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May 21, 1991

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
Reply to a Notice of Violation
San Onofre Nuclear Generating Station, Units 2 and 3

Reference: Letter from Mr. R. P. Zimmerman (USNRC) to
Harold B. Ray (SCE), dated April 12, 1991

The referenced letter forwarded a Notice of Violation (NOV) resulting from the NRC Setpoint Methodology Team Inspection conducted from January 14, 1991 through March 22, 1991, at the San Onofre Nuclear Generating Station, Units 2 and 3. In accordance with 10 CFR 2.201, enclosure 1 to this letter provides the Southern California Edison (SCE) reply to the NOV. A request to reduce the severity level of Part C of the violation is provided in the Part C response. As discussed with Mr. Phil Johnson (NRC) on May 3, 1991, this response was delayed in order to provide a complete response.

In addition to the NOVs, the inspection report addresses a number of other issues related to the quality of our engineering program. We are reviewing the engineering program in relation to tne inspection report and will provide a separate response concerning those issues.

This inspection was documented in NRC Inspection Report Nos. 50-361/91-01 and 50-362/91-01. A detailed review of the inspection report by SCE identified a number of specific observations to which SCE wishes to provide a response. Enclosure 2 lists these observations and our responses.

If you have any questions regarding SCE's response to the Notice of Violation or require additional information, please call me.

Sincerely,

Harold B. Ray

Enclosures
cc: J. B. Martin, Regional Administrator, NRC Region V
C. W. Caldwell, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3

IE01

ENCLOSURE 1

Reply to a Notice of Violation

The enclosure to Mr. Zimmerman's letter dated April 12, 1991, states in part:

"A. Technical Specification (TS) 2.2, Limiting Safety System Settings, Reactor Trip Setpoints, states in section 2.2.1, 'The Reactor Protective Instrumentation setpoints shall be set consistent with the trip setpoint values shown in Table 2.2-1.'

"Table 2.2-1 states that the 'Steam Generator (SG) Level - low trip setpoint shall be set at greater than or equal to 21% of the distance between SG upper and low level instrument nozzles, with a minimum allowable value of greater than or equal to 20% of the same distance.'

"The SG low level reactor trip provides protection for a loss of feedwater accident and assures that the design pressure of the Reactor Coolant System (RCS) will not be exceeded due to a loss of SG heat removal. The trip setpoint ensures that the SG water inventory is sufficient to provide for RCS heat removal at least 10 minutes prior to the delivery of emergency feedwater by the Emergency Feedwater System. A SG low level Reactor Protection System (RPS) trip is initiated by any 2 out of 4 independent channels associated with either of two SGs. In addition, the Emergency Feedwater Actuation System (EFAS) associated with each SG is initiated at the low level trip setpoint for that SG. The reactor trip and EFAS actuation setpoints are established at a level greater than or equal to 21%, utilizing SG narrow range instrumentation.

"Contrary to the above, low level trip setpoints were set at approximately 17% of the distance between the SG upper and low level instrument nozzles, at the time of the NRC inspection, for both of the SGs in Unit 2 (2E-088 and 2E-089) and one of the two SGs in Unit 3 (3E-089).

"This item has been categorized as a Severity Level IV Violation (Supplement 1), applicable to Units 2 and 3."

RESPONSE TO ITEM A

1. Reasons for the Violation.

Bechtel Design Failure

During the original plant construction, Bechtel Power Company (BPC), who was contractually responsible for the calibration calculation as well as for the interface between BPC and Combustion Engineering (CE) design efforts, did not utilize the correct information concerning the instrument tap span dimensions and the value for SG thermal growth in developing the instrument scaling calibration calculations.

This occurred in spite of CE letter S-SU-1620 dated January 20, 1982, noting inconsistencies between CE specifications and measured SG tap dimensions. Due to the lapse of time since the calculations were completed, SCE has been unable to identify the specific reason why the correct values were not used.

2. Corrective steps that have been taken and the results achieved.

Steam Generator Level Setpoint Correction

Upon discovery of these specific errors a Temporary Waiver of Compliance from TSs 3.3.1 and 3.3.2 was requested and received on February 8, 1991, since the TS Operability requirements of the Steam Generator Narrow Range Level (SGNRL) instruments were determined to not have been met as a result of the calibration errors. However, the affected level instruments were capable of performing the required safety function, since sufficient margin in other elements in the calibration which contribute to the overall measurement loop uncertainty existed to accommodate the identified errors. The temporary waiver was obtained to avoid an unnecessary shutdown of both Units 2 and 3 in accordance with TS 3.0.3, as would otherwise have been required.

