From: Linda.Conklin@sce.com
To: Benney, Brian
Subject: AREVA Fuel VQP LAR; SCE-9801-P-A SER Pages of Interest
Date: Friday, September 28, 2012 3:48:55 PM
Attachments: SONG_NT-022329[1].pdf

Brian,

Here is some additional background on the methodology that SCE uses for reload analysis for SONGS. In 1998 we submitted a topical report and subsequently received approval (Legacy Accession No 9906090108) for Topical Report SCE-9801-P, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3, The staff concluded in the SE, (based on the topical report and on an onsite audit of the implementation of the Reload Technology Transfer (RTT) Program), that SCE has the capability to perform reload design and non-LOCA accident analyses for the San Onofre Nuclear Generating Station Units 2 and 3.

Included in the SE is the following paragraph that specifically addresses change in fuel vendor.

3.2.7 Change in Fuel Vendor
"The models and methods discussed in this SE and reviewed during the audit were approved by the NRC for use by ABB/CE and have been transferred to SCE via the RTT Program. Any change in fuel vendor would require an evaluation of changes required to the physics and safety analysis methodology to accommodate that vendor's particular fuel designs. Changes of this type would require a thorough engineering evaluation, verification, and validation prior to use of the new fuel design. If necessary, a topical report covering significant changes in models and/or methods would be required by the NRC for review and approval."

If there is any NRC guidance that contradicts this please let us know.

Linda T. Conklin
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San Onofre Nuclear Generating Station
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
June 2, 1999

Mr. Harold B. Ray
Executive Vice President
Southern California Edison Company
San Onofre Nuclear Generating Station
P. O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATIONS, UNITS 2 AND 3 - EVALUATION OF RELOAD ANALYSIS METHODOLOGY TECHNOLOGY TRANSFER (TAC NOS. MA4289 AND MA4290)

Dear Mr. Ray:

By letter dated November 30, 1998, as supplemented December 31, 1998, and March 1, 1999, you submitted Topical Report SCE-9801-P, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3," for review. Based on a review of Southern California Edison's topical reports and on an onsite audit of the implementation of the Reload Technology Transfer Program, the staff has concluded that Southern California Edison has the capability to perform reload design and non-loss-of-coolant accident analyses for the San Onofre Nuclear Generating Station Units 2 and 3. Our evaluation is enclosed.

If you have any questions, please contact the San Onofre Project Manager, L. Raghavan, at (301) 415-1471. This completes our efforts on TAC Nos. MA4289 and MA4290.

Sincerely,

Stephen Dembek, Chief, Section 2
Project Directorate IV and
Decommissioning
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosure: Evaluation

cc w/encl: See next page

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EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE ABB/CE RELOAD ANALYSIS METHODOLOGY

TECHNOLOGY TRANSFER PROGRAM

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE UNIT 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By letter dated November 30, 1998, Southern California Edison Company (SCE), the licensee for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3, submitted Topical Report SCE-9801-P, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3," (Ref. 1) to the U.S. Nuclear Regulatory Commission (NRC) for review. This report describes the program that was undertaken by the licensee to develop the capability to perform independently the analysis of reload cycles for SONGS Units 2 and 3.

To develop the capability to perform independently the analyses required for the design, licensing, and operation and surveillance of a reload fuel cycle, SCE contracted Asea Brown Boveri/Combustion Engineering (ABB/CE) to provide a training program, referred to as the Reload Technology Transfer (RTT) Program. The RTT Program consisted of classroom lectures, on-the job training, and independent analysis. The scope of the program included all reload engineering technology except loss-of-coolant accident (LOCA) analysis, fuel mechanical design, and fuel fabrication engineering. These areas remain the responsibility of the fuel vendor, currently ABB/CE. The models and methods used for the reload cycle analyses covered by the RTT Program have been previously approved by the NRC for use by ABB/CE. In addition, the RTT Program is the same as that used to provide similar capability to Arizona Public Service Company (APS) for use at the Palo Verde Nuclear Generating Station (PVNGS) in 1993. APS's use of the ABB/CE methodology for PVNGS reload analyses was approved by the NRC in June 14, 1993 (Ref. 2).

