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A. Edward Scherer
Director
Nuclear Regulatory Affairs

September 29, 2009

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: **San Onofre Nuclear Generating Station, Units 2 and 3
Docket Nos. 50-361 and 50-362
Response to Request for Additional Information on Request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies, and Proposed Change Number (PCN)-589, Amendment Application Numbers 254 and 240, Respectively for Units 2 and 3 Request to Revise Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)"**

References:

1. Letter from Michael P. Short (SCE) to NRC (Document Control Desk) Dated January 30, 2009, Subject: Docket Nos. 50-361 and 50-362 Request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies, and Proposed Change Number (PCN)-589, Amendment Application Numbers 254 and 240, respectively for Units 2 and 3 Request to Revise Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)" San Onofre Nuclear Generating Station, Units 2 and 3

2. Letter from A. E. Scherer (SCE) to NRC (Document Control Desk) Dated March 16, 2009, Subject: Docket Nos. 50-361 and 50-362 SCE Response to NRC Questions Regarding SCE Request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies, and Proposed Change Number (PCN)-589, Amendment Application Numbers 254 and 240, respectively for Units 2 and 3 Request to Revise Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)" San Onofre Nuclear Generating Station, Units 2 and 3

Dear Sir or Madam:

This letter responds to Nuclear Regulatory Commission (NRC) request for additional information which was made on August 26 regarding the referenced request for

A001
NRR

P.O. Box 128
San Clemente, CA 92674

Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies, and Proposed Change Number (PCN)-589, Amendment Application Numbers 254 and 240, respectively for Units 2 and 3 Request to Revise Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)."

The enclosure repeats the NRC requests for additional information along with their respective Southern California Edison (SCE) responses.

There are no new commitments contained in this letter and the No Significant Hazards Consideration and Environmental Evaluation provided with PCN-589 remains bounding.

We discussed the timing of this response with the NRC staff on September 24, 2009 and based on that discussion, we understand that this letter is considered timely.

Should you have any questions, or require additional information, please contact Ms. Linda T. Conklin at (949) 368-9443.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 09/29/09
date

Sincerely,

Enclosure:

Responses to NRC Request for Additional Information on SCE request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies, and Proposed Change Number (PCN)-589, Amendment Application Numbers 254 and 240, respectively for Units 2 and 3 Request to Revise Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)"

cc: E. E. Collins, Regional Administrator, NRC Region IV
R. Hall, NRC Project Manager, San Onofre Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, San Onofre Units 2 and 3
S. Y. Hsu, California Department of Health, Radiologic Health Branch

Enclosure

Responses to NRC Request for Additional Information on SCE request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies, and Proposed Change Number (PCN)-589, Amendment Application Numbers 254 and 240, respectively for Units 2 and 3 Request to Revise Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)"

<u>RAI 1</u>: Nonlimiting Locations for the LFAs in the core

Enclosure 1 of SCE's request for temporary exemption and TS amendment states (paragraph 3 on page 5 of 7) that, "An underlying assumption of the LFA program is that a 5% radial power peaking penalty will be sufficient to ensure that the LFAs will be nonlimiting in the safety, fuel performance, thermal hydraulic, and ECCS performance analyses...." Also, the licensee states that, "since the LFAs will not be in the highest core power density locations, the placement scheme assures that the behavior of the LFAs is bounded by the safety analyses performed for the co-resident Westinghouse fuel." Provide the following information based on the above statements:

(1) Provide a detailed technical <u>basis</u> for the statement that a 5% radial power peaking penalty will be sufficient to ensure that the LFAs will be nonlimiting in the safety, fuel performance, thermal hydraulic, and ECCS performance analyses.

(2) Provide the details of analyses and methodologies used to identify the locations that are non-limiting locations. Provide the criteria and the key parameters used to determine the non-limiting locations; and

(3) Since LFAs are to be used in the subsequent cycles (Cycles 17 and 18), provide justification for the assumption/prediction that the LFAs will remain in nonlimiting core regions.