On February 9, 1991, the setpoints for SG low level RPS trip and EFAS actuation were raised to 27% narrow range SG level to accommodate the calculational errors. This setpoint value will ensure that the trip/actuation will occur at or above the safety analysis values.

Procedural Revisions

The procedures affected by the calculational errors have been revised to indicate that the SG low level RPS trip and EFAS actuation setpoints have been changed from 21% to 27%, and to reflect that the reset has been changed from 26% to 32%.

Setpoint Program

Subsequent to the Electrical Systems SSFI in November 1989, SCE determined that a specific setpoint reconstitution effort was necessary as a sub-task of our ongoing Design Basis Document (DBD) program. In April 1990, SCE established a dedicated setpoint group and a comprehensive program to re-evaluate all safety related instrumentation setpoints at Units 2 and 3. This program includes the following:

- Performing loop accuracy calculations.
- Reviewing calibration techniques.
- Reviewing setpoint values to establish consistency with the design bases.
- Resolution of any inconsistencies identified.

Consequently, SCE believes the setpoint reconstitution effort in conjunction with the Design Basis Document program would have found and corrected this deficiency.

3. Corrective steps that will be taken to avoid further violations.

Calculation Revisions

The scaling calibration calculation for each safety-related narrow range SG transmitter will be revised to incorporate the correct instrument tap span dimension and thermal expansion value. This item will be completed by the Unit 2 Cycle 6 outage for both Units 2 and 3.

SG Narrow Range Level Transmitters (Recalibration)

At the next refueling outage for Unit 2 (scheduled for 3RD Quarter-1991) and Unit 3 (scheduled for early 1992), the SG narrow range level transmitter instruments will be recalibrated in accordance with the revised scaling calculation. After this is accomplished on both Units 2 and 3, the SG low level RPS trip and EFAS actuation setpoints will be restored to values consistent with TS requirements.

Review of Vendor Correspondence

Bechtel Power Company no longer has the contractual responsibilities for calibration calculations.

SCE will perform a search of correspondence from CE to determine if corrective action as a result of notifications from the reactor vendor have been conducted. This search will be completed by the DBD Group as part of the preparation of the Unit 2/3 NSSS System Design Basis Document.

The DBD Program, Revision 5, was submitted to the NRC in March 1991. DBD Quality NES&L Procedure, "Setpoint Studies and Documentation Review", is being revised to include requirements to retrieve and assess CE letters which contain recommended control or operational changes. In addition to the DBD Group's final check of CE correspondence, any advance retrieval of CE setpoint references will be provided to the Loops/Setpoint Project for their verification.

4. Date when full compliance will be achieved.

Full compliance was achieved on February 9, 1991, when the steam generator level trip setpoints were reset, placing Units 2 and 3 in compliance with the Technical Specifications.

The enclosure to Mr. Zimmerman's letter dated April 12, 1991, states in part:

"B. 10 CFR 50.9, 'Completeness and Accuracy of Information,' states in paragraph (a) 'Information provided to the Commission by an applicant for a license or information required by a statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects.'

"Contrary to the above, on January 8, 1990, Southern California Edison submitted, and subsequently obtained NRC approval of, Technical Specification amendment 88 for license NPF-10 and Technical Specification amendment 78 for license NPF-15, to extend surveillance periods for certain remote shutdown instrumentation from 18 to 24 month periods. The amendments were partially supported by a functional analysis (calculation), which at the time of the inspection, was found to be in error for the instruments that were the subject of the amendment request.

"This is a severity le' l IV violation (Supplement I), applicable to Units 2 and 3.

RESPONSE TO ITEM B

Background

Southern California Edison (SCE) submitted a Technical Specification (TS) amendment request on January 8, 1990 to extend the refueling interval surveillances from 18 months to 24 months for the Remote Shutdown (RS) instrumentation. One of the documents used to develop the amendment request was a Functional Analysis (FA) M-89068. A summary of the FA was included as attachment H to the amendment request.

The purpose of the FA was to identify the required function of the RS instrumentation per the applicable Station procedure and make an assessment of the impact of the instrument drift that occurs during the proposed surveillance interval on the required function. An analysis, in lieu of a calculation, was believed to be sufficient to make this assessment since 1) the required function of the Remote Shutdown Instrumentation is limited to providing indications to operators and 2) the Remote Shutdown Instrumentation does not initiate automatic actuations of protection systems.