In addition to Topical Report SCE-9801-P, ABB/CE compiled, at the NRC staff's request, Topical Report CEN-635(S)-P, "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology," (Ref. 3), the purpose of which was to provide a single source listing all of the limitations contained in NRC safety evaluations (SEs) on the individual computer codes used by SCE as part of the reload design and analysis process. The NRC staff has reviewed Topical Reports SCE-9801-P and CEN-635(S)-P as part of its evaluation of SCE's capability to perform reload design and analysis. The staff also determined that it was necessary to evaluate the documentation associated with SCE's implementation of the RTT program. Accordingly, the NRC conducted an onsite audit at SCE's

ENCLOSURE

SONGS engineering offices in San Clemente, California, from March 9-12, 1999. The NRC audit team consisted of Alan Levin, Edward Kendrick, and Anthony Attard of the Office of Nuclear Reactor Regulation. This SE reflects the results of the staff's review of the topical reports and the staff's evaluation of the RTT Program implementation during the onsite audit.

2.0 COMPUTER CODES USED IN THE RELOAD DESIGN AND ANALYSIS PROCESS

The reload design and analysis process uses a large number of ABB/CE-developed computer codes, which have been provided by ABB/CE to SCE. The major codes used in the process are listed below.

The following computer codes are used in the physics analyses:

1. ROCS (Ref. 4), a coarse-mesh, two-energy group higher order difference diffusion theory neutronics code that can model all aspects of reactor operations from startup to refueling;

2. MC (Ref. 4), a fine-mesh, two-energy group diffusion theory neutronics code that calculates fine-mesh (pin-wise) flux, power, and burnup distributions through the application of the nodal imbedded method to individual fuel assemblies using inter-assembly currents calculated by the coarse-mesh ROCS code;

3. HERMITE (Ref. 5), a few-group, space- and time-dependent neutron diffusion code that includes feedback effects of fuel temperature, coolant temperature, coolant density, and control rod motion;

4. QUIX (Ref. 6), a two-group, one-dimensional diffusion code used for axial shape analysis; and

5. CECOR (Ref. 7), a code that determines fuel assembly burnup for the purpose of evaluating spent fuel storage and boraflex requirements.

The core thermal-hydraulic analyses use the following codes:

1. TORC (Ref. 8), a three-dimensional, open-lattice core thermal-hydraulics code used to determine the local coolant conditions and, in turn, the minimum departure from nucleate boiling ratio (DNBR) for the core; and

2. CETOP-D (Ref. 9), a fast-running variant of the TORC code used as a design code in thermal margin analyses.

The fuel performance analyses use the following codes:

1. FATES3A (Ref. 10), a fuel evaluation code that predicts the steady-state fuel rod temperature distribution, gap conductance, fuel and clad dimensions, plenum pressure, and stored energy for ABB/CE-designed fuel, and includes the NRC-required grain size restriction in the fission gas release calculation; and

2. FATES3B (Ref. 11), a revised version of FATES3A with an improved predictive capability at high

burnup.

In addition to the HERMITE and CETOP-D codes described above, the non-LOCA transient and accident analyses are performed with the following codes:

1. CESEC-III (Ref. 12), a system code that incorporates point kinetics, reactivity feedback, and core thermal-hydraulics to calculate system parameters, including core heat flow, pressure, and temperature during a transient;

2. CENTS (Ref. 13), an upgraded version of CESEC-111;

3. STRIKIN-II (Ref. 14), a code that provides a single, or dual, closed-channel model of a core flow channel to calculate the clad and fuel temperatures during a transient; and

4. HRISE (Ref. 15), a code used in non-LOCA analyses to determine the fuel DNBR using the MacBeth correlation, at specified transient conditions.