<u>RAI 1, Part (1) Response</u>

Under the SCE AREVA Lead Fuel Assembly (LFA) Program, up to sixteen LFAs manufactured by AREVA NP may be inserted into the SONGS (San Onofre Nuclear Generating Station) Unit 2 core or potentially into the SONGS Unit 3 core. Currently, eight LFAs are scheduled for installation in Unit 2 Cycle 16, with use for up to three operating cycles (Cycles 16, 17, and 18). AREVA LFAs will use proven designs (grids, fuel rod, guide tubes, etc.) and proven cladding material (M5) used in AREVA fuel assemblies in Combustion Engineering units similar to SONGS (e.g., Palo Verde, Calvert Cliffs). The LFAs will be analyzed by SCE, Westinghouse, and AREVA using NRC approved methodologies and will be operated within the current licensing limits.

SCE will model the AREVA LFAs in the SONGS core physics models and their impact will be analyzed in the cycle-specific core physics calculations that support the reload analyses (commitment #5 in Enclosure 2 of the request). Per Technical Specification 4.2.1, the AREVA LFAs will be placed in non-limiting core locations. Consistent with commitment #4 in Enclosure 2 of the request, this is defined as the peak integrated radial power peaking factor in the LFAs is 0.95 or less of the core maximum integrated radial power peaking factor (Fr) at all times in life. The 0.95 power peaking factor criterion represents a 5% allowance which bounds the 95/95 calculational uncertainty for Fr, ensuring that the LFAs will be nonlimiting in terms of power peaking. Therefore, the LFAs will not contain the limiting rod in the core and will have margin relative to the bounding peaking factors used in the safety analyses. The 0.95 power peaking factor criterion is consistent with that used in the AREVA LFA program at Palo Verde (ADAMS Accession Number ML082620212). Since the LFAs will not be in the highest core power density locations, the placement scheme assures that the behavior of the LFAs is bounded by the safety analyses performed for the co-resident fuel rods. This is verified by analyses performed by SCE, AREVA, and Westinghouse.

This 5% reduction in power peaking for LFAs also provides additional margin for fuel performance, thermal hydraulic, and ECCS analyses for LFAs. Additional discussion is provided in the response to RAI 2.

RAI 1, Part (2) Response

SCE will perform explicit SONGS core physics modeling and analysis of the reactor cores containing AREVA LFAs using SCE NRC approved methodology (see Section 3.1 of SCE-9801-P-A. "Reload Analysis Methodology for the San Onofre Nuclear Generating Station, Units 2 and 3").

Core physics modeling of the AREVA LFAs will be performed in the same manner as the co-resident fuel. Two–group neutronic cross-section data will be generated for the specific LFA fuel enrichment and burnable poison loadings. These cross-section data will be used in two-group coarse-mesh (lumped multiple fuel rods) and fine-mesh (individual fuel rods) core simulation depletions to obtain flux, power and burnup distributions at representative time-points throughout the cycle. At each time-point, the peak integrated radial power peaking factor in the LFAs will be verified to be 0.95 or less of the core maximum integrated radial power peaking factor (Fr).

RAI 1, Part (3) Response

Consistent with Commitments 4 and 5 in Enclosure 2 of the request, for subsequent cycles (Cycles 17 and 18), SCE will place the AREVA LFAs in non-limiting locations and will continue to model AREVA LFAs in the SONGS core physics models. The impact will be analyzed in the cycle 17 and 18 core physics calculations that support the reload analyses. The analysis methodology and process will be consistent with those described in SCE response to RAI 1 Part (2) above.

RAI 2: *AREVA and Westinghouse Design Analyses*

AREVA will be performing detailed design analyses for the LFAs, including thermal-hydraulic compatibility, LOCA and non-LOCA criteria, mechanical design, thermal hydraulic and seismic analyses of the AREVA LFAs in the SONGS reactor cores. Westinghouse will perform a compatibility analysis to ensure that the insertion of AREVA LFAs will not cause the remaining Westinghouse fuel to exceed its operating limits and to ensure there is no adverse impact on the fuel performance or mechanical integrity. The staff needs the following additional information regarding these analyses:

(1) Provide the methodology, data used in, and results expected from AREVA design analyses that include thermal-hydraulic compatibility, LOCA and non-LOCA criteria, mechanical design, thermal hydraulic and seismic analyses.