The FA was used to support the conclusion that instrument drift over the 24 month interval had little effect on the accuracy of RS instrument indications used by the operators. No other plant instruments were assessed by the FA.

During the conduct of the NRC Setpoint Methodology Team Inspection, the FA was found to contain errors concerning the "as built" plant instrumentation used in the total instrument loop uncertainty calculations.

1. Reasons for the Violation.

Inadequate Supervisory Direction and Guidance

During the fall of 1989 when the FA was written, the group supervising engineer responsible for the FA failed to provide direction and ongoing supervision to the lead project engineer to ensure that the FA was prepared correctly and without errors. The lead project engineer in charge of preparing the FA assigned tasks and provided guidance to the engineer responsible for the total loop uncertainty (TLU) calculations without requiring and without verifying that the details of the TLU calculations were given sufficient attention to ensure they were correct and without error.

Contributing Cause - Inadequate Review

Review of the FA was performed in accordance with 10CFR50 Appendix B Criteria III and included interdisciplinary reviews but did not include the rigorous review performed by an Independent Review Engineer (IRE). The reviews conducted were inadequate.

2. Corrective steps that have been taken and the results achieved.

Administratively Re-established 18 Month Surveillance Interval

On February 28, 1991, SCE administratively re-established the 18 month surveillance interval for RS instrumentation. SCE never used the RS instrumentation TS surveillance interval of 24 months.

Increased Controls for Engineering Studies and Analyses

The group supervising engineer in charge of the FA is no longer with Nuclear Engineering Design Organization (NEDO). The lead project engineer is no longer with SCE. The engineer who performed the TLU calculations has been counseled on this issue.

Because it was recognized that the FA reviews were inadequate, NEDO engineers were directed in March 1991, to use Nuclear Engineering, Safety & Licensing (NES&L) Quality Procedure 24-7-15, "Preparation and Verification of Design Calculations," for all engineering studies that are used to support licensing activities. Use of this procedure will provide the controls to ensure this type of engineering work receives the required attention to detail and receives an independent (IRE) review.

3. Corrective steps that will be taken to avoid further violations.

Preparation of a New Functional Analysis

A new analysis will be prepared in accordance with NES&L Design Calculation Procedure 24-7-15 and the results will be transmitted to the NRC by July 29, 1991.

Procedure Revisions

NES&L procedures will be revised by September 15, 1991 to incorporate the direction provided to NEDO engineers concerning preparation of engineering work used to support licensing submittals as previously discussed in item 2 above.

4. Date when full compliance will be achieved.

Full compliance was achieved on February 28, 1991 when SCE administratively returned to the 18 month surveillance interval for Remote Shutdown Instrumentation.

The enclosure to Mr. Zimmerman's letter dated April 12, 1991, states in part:

"C. 1. Technical Specification 6.8.1 states, that '... written procedures shall be established, implemented, and maintained covering ... Modification of Core Protection Calculator (CPC) Addressable Constants.' This requirement is implemented by: (a) SONGS procedure SO23-V-4.7, Revision 6, Temporary Change Notice 6-2, entitled 'Control of Core Protection Calculator Addressable Constants', which states in step 6.2.2.4: 'Documentation shall be attached (to the CPC Addressable Constants Change Request) which shows how the requested values were obtained or calculated;' (b) SONGS procedure SO23-V-1.12, Revision 10, Temporary Change Notice 10-4, 'Power Distribution Monitoring', which states in Attachment 5 the 'c1' and 'f' values from Attachment 7 are to be used in the calculation of CPC Addressable Constant ARM1 and Core Operating Limits Supervisory System (COLSS) Addressable Constant AB1(01).

"Contrary to the above: (a) documentation was not attached to ten of twenty CPC Addressable Constant Change Requests for forty three of forty five individual CPC Addressable Constant Changes, from the period of April 18, through December 7, 1990, for Units 2 and 3. For two of the forty five CPC Addressable Constant Changes, the required documentation was never generated; and (b) CPC Addressable Constants Change Request 89-008-2 dated December 19, 1991 [sic], for ARM1 and AB1(01), was performed using data received from the NSSS vendor, not from SO23-V-1.12 Attachment 7. Procedure SO23-V-1.12 Attachment 7 was updated, to include cycle 5 data, on December 26, 1991 [sic].

"2. Technical Specification 6.8.1 states, in part: 'Written procedures shall be established, implemented, and maintained covering ... the recommendations of Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

"Regulatory Guide 1.33, Rev 2, Feb 1978, Paragraph 9, recommends 'Procedures for Performing Maintenance.'

