 2, January 21, 1986.

24. "HERMITE Space-Time Kinetics," CENPD-188-A, Combustion Engineering, July 1976.

25. Letter from A. E. Scherer (CE) to G. Knighton (NRC), "CPC and Methodology Changes for the CPC Improvement Program," October 18, 1985.

26. "CPC and Methodology Changes for the CPC Improvement Program," CEN-310-P, Combustion Engineering, October 1985.

27. Letter from H. Berkow (NRC) to A. E. Scherer (CE), Acceptance for Referencing of Licensing Topical Reports CEN-308-P and CEN-310-P, March 1986.

28. Letter from K. P. Baskin (SCE) to H. R. Denton (NRC), Amendment Application No. 32 to Facility Operating License NPF-10 and Amendment Application No. 18 to Facility Operating License NPF-15, October 9, 1985.

29. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, U.S. Nuclear Regulatory Commission.

30. Letter from G. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 43 to Facility Operating License NPF-10 and Amendment No. 32 to Facility Operating License NPF-15 San Onofre Nuclear Generating Station, Units 2 and 3," March 1986.

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ISSUANCE OF AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE NPF-10
AND AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NPF-15
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