(2) Provide details of the Westinghouse compatibility analysis including methodology and expected results from analysis. Explain how the results will be interpreted to ensure that the insertion of the AREVA LFAs will not cause the remaining Westinghouse fuel to exceed its operating limits and to ensure there is no adverse impact on the fuel performance or mechanical integrity.

RAI 2, Part (1) Response

The thermal hydraulic analyses are based on hydraulic flow testing performed both on the AREVA and Westinghouse fuel designs for input to the NRC approved thermal hydraulic analysis codes XCOBRA-IIIC and LYNXT. Flow tests were performed on each assembly in the AREVA Richland Test Facility. These flow tests provide the pressure drops for the overall assembly as well as the individual assembly components. These component pressure drop coefficients are then used in the approved thermal hydraulic codes to determine the flow distributions in the different fuel assemblies. These flow distributions are used for the thermal hydraulic compatibility assessment and the non-LOCA transient assessments.

To assure that the Specified Acceptable Fuel Design Limit (SAFDL) for departure from nucleate boiling (DNB) limit was satisfied, flow/pressure/power statepoints were provided to AREVA by SCE for limiting conditions in their reload design analyses. These statepoints were explicitly analyzed by AREVA using the XCOBRA-IIIC code and the NRC approved DNB correlation for the fuel with HTP spacer grids. The 5% power peaking reduction for the AREVA design was not credited in these evaluations. All of the statepoint analyses showed that there was substantial margin to the DNB limits for the AREVA fuel at these specified conditions. SCE evaluations concluded that the thermal hydraulic performance of the AREVA design provides at least 10% additional margin when compared with the co-resident fuel. Therefore, for the non-LOCA accident evaluations, the analyses of record for the co-resident fuel remain bounding and applicable. The LFAs will have margin to the limits resulting from the improved thermal hydraulic performance as well as the 5% reduction in power peaking.

LOCA performance was evaluated by comparing the thermal hydraulic and geometric characteristics of the AREVA design with those of the co-resident fuel. Cladding swelling, rupture and oxidation, Gadolinia content, fuel assembly power, peak cladding temperatures, small break LOCA, coolable geometry, and long-term cooling were considered. The U-235 enrichment of the Gadolinia bearing rods was reduced more than required by the AREVA guidelines, assuring that these rods would not be limiting. Based on these assessments, the differences between the two fuel designs were determined to be minimal and easily offset by the 5% reduction in power peaking. Therefore, the LFAs are non-limiting with respect to the co-resident fuel and are covered by the reload analysis of record, which demonstrates compliance with the 10 CFR 50.46(b) criteria.

The mechanical design was evaluated using AREVA's NRC approved mechanical design methods and NRC approved mechanical design criteria. These evaluations and criteria address the NUREG-0800 Standard Review Plan, Chapter 4 SAFDL evaluations. Power histories were provided by SCE for these analyses. The histories were created to assess the limiting power rods in the core for the first, second, and third cycle of operation, and were extended to the AREVA licensed burnup limit of 62 GWd/kgU peak rod although the LFAs will not exceed the SONGS licensed burnup limit of 60 GWd/kgU. The 5% power peaking reduction restriction was not included in these assessments. The results of these evaluations show that, even without considering the 5% reduction, all of the approved mechanical design criteria are met for the AREVA fuel for the design lifetime of the AREVA fuel.

The analysis of the seismic performance of the fuel used the NRC approved methods to determine the seismic response of the AREVA and Westinghouse fuel for both the as-designed core and the all-AREVA core conditions. The methodology is a finite-element analysis to determine the displacements and resulting loads on the fuel assembly and fuel assembly components. Fuel assembly test data from CE16x16 test assemblies, adjusted to the SONGS fuel assembly design were used as input. SCE provided the core plate time histories as well as co-resident fuel information to AREVA for these analyses. The analyses demonstrate that the ASME design criteria, as identified in Chapter 4.2 of the Standard Review Plan, are met for AREVA LFAs in the Westinghouse core.