"Regulatory Guide 1.33, Revision 2, February 1978, is implemented by SONGS procedure SO23-V-1.12, Revision 10, Temporary Change Notice 10-4, 'Power Distribution Monitoring,' which states:

a. In Attachment 2, that the Burnup Conversion Factor for the burnup verification calculation is to be obtained 'from Attachment 7 for the appropriate Unit and fuel cycle.'

b. In Attachment 3, that 'AB1(01), c1, f, PF3, and ARM1 values from Attachment 7' shall be used for the CPC and COLSS Peaking Factor Calculations.

"Contrary to the above, procedure SO23-V-1.12, including Attachments 2 and 3, was performed on December 16, 1989, and again on December 19, 1989, for Unit 2 cycle 5 using cycle 4 data for cycle 5 burnup calculations in Attachment 2, and using values from NSSS vendor information, and not from Attachment 7, for Attachment 3, CPC and COLSS Peaking Factor Calculations.

"This is a severity IV violation (Supplement 1), applicable to Units 2 and 3."

RESPONSE TO ITEM C

1. Reasons for the Violation.

Failure to comply with procedures.

Lack of administrative attention to detail in execution of procedures.

Supporting documentation was not attached to 10 of 20 Core Protection Calculator (CPC) Addressable Constant Change Requests (ACCR) as required by SO23-V-4.7, "Control of Core Protection Calculator Addressable Constants", step 6.2.2.4. The supporting documentation for 8 of the 10 ACCRs, which shows how the requested values were obtained or calculated, was used by the engineers to generate the ACCRs but was not attached to the ACCRs when transmitted to CDM. However, this documentation was transmitted separately to CDM as required by other Computer Engineering procedures. Forty three of forty five individual CPC Addressable Constant changes were contained in these 8 ACCRs.

The computer engineers were cognizant that this supporting documentation was required to be transmitted to CDM. Because the documentation was preserved and retrievable from CDM, the engineers considered that the intent of step 6.2.2.4 was met. This failure to comply with the procedure was due to a lack of formal attention to administrative details in execution of the procedure.

Supporting documentation was not needed for the other 2 ACCRs, because these 2 ACCRs dealt with the azimuthal tilt (TR) constant which is required to be more conservative than the COLSS computed values and consequently, the operators are required to change TR as necessary to meet this condition. However the absence of the documentation was not explained and attached to the ACCRs. The failure to document the acceptable absence of supporting documentation was also due to a lack of attention to detail when executing the procedure.

The CPC and COLSS Peaking Factor Calculations performed using SO23-V-1.12, "Power Distribution Monitoring", Attachment 3, correctly used data obtained from the NSSS vendor rather than from Attachment 7. However, the engineers failed to formally update Attachment 7 based on this NSSS vendor data before using the procedure. This failure to comply with the procedure was due to a lack of attention to detail.

CPC ACCR 89-008-02 for ARM1 and AB1(01) was performed using data received from the NSSS vendor instead of Attachment 7 of SO23-V-1.12, "Power Distribution Monitoring", contrary to the procedure. Similar to the COLSS Peaking Factor Calculations previously discussed, the engineers failed to formally update Attachment 7 based on this NSSS vendor data before using the procedure. This failure to comply with the procedure was due to a lack of attention to detail.

Inapplicable use of engineering judgement in execution of a procedure.

Step 6.3.1 was added to procedure SO23-V-1.12, "Power Distribution Monitoring" on 5/26/88 as a good engineering practice. This step converts the CECOR output EOTCYC (Cycle Average Exposure) from Megawatt Days/Metric Ton (MWD/MTU) units to Effective Full Power Days (EFPD) units. This conversion was added to the procedure as an "approximate cross check" between two independent methodologies, the hand calculated EFPD and CECOR calculated burnup. This conversion calculation requires use of the appropriate cycle Burnup Conversion Factor from Attachment 7. If the cross check fails to indicate agreement within ± 3 EFPD, step 6.3.1 states that further action is left to "discretion of the test engineer".

The reactor engineers involved in this event are intimately familiar with the procedure. They understand this step to be a cross check which in this instance had no technical significance because the core was at the Beginning of Cycle (BOC) and consequently, the burnup was zero. Nevertheless, personnel involved should have either properly completed the procedural steps or obtained a Temporary Change Notice to modify the procedure. Therefore, this failure to comply with the procedure was due to inapplicable use of engineering judgement in execution of the procedure.