3.0 EVALUATION

3.1 Reload Desi~n Process

The specific disciplines required of SCE to perform reload design and analyses are:

1. Physics design
2. Core thermal-hydraulic design
3. Fuel performance design
4. Transient and accident analysis (except LOCA)
5. Generation of Core Operating Limit Supervisory System (COLSS) and Core Protection Calculator System (CPCS) setpoints, database constants, and core operating margin assessment

Topical Report SCE-9801-P presented a description of the entire RTT Program, including training, methodology, and comparisons of analysis results from several reload fuel cycles. Following classroom instruction in ABB/CE's codes and methods, on-the job training was provided to SCE engineers. On-the job training comprised a series of reload fuel cycle analyses, performed by ABB/CE and SCE engineers working together. SCE engineers performed parts of SONGS Unit 3, Cycle 8, analyses at ABB/CE's offices as part of ABB/CE reload analysis teams. ABB/CE engineers provided an independent review of SCE's analyses. For Unit 2, Cycle 9, analyses were performed by SCE engineers at SCE's offices, with ABB/CE engineers as co-authors and independent reviewers. The transition from on-the job training to the final phase of the RTT Program, independent SCE analysis, was made for Unit 3, Cycle 9, and Unit 2, Cycle 10. All Unit 3, Cycle 9, analyses were performed by SCE engineers independent of ABB/CE, with an initial independent review by SCE engineers, and a second independent review by ABB/CE engineers, followed by ABB/CE management approval. Unit 2, Cycle 10, analyses were also performed by SCE engineers, reviewed by SCE engineers, and approved by SCE management under SCE's Quality Assurance (QA) Program. The topical report presents comparisons of parameters derived from reload fuel cycle analyses

for Unit 2, Cycle 10, with calculations for Unit 2, Cycle 9, and Unit 3, Cycle 9. Comparisons were also presented between Unit 3, Cycle 9, and Unit 2, Cycle 9, to provide a more complete set of comparisons.

This report tables of typical physics parameters as calculated Dy ~t, In comparison to those calculated by ABB/CE. Typical physics parameters include control rod worth, control element assembly reactivity worth, peak linear heat rate, peak rod burnup, fuel assembly relative power densities, boron dilution, Doppler coefficients, and moderator temperature coefficients (MTCs). The data presented in the tables indicate very good agreement between the SCE and ABB/CE analyses. Typical cycle-to-cycle and unit-to-unit differences are present as one would expect, particularly radial power and peak linear heat rate. Differences are attributable to changes in burnable absorber from boron carbide rods to erbia incorporated in the fuel rods, and to reduction of the reactor coolant system inlet temperature (TCold reduction program), undertaken by SCE to prolong steam generator life. There is a slight difference in the MTC values for Unit 2, Cycles 9 and 10, due to a bias applied for Unit 2, Cycle 10.

One of the limits imposed on the power level of a reactor is the minimum allowable value for the DNBR (MDNBR), referred to as the specified acceptable fuel design limit. MDNBR expresses the adequacy of cooling in the most limiting flow channel in the reactor core and is therefore a measure of the core thermal margin.

The ABB/CE reload design and analysis process employs a modified statistical combination of uncertainties (MSCU) methodology to calculate the MDNBR limit value, and the DNBR probability density function (PDF) is used with a 95/95 probability/confidence DNBR tolerance limit in the COLSS and CPCS uncertainty analyses. The MSCU methodology has been approved by the NRC (Refs. 16, 17).

Thermal-hydraulic system parameters and state parameters are those that describe the physical system and the operational state of the reactor, respectively. Uncertainties are associated with the determination of these parameters; consequently, the MSCU methodology is used to combine the parametric uncertainties statistically to generate the DNBR PDF, and to incorporate their effects on the DNBR to derive the MDNBR limit. This DNBR limit ensures that when the nominal values of system parameters are used in design analysis, there is a 95% probability, with 95% confidence, that departure from nucleate boiling will not occur anywhere in the core.

SCE performed cycle-specific comparisons and sensitivity analyses of the DNBR response surface derived using the MSCU methodology. The sensitivity analyses were performed with the TORC computer code. All values for Unit 2, Cycle 10, TORC analyses were bounded by the reference response surface, and the results were comparable to those for Unit 2, Cycle 9, and Unit 3, Cycle 9.

Non-LOCA transients, such as decrease in feedwater temperature, increase in feedwaler flow, loss of load, turbine trip, and loss of condenser vacuum form part of the Updated Final Analysis Report (UFSAR) Chapter 15 events. Not all Chapter 15 events are analyzed for every reload; however, events that establish limits on fuel failures and thermal margin are expected to be affected by a reload design. Events that are normally not reanalyzed are insensitive to or unaffected by reload changes or are bounded by other events.