<u>RAI 2, Part (2) Response</u>

SCE provided Westinghouse a set of pressure drop profiles for both AREVA and the current Westinghouse (CE16STD) fuel bundles based on tests performed at AREVA's pressure drop test facility, and a set of mechanical design parameters for AREVA LFAs. Based on these data, the following calculations will be performed:

A. Estimation of grid loss coefficients for AREVA LFAs.
B. Impact of AREVA LFAs on core inlet flow distribution.

C. Cross flow and assembly bow force calculations.

The crossflow velocities associated with only the 8 AREVA LFAs are expected to be within the experience base with the uniform core of Westinghouse fuel. Effects of the AREVA LFAs are addressed in the evaluations of fuel performance or mechanical integrity as part of the reload evaluation, in order to confirm that the Westinghouse fuel will not exceed the operating limits.

D. Mechanical Design Compatibility

The mechanical design portion of the fuel compatibility study includes two major topics: general compatibility of the Westinghouse and AREVA fuel assembly designs, and effects on the seismic/LOCA performance of the Westinghouse fuel and CEAs residing in the Westinghouse assemblies.

The general fuel assembly compatibility portion of the study primarily involves dimensional comparisons of the fuel assembly designs to assess whether the presence of the AREVA LFAs could have a negative impact on the adjacent Westinghouse assemblies. In addition to dimensional considerations, the mechanical design study will evaluate the effect that the differences in the pressure drop of the two fuel assembly designs could have on the mechanical design performance of the Westinghouse fuel. Parameters that could be affected by the pressure drop include bundle growth, holddown margin, and grid-to-rod fretting. Based on preliminary results and results of a similar analysis performed for AREVA LFAs that were being inserted in a different CE-NSSS reactor, it is anticipated that the presence of the AREVA LFAs will not have a negative impact on the fuel performance or mechanical integrity of the Westinghouse fuel.

The portion of the compatibility study related to the effect of the AREVA LFA and Westinghouse Fuel Assembly loading pattern configuration in the SONGS Unit 2 Cycle 16 core on the seismic/LOCA performance of the Westinghouse fuel is being done in accordance with the licensed methodology and criteria of CENPD-178. As with the general compatibility study, based upon preliminary results, it is anticipated that the presence of AREVA LFAs does not have a significant effect on the co-resident Westinghouse fuel.

<u>RAI 3:</u> Poolside LFA Examinations

Provide details of the poolside LFA examinations to assess key performance measures of the lead fuel assembly. List methods and procedures for each of the examinations, inspections and measurements that are required to be done during the post-irradiation examination to obtain sufficient data to substantiate in-reactor fuel performance behavior such as; oxidation behavior, hydriding behavior, fretting and diameter measurements, assembly and fuel rod growth, assembly and channel bow, and guide tube wear measurements.

<u>RAI 3 Response</u>

As described in the exemption request, the 10CFR 50.46 exemption is being submitted to allow evaluation of AREVA M5 fuel under SONGS flow condition in order to eliminate grid to rod fretting induced fuel failures. Unlike traditional high burnup LTAs, the SONGS LFAs will use approved designs, and will be operated in non-limiting locations, within the existing SONGS burnup limit, and within the licensed burnup limit for M5 cladding.

The planned core locations for initial 8 LFAs are as follows:

1st cycle: 4 on core periphery , 4 in core center region (for grid spring relaxation).

2nd cycle: all 8 on core periphery

3rd cycle (if necessary): all 8 on core periphery

This core design strategy maximizes the LFA time in the high crossflow areas on the core periphery susceptible to grid to rod fretting wear.

The LFA key parameter of interest is the resistance to grid to rod fretting (GTRF) under SONGS flow conditions. As such, the planned poolside evaluations are focused on evaluating the fretting performance. GTRF wear will be examined through four-face visual examinations and if necessary, oxide/crud liftoff measurements, fretting and diameter measurements, shoulder gap, assembly length and guide tube measurements.

The AREVA LFAs will be operated in non-limiting locations, and will be operated below the licensed SONGS burnup limit. Therefore, other characteristic exams for rod/assembly growth, oxidation behavior, channel bow, etc., are not planned since the SONGS LFAs are all within the NRC approved models for the AREVA CE-HTP fuel with M5 cladding.