SCE has reviewed the discrepancies identified in the violation and concludes there was no safety significance. Therefore, SCE believes it qualifies as a Severity Level V Violation (Supplement I) and requests the NRC reduce the Severity Level.

2. Corrective steps that have been taken and the results achieved.

Procedural Compliance - Verbal Guidance

Technical Division Investigation Report (TDIR) 91-001 was prepared to evaluate the use of the correct cycle burnup conversion factor during execution of SO23-V-1.12. During preparation of the TDIR computer engineering supervision interviewed, and subsequently counselled, all members of Reactor Engineering and reaffirmed the requirement for explicit procedural compliance. Direction was given that if the procedure can not be followed as written, the procedure must be corrected before proceeding. The engineers were also advised that failure to comply explicitly with procedures may result in disciplinary action.

Nuclear Oversight (NOD) Audit

As a result of the NRC finding regarding use of the incorrect cycle burnup conversion factor, Safety Engineering performed an audit, FE 91-01, of all SO23-V-1.12 executions from August 1 1988 to December 31, 1990 and a random sample of executions of other Reactor Engineering procedures pertaining to CPCs. The audit did not identify any safety significant findings and was consistent with the NRC inspection findings regarding SO23-V-1.12.

Procedure Modifications

SO23-V-4.7 was revised on May 1, 1991 to explicitly state that the supporting documentation for CPC ACCRs must be attached to the CPC ACCRs and submitted to CDM.

Procedure SO23-V-1.12 was revised on April 26, 1991 to clarify the use of engineering judgement in step 6.3.1.

3. Corrective Actions that will be taken to avoid further violations.

Required Reading

After issuance, the TDIR 91-001 will be required reading for all Station Technical (STEC) personnel, which includes Computer Engineering personnel. This TDIR underscores the requirement for explicit procedural compliance and the potential for disciplinary action for failure to explicitly comply with procedures. STEC personnel will complete this required reading by June 30, 1991.

4. <u>Date when full compliance will be achieved.</u>

Full compliance was achieved regarding the supporting documentation for eight of the ten CPC ACCRs on February 8, 1991, when the required documentation was verified and retrieved from CDM. The remaining 2 CPC ACCRs did not require documentation.

Full compliance was achieved regarding Attachment 7 of SO23-V-1.12 on November 8, 1990, when Attachment 7 of the procedure, was updated (TCN 10-7) with the current data.

The enclosure to Mr. Zimmerman's letter dated April 12, 1991 states in part:

"D. Technical Specification 6.8.1 states, in part: 'Written procedures shall be established, implemented and maintained covering ... c. Surveillance and test activities of safety related equipment.'

"Contrary to the above, acceptance criteria for instrument and control measuring and test equipment (M&TE) in Southern California Edison (SCE) surveillance procedures do not require the four to one (4:1) accuracy ratio assumed in instrument loop uncertainty calculations. SCE instrument and control procedures allow an accuracy of 1:1 for M&TE.

"This is a level IV violation (Supplement 1), applicable to Units 2 and 3."

RESPONSE TO ITEM D

1. Reasons for the Violation.

Incorrect Design Assumptions

SCE has investigated this situation and has concluded that it is a violation of 10CFR50 Appendix B Criterion III, Design Control, rather than a violation of Technical Specification (TS) 6.8.1. Approved surveillance procedures were established, maintained, and used; however a discrepancy existed between the accuracy required by the surveillance procedures and that assumed by the design calculation.

Combustion Engineering (CE), the Nuclear Steam Supply System (NSSS) vendor, prepared the instrument loop uncertainty calculations which support plant instrumentation. The calculations cover instrumentation associated with the Core Operating Limits Supervisory System (COLSS), Core Protection Calculator (CPC), and Plant Protection System (PPS).

In performing the calculation, CE incorrectly assumed that measuring and test equipment (M&TE) is four times the accuracy of the instrumentation being calibrated (4:1). SCE's program, as described in the Topical Quality Assurance Manual (TQAM), requires an M&TE accuracy "equal to or greater than" the permanent plant instrumentation under calibration. CE did not review its 4:1 M&TE assumption against SCE's standards and did not identify to SCE the need to utilize test equipment with greater accuracy than that required by those standards.

Previously Identified Design Interface Problem

As was discovered in conjunction with the Component Cooling Water (CCW) Safety System Functional Inspection (SSFI), the Electrical SSFI, and during SCE's Design Basis Documentation (DBD) work, engineers made assumptions in calculations during the original plant design phase that were not incorporated into or otherwise reconciled with the operation and maintenance of the plant. The secondary calorimetric instrumentation loop uncertainty calculation's use of the 4:1 accuracy assumption for M&TE is another example of this previously identified design interface problem.