Review of the thermal-hydraulic data presented in the topical report shows good agreement for cycle-to-cycle and unit-to-unit comparisons. Minor differences are again attributable to changes in burnable poison and TCOLD as was true for the physics parameters. Core-wide parameters, such as core average heat generation, core heat transfer area, core average heat flux, and core mass flux, show very good agreement when compared on a cycle-to-cycle basis, and are consistent on a unit-to-unit basis. The low power physics tests and the power ascension tests currently performed at beginning of cycle cover sufficient physics parameters to reasonably assure that the core is operating as designed, and that adequate shutdown margin is available. In addition, SCE conducts a core follow program, monitoring plant data and comparing those data to predicted performance to ensure that the core is performing as expected. The staff concludes that the core follow program currently performed by SCE is adequate to monitor physics and thermal-hydraulic parameters throughout core lifetime.

At the conclusion of the RTT Program, ABB/CE evaluated SCE's capability to perform independent reload fuel cycle analyses using CE's methodology and determined that SCE had satisfactorily completed the RTT Program and certified SCE's capability to perform independent reload fuel cycle analyses.

3.2 Staff Audit

In conjunction with its review of Topical Report SCE-9801-P, the NRC staff determined that it was necessary to evaluate the implementation of the RTT Program at SCE. An onsite audit was performed by an NRC team to examine documentation associated with the reload process, including procedures, checklists, design calculations and associated reviews, facility change evaluations, and other QA records. In addition, the audit team reviewed training requirements, training materials, SCE training records, and correspondence between ABB/CE and SCE. SCE also provided oral presentations on the overall RTT Program, the results of an independent audit of SCE's reload analysis process by ABB/CE, and the MSCU methodology. The results of the audit are summarized below.

3.2.1 Reload Calculations and Associated Documentation

During the staff audit, selected calculations were reviewed and discussions were held with SCE engineers to clarify specific points. The SONGS Unit 2, Cycle 10, reload design was the primary focus, since this was the first reload for which SCE assumed full responsibility, with independent reviews by ABB/CE, including an onsite audit by a senior ABB/CE engineer, which found no major deficiencies in SCE's implementation of the reload design and analysis methodology. The review also cross-referenced design areas from the Units 2 and 3, Cycle 9, reload processes, as a transition from ABB/CE to SCE responsibility. The calculations reviewed included fuel performance design, physics, thermal-hydraulics, transient analyses, CEA worth (by boron dilution and by exchange worth), and shutdown analyses. The documents establishing the bases for the reload design and analyses, particularly the Reload Ground Rules (RGR), and their associated procedures, were also reviewed.

3.2.2 Reload Ground Rules

The SCE RGR Control Methodology procedure (SO23-XXXVI-4.2) determines the requirements and structure for the reload-cycle-specific RGR document. The RGR establishes the bases and controls for the reload design analysis process. This includes the specification of SONGS plant-specific data and the SCE-approved inputs and assumptions that are provided to ABB/CE via a QA interface. This document also specifies that the Supervisor, Nuclear Safety Analysis, assigns both the analyst and the independent review engineers, in compliance with the SCE Topical QA Manual (TQAM). Another purpose is to ensure that all analysis values bound plant values, including mechanical and analytical uncertainties and margins. The RGR is also a link to the Facility Change Evaluation (FCE), which is developed according to procedure SO123XXIV-10.16. The Unit 2, Cycle 10, RGR (RGR-U2-C10, Revision 1, February 22, 1999) was reviewed in detail to ensure compliance with the governing procedures and other related procedures and QA requirements. The staff found the RGR notebook to be complete, from the documentation of reactor core design operating margins in the Ground Rules Memorandum to the completion of the Reload Analysis Report (RAR), the FCE, and the required technical specification and licensee-controlled specifications changes. The RGR also acknowledged the industry guidelines provided in INPO SOER 96-02, "Design and Operation Considerations for Reactor Cores." The SONGS plant input data and assumptions were judged to be complete and adequate to serve as a QA data source for the ABB/CE interface to mechanical design and LOCA analyses input.