RAI 4: *Generic Letter 83-11, Supplement 1: Licensee qualification for performing safety analyses, Attachment 1: Guidelines for qualifying licensees to use generically approved analysis methods.*

This NRC Generic Letter (GL) was issued to notify licensees and applicants of modifications to the NRR practice regarding licensee qualification for performing their own safety analyses. This includes the analytical areas of reactor physics, core thermal hydraulic analysis, transient analysis (non-LOCA), and COLR parameter generation. The agency encourages utilities to perform their own safety analyses since doing this will significantly improve the licensee's understanding of plant behavior. Attachment 1 of GL 83-11, Section 2.4, "Comparison Calculations," stipulates that licensees should verify their ability to use the methods by comparing their calculated results to an appropriate set of benchmark data, such as physics startup tests, measured flux detector data during an operating cycle, higher order codes, published numerical benchmarks, analyses of record, etc. (emphasis added). These comparisons should be documented in a report which is part of the licensee's quality assurance (QA) records. Significant, unexpected, or unusual deviations in the calculations of safety-related parameters should be justified in the report. All comparisons with startup test data should agree within the acceptance criteria defined in the plant startup test plan.

(1) In accordance with the Generic Letter 83-11 and its Attachment 1, comparisons of calculated results from an adopted code with benchmark data, such as experimental data, analysis of record, and/or data from measurements such as plant startup/low power physics tests, should be documented. Tables 4.1 through 4.13 list various physics parameters from CASMO-4/SIMULATE-3 and compare them with the respective measured values. Please revise these Tables with values from CASMO-3/SIMULATE-3 methodology. The revised Tables must include comparisons of both CASMO-4/SIMULATE-3 values with the measured data and comparison of CASMO-3/SIMULATE-3 values with their respective measured values and calculate the percentage difference between the two sets of data.

(2) Guidelines listed in Attachment 1 of GL-83-11 stipulate that "licensees should verify their ability to use the methods by comparing their calculated results to an appropriate set of benchmark data..." Section 6 of Enclosure 4 of the LAR describes the benchmark results from KRITZ-3, KRITZ-4, and B&W critical experiments for gadolinia burnable absorber.

(i) Describe how SCE has applied the measurements and results from the above mentioned criticals to the proposed SONGS core with gadolinia burnable absorber.

(ii) SCE should provide details of how to benchmark the CASMO-4/SIMULATE-3 methodology against B&W critical data for the use of gadolinia bearing fuel at SONGS with respect to fuel pin power and associated uncertainties for UO_2 and gadolinia pin power.

<u>RAI 4, Part (1) Response</u>

Tables 4.1 through 4.13 provide comparisons of CASMO-4/SIMULATE-3 calculation of physics parameters with plant measurements consistent with GL 83-11. For each parameter, the CASMO-4/SIMULATE-3 calculated values (c) were directly compared with plant measurements (m). The observed differences (m – c) were used to determine the bias, standard deviation, and 95/95 uncertainty using the statistical method described in the SCE submittal and clarified in SCE response to RAI 5. This benchmarking process doesn't involve CASMO-3. This process is consistent with the Generic Letter 83-11 expectation that "licensees should verify their ability to use the methods by comparing their calculated results to an appropriate set of benchmark data, such as physics startup tests...." Specifically, the proposed code (CASMO-4/SIMULATE-3) results were compared with data from measurements in Tables 4.1 through 4.13. Therefore, CASMO-3/SIMULATE-3 results are not presented in these tables. Parameters included in these tables are, Critical Boron Concentration (CBC), Isothermal Temperature Coefficient (ITC), Power Coefficient (PC), and Control Rod Worth. In addition, Table 4.15 presents the comparison for Inverse Boron Worth (IBW). The biases and 95/95 uncertainties for CASMO-4/SIMULATE-3 are similar to the values developed for CASMO-3/SIMULATE-3 in Reference 1 of Section 8 of the SCE topical.

Table 1 compares a sample set of CASMO-4/SIMULATE-3 results with CASMO-3/SIMULATE-3. In general, the CASMO-4/SIMULATE-3 results are consistent with the CASMO-3/SIMULATE-3 results and both agree well with the measurements. Comparisons are all within the test criteria provided in ANSI/ANS-19.6.1-2005 (Reference 1) for reload startup physics tests.