2. Corrective steps that have been taken and the results achieved.

Evaluation of Secondary Calorimetric Uncertainty Calculation

On March 25, 1991, NEDO completed an evaluation of the secondary calorimetric instrumentation loop uncertainty calculation. The evaluation determined that the secondary calorimetric power measurement uncertainty setpoint had sufficient conservatism to compensate for the incorrect 4:1 accuracy assumption. Therefore, the results of the calculation are unaffected by the incorrect assumption of a 4:1 accuracy for M&TE used to calibrate the instrumentation.

Engineering Program Changes

In June 1990, the new SCE setpoint program was implemented. The new setpoint program procedures require the Nuclear Engineering Design Organization (NEDO) to review I&C calibration procedures prior to proceeding with setpoint calculations. This ensures that NEDO will identify inconsistencies between design calculation assumptions and I&C calibration practices.

3. Corrective steps that will be taken to avoid further violations.

Revision of Program Controls for Vendor Design Interface

Revisions will be made to procedures that control the SCE interface with vendors to ensure a more rigorous review of vendor supplied calculations. These changes will be completed by November 1, 1991.

4. Date when full compliance was achieved.

Full compliance was achieved on March 25, 1991 when it was confirmed that the secondary calorimetric instrument loop uncertainty calculation was unaffected by the incorrect CE assumption concerning M&TE accuracy.

ENCLOSURE 2

The purpose of the information below is to provide data concerning noted sections of the inspection report.

Executive Summary Page ii	"The team also identified another Functional Analysis M89047, which was performed outside of the quality assurance procedural requirements..."
	Our review indicates that the Drift Study, M89047, was completed under the QA program as it existed when the work was performed. In fact, the Study received several reviews above and beyond the QA requirements of the program.
Inspection Report Page 4	"The team reviewed calculation 1370-TS-103. The Team identified a number of errors and incorrect assumptions. Some of the errors appeared to be due to misinformation provided by SCE to CE as the bases for the calculations. The total number of errors indicated an inadequate SCE review of the calculation."
	In calculation 1370-TS-103 CE made the assumption that the instrumentation used remained installed in the plant. This was not a result of misinformation provided by SCE to CE, as indicated. The "as built" instrumentation configuration was properly documented and the information was available upon request from CE to SCE.
Inspection Report Page 5 2.3.5	The IR, when referring to Rosemount model 1151DP transmitters, states that "Five of the eight feedwater flow transmitters were Rosemount model 1151DP. None of these 5 transmitters appeared to be calibrated in accordance with SCE and vendor procedures."
	Our review indicates that SCE has been calibrating these instruments in accordance with SCE-approved procedure SO123-II-9.10. SCE is unaware of the basis for the indicated perception.

Inspection Report Page 10 2.10.1	This paragraph included "The review cycle did not include the verification or checking for the adequacy of design by the performance of design reviews using alternate or simplified calculational methods as prescribed in 10 CFR 50 Appendix B, Criteria III."

In the case of the Functional Analysis M89068, SCE elected to conduct an ordinary design review in accordance with 10CFR50, Appendix B, Criterion III. A design review by the use of alternate or simplified methods is not required in this circumstance.

Inspection Report Page 12 2.10.2.5	This paragraph discussed current converters as used in the Functional Analysis M89068 stating: "There were no discrete current converters in some loops for units 2 or 3 although the device was included in the uncertainty calculations... The current converter was incorporated in the calculation of loop uncertainties for the EOIs. The licensee stated that the current converters were replaced with another device under a design change; however, after further investigation it appeared that the devices may never have been installed".

M89068 has never been used in the calculations of loop uncertainties for current EOIs, and the present installed equipment and plant drawings are in agreement with each other. SCE concurs that some of the instruments used in the M89068 Functional Analysis were incorrect and in this case the current converters used in M89068 were not the instruments installed in the plant.

General Summary Page 2	"The Licensee depended on the NSSS supplier, contractors and vendors for design basis calculations, to which the licensee applied only a cursory check."

Both Bechtel and CE performed safety-related design calculations in accordance with QA programs developed in accordance with 10CFR50, Appendix B, and submitted to the NRC. SCE conducted regular audits to ensure that the QA programs by these major contractors and by other vendors were effectively implemented. Consistent with industry practice and our understanding of NRC requirements, SCE did not independently verify the design calculations so performed.