3.2.3 Reload Checklist

The SCE Reload Checklist procedure (S023-XXXVI-2.10) both lists and tracks the tasks required for the completion of the reload design process. The Unit 2, Cycle 10, checklist document was reviewed to evaluate compliance with the governing procedure and with other related procedures and QA requirements. The checklist provides the baseline for the RAR and is a documented record of meetings, dates, and correspondence, beginning with the reload kickoff meeting, specification of the end of Cycle 9 and beginning of Cycle 10 dates, the Cycle 10 energy requirements and fuel delivery date, and the RGR review meeting. The generation of the Nuclear Design Data Book and Plant Physics Data Book is also documented in the Reload Checklist. The generation, independent review, and multi-level approvals of the FCE are also documented. The checklist document was judged to be complete and adequate to document the reload design process.

3.2.4 Review of Reload Design Calculation Files

As noted previously, the staff reviewed selected fuel performance, physics, and thermal-hydraulics analyses and supporting documentation. Numerous calculation file packages for the Unit 2, Cycle 10, reload design were reviewed in detail to evaluate the overall reload design and analysis process. Sources of design inputs and verifications were evaluated, as well as design interface control and documentation of results. The staff verified that only approved procedures and software were used, and that the principal analyst (author) and independent reviewer were certified as qualified for the individual task calculations. In the course of this review, a number of other related calculation files were discussed, either as a source of input or an interface to a downstream calculation. The files reviewed were complete and generally showed evidence of thorough independent review and discussion.

Low power physics, such as assembly relative power density distribution, relative axial power distribution, peaking factors, and power ascension calculations, was reviewed to verify that all acceptance criteria were met.

The staff examined calculations pertaining to Unit 2, Cycle 10, depletion analyses, stuck rod analyses, fuel management patterns, enrichment zone patterns, and full-core CEA patterns, as well as the assumptions made in the calculations. For some of the analyses examined, SCE referenced calculations performed for a previous cycle by ABB/CE. However, corresponding SCE calculations showed a level of detail and accuracy comparable to those performed by ABB/CE.

Calculations pertinent to LOCA and non-LOCA analyses (UFSAR Chapter 15) were also examined, including steam line break, CEA ejection, fuel failure, and inadvertent boron dilution analyses. (Although LOCA analyses are still performed for SCE by ABB/CE, data are provided by SCE to support these analyses.) Statistically determined DNB calculations were examined to ensure that SCE staff were adequately trained and that appropriate statistical methods were used for the calculations.

3.2.5 Review of COLSS/CPSC Input Analyses

Generation and QA verification of inputs to the COLSS and CPCS were reviewed, since they perform the primary core thermal limit monitoring functions. The overall uncertainty analyses for the 95/95 probability/confidence DNBR tolerance limit used by these systems is generated by the NRC-approved MSCU methodology, discussed previously. Generation of the addressable constants was evaluated with respect to the change in TCold to 540 °F and the new COLSIM algorithm used with the blowdown flow signal. The CPCS/COLSS inputs were observed to be generated in accordance with procedures and with satisfactory independent review and management approval.

3.2.6 Software Usage and Control

SCE has installed the reload analysis computer codes on an IBM AIX-RXJ6000 computer network. Configuration and user access is controlled by Nuclear Fuel Management (NFM) procedures, in cooperation with the corporate Information Technology Department. When the ABB/CE codes were brought into SCE as part of the RTT Program, changes to the codes were made to allow them to be executed on SCE's computers. The codes were fully verified and validated (V&V'd) after these changes were performed.

Although ABB/CE maintains the source codes for SCE,SCE's contract with ABB/CE and SCE's software control procedures contain provisions for SCE to maintain the source codes and to make changes to those codes. If SCE makes any source code changes, independent of ABB/CE, that change the calculated values of any safety-related parameters, SCE will perform V&V prior to use of the modified source code for licensing-related activities. If appropriate, a topical report covering the source code modifications will be prepared for NRC review and approval. In addition, ABB/CE may update or refine its NRC-approved reload analysis computer codes, or develop new reload analysis computer codes for NRC review and approval. All code upgrades and improvements provided by ABB/CE to SCE are installed using SCE's software control procedure.

In the course of the staffs review of the Unit 2, Cycle 10, calculation files, the usage and control of

computer codes was evaluated as encountered in the files. In addition, the SCE V&V process was specifically reviewed for the latest update of the ROCS 5.1 M1 coarse-mesh core simulator with the fine-mesh MC module incorporated. This code uses the ABB/CE Users Manual (CES-4, Revision 11-P, August 1996). The staff verified that the SCE code sponsor prepared the appropriate plan and verified proper code installation and operation by executing a test case with two maneuvers from the ABB/CE V&V report (VV-FE-0375, Revision 0, July 1997).