Table 1. Comparison of CASMO-3/SIMULATE-3 and CASMO-4/SIMULATE-3 with Measurements

Parameter	Unit/Cycle	Condition	Units*	Measured	C-3/S-3	C-4/S-3	Test Criteria
CBC	S2C15	HZP	ppm	2148	2109	2139	±50 ppm
CBC	S3C14	HZP	ppm	2093	2060	2091	±50 ppm
ITC	S2C15	HZP	pcm/ °F	-1.16	0.17	-0.84	±2 pcm/ °F
ITC	S3C14	HZP	pcm/ °F	-1.72	-0.54	-1.44	±2 pcm/ °F
ITC	S3C4	HFP	pcm/ °F	-8.23	-7.61	-7.38	N/A
PC	S3C1	HFP	pcm/ % power	-8.93	-8.93	-9.25	N/A
IBW	S2C14	HZP	ppm/% $\Delta\rho$	138.7	141.8	143.1	N/A
IBW	S3C14	HZP	ppm/% $\Delta\rho$	132	144.0	144.9	N/A
Control Rod Worth							
CEA Group 6	S2C14	HZP	pcm	385	383	403	±100 pcm
CEA Group 5	S2C14	HZP	pcm	356	329	340	±100 pcm
CEA Group 4	S2C14	HZP	pcm	709	696	727	±106 pcm
CEA Group 3	S2C14	HZP	pcm	786	729	717	±118 pcm
CEA Group 1	S2C14	HZP	pcm	708	692	711	±106 pcm
CEA Group A	S2C14	HZP	pcm	1414	1344	1356	±212 pcm

*ppm: parts per million, pcm: 1x10-5 reactivity unit.

RAI 4, Part (2)(i) Response

Section 6 of the SCE topical references the CASMO-4 developer's benchmarking of the capabilities of CASMO-4 in accurately predicting the k-effective and fuel pin fission rates for fuel assembly geometries containing various types of burnable absorbers including gadolinia and other burnable absorbers used in the industry. A good agreement in fission rates confirms CASMO-4 capability in predicting local pin peaking. The benchmark problems discussed are KRITZ-3, KRITZ-4, B&W 1810 series criticals, and MCNP (Monte Carlo N-Particle Transport Code) calculations. These problems encompass a wide range of fuel designs including BWR, PWR and Mixed Oxide (MOX) fuels. The B&W 1810 series criticals are also included in the validation of the local pin power reconstruction capabilities of SIMULATE-3 discussed in Section 5.1 of the SCE topical.

The benchmarking results validate the CASMO-4 capability in predicting reactivity. For example, k-effective values from the B&W 1810 criticals, which have the most realistic configuration to represent PWR fuel, have a narrow spread of 0.99952 to 1.00208. It is also noteworthy that the cores with gadolinia (V, XIV, and XX) are bounded by the base CE core with no absorbers. This confirms the CASMO-4 capability to accurately predict the reactivity for various cores with burnable absorbers including gadolinia. It is noteworthy that, in all cases, the fuel pin fission rate Root-Mean-Square (RMS) values are within 1.3%, which is smaller than the local pin power uncertainty of 1.78%, established in Section 5.1 of the SCE topical. The 1.78% local pin power uncertainty value is bounded by the 2% value SCE used for CASMO-3/SIMULATE-3 (Reference 1 of Section 8 of the SCE topical).

The good agreement described above is consistent with the CASMO-4 technology improvements described in Section 2 of the SCE topical, namely, heterogeneous geometry, and method of characteristics. In addition, CASMO-4 incorporates the microscopic depletion of burnable absorbers (e.g., gadolinia) directly into the main calculations, and the MICBURN-3 auxiliary code needed by CASMO-3 for gadolinia is no longer required. From the user perspective, there are no special requirements for modeling fuel rods in CASMO-4 with gadolinia burnable absorber. The cross-section generation process with CASMO-4 has also been automated such that all the requisite nuclear data for SIMULATE-3 can be generated in one execution (for one segment). This automated case matrix feature of CASMO-4 results in reduced potential for human error that may arise from the manual construction of the case matrix as was required with CASMO-3.