3.2.7 Change in Fuel Vendor

The models and methods discussed in this SE and reviewed during the audit were approved by the NRC for use by ABB/CE and have been transferred to SCE via the RTT Program. Any change in fuel vendor would require an evaluation of changes required to the physics and safety analysis methodology to accommodate that vendor's particular fuel designs. Changes of this type would require a thorough engineering evaluation, verification, and validation prior to use of the new fuel design. If necessary, a topical report covering significant changes in models and/or methods would be required by the NRC for review and approval.

3.3 Training

The staff met with the NFM training coordinator and reviewed training records of several SCE engineers. The staff determined that the NFM program for training and qualifying the staff assigned to perform core reload design and analyses is adequate to assure a well-qualified staff. The training procedures were reviewed, and found to detail an appropriate mix of training lectures, on-the job training, job performance measures, and knowledge assessment techniques to assure an adequately qualified staff. Written training materials used by the instructors were reviewed and found to contain the requisite level of detail to accomplish the training lecture purposes. Training records demonstrated that the NFM organization contains a staff that has been adequately trained and certified to perform the required functions. The records were complete and up-to-date, and comply with applicable QA requirements. SCE maintains a qualification guide based on ABB/CE's program and maintains staffing levels for the various specialties involved in reload design and analysis that are consistent with ABB/CE guidelines.

The NFM organization has performed reload analyses in parallel with ABB/CE to verify the ability of the NFM staff to perform these tasks acceptably. This activity provides a satisfactory assessment of the NFM organization's readiness to independently perform core reload analyses.

3.4 Additional Examples of SCE Reload Design Ability and Other Observations

SCE staff engineers performed the fuel management for the current reload cycle and for portions of other previous cycles for SONGS. The SCE staff was technically trained in fuel management design and reload analysis and adhered to fuel management guidelines designed by SCE and reviewed and accepted by ABB/CE. These guidelines were based on SCE prior reload analyses and provide up-front guidance to avoid safety analysis problems, placing limits on key core physics parameters. SCE has also demonstrated that it has the

capability independently to complete justifications for continued operation, evaluations pursuant to 10 CFR 50.59, nuclear safety assessments following plant events, and safety assessments for technical

specification changes.

QA was a particular strength that was evident to the NRC staff throughout the audit. Analyses are "tailboarded" prior to being conducted, to promote clear communication between management and the analysts. Completed analyses are given a thorough independent review and evaluation. To provide an additional QA check on reload analyses, SCE has established an Analysis Review Committee (ARC), comprising the Manager, Nuclear Fuel Engineering and Analysis, and the three supervisors reporting to him (Nuclear Fuel Analysis, Nuclear Safety Analysis, and Core Performance Analysis). SCE analysts present their calculations to the ARC, which then provides comments that the analyst(s) must resolve. Minutes of the ARC meetings are maintained; the NRC staff reviewed the minutes from meetings during 1998 and 1999. The staff concluded that this step provided a valuable additional method of reviewing analyses. QA procedures are comprehensive and appear to be followed closely. The training program is also comprehensive and well documented. Management commitment to and involvement in QA activities was evident at all levels. These observations were consistent with those developed in ABB/CE's audit of SCE. A minor weakness was identified in interface activities by ABB/CE's auditor, and the NRC staff noted some examples of this weakness. However, both ABB/CE and SCE are aware of these weaknesses and have determined to work to improve them.

4.0 CONCLUSION

Based on the staff's review of SCE's topical report and the onsite audit, and recognizing that SCE has participated in the ABB/CE RTT Program, the staff concludes that the SCE staff has the capability to use the approved ABB/CE reload design and analysis methodology for reload design and non-LOCA reload analyses of the ABB/CE-fueled SONGS cores. The Topical Report, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3," SCE-9801-P (Ref. 1) describes the reload design process and the scope of the analyses which may be performed by SCE, and is acceptable for referencing in SONGS licensing applications.

Principal Contributor: A. Levin

Date: June 2, 1999

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