To demonstrate SCE proficiency in CASMO-4 modeling of fuel with gadolinia, SCE performed benchmarking of B&W Core XIV. As shown in Figure 1 of this RAI response, the SCE pin-by-pin power results agree very well with measurements. The root-mean-square value of the differences is 0.75%, which is well within the 1.78% local pin power uncertainty established above.

Based on the descriptions above, it is concluded that CASMO-4 can also accurately model the SONGS core with gadolinia burnable absorber within the uncertainties established in Table 1-1 of the SCE topical.

RAI 4, Part (2)(ii) Response

Section 5.1 of the SCE topical establishes the local pin power uncertainty of 1.78% based on the work by an industry peer (Reference 2). SCE did not reperform the work for this topical because a subset of the work was performed by SCE as part of the CASMO-3/SIMULATE-3 benchmarking in Reference 1 of Section 8 of the SCE topical. In addition, due to the technology improvement incorporated in CASMO-4 as described in Section 2 of the SCE topical and summarized in SCE response to RAI 4(1), modeling with CASMO-4 is similar and in some cases simpler than with CASMO-3. The 1.78% uncertainty includes the results of the CASMO-4/SIMULATE-3 benchmarking of fuel pin power distributions for B&W criticals, including Cores I, V, XII, XIV, XVIII, and XX. Among these cases, Core V, XIV, and XX contain gadolinia. SCE previously performed Cores I, XII and XVIII as part of CASMO-3/SIMULATE-3 benchmarking.

In response to this RAI, SCE has performed B&W cores XIV and XVIII and compared the reaction rates with measurements and predictions in Reference 2. As shown in Figures 1 and 2, SCE results agree well with results from Reference 2 with insignificant pin power differences in third decimal places. Figure 2 also includes the SCE CASMO-3/SIMULATE-3 results included in the SCE CASMO-3/SIMULATE-3 topical. The good agreement between Reference 2 and SCE results presented in these figures confirmed the fuel pin power uncertainties provided in Table 1-1 of the SCE topical.

The B&W core XIV contains 3920 low enriched (2.46 w/o) and 860 high enriched (4.02 w/o) fuel pins, along with 28 fuel pins at 1.94 w/o enrichment containing 4 w/o of gadolinia. The fuel pins are supported by the top and bottom grid plates in a square matrix. The fuel pin pitch is 0.644 inches. In the SCE model, the fuel pins are grouped along imaginary lines into "assemblies" of 15x15 pitch. The center assembly contains 12 fuel pins with gadolinia, 196 UO2 fuel pins, 16 water holes and one instrument tube. Once the fuel assembly types are defined, CASMO-4 is run for each of the fuel assembly types and the reflector to generate cross-section data for SIMULATE-3. The SIMULATE-3 model consists of ¼ of the core with 4 radial nodes per assembly. Following the execution of the SIMULATE-3 case, the pin-by-pin power distribution for the center assembly is extracted for comparison with the measurements. The results for Core XIV are shown in Figure 1, along with results from Reference 2.

Figure 1. Pin Power Comparison for B&W Core XIV 15x15, 12 Pins with Gadolinia.

0.000 0.000 0.000	1.091 1.091 1.089	0.992 0.997 0.993	0.162 0.159 0.159	0.976 0.983 0.981	1.057 1.052 1.053	1.038 1.042 1.043	1.028 1.023 1.024
	1.080 1.086 1.082	1.118 1.112 1.110	1.000 1.006 1.002	1.054 1.054 1.052	1.158 1.145 1.146	1.091 1.076 1.076	1.028 1.030 1.030
		0.000 0.000 0.000	1.072 1.084 1.082	1.138 1.146 1.145	0.000 0.000 0.000	1.140 1.145 1.147	1.038 1.035 1.035
			0.164 0.161 0.161	1.114 1.126 1.125	1.151 1.171 1.172	1.059 1.069 1.070	1.013 1.019 1.020
				0.000 0.000 0.000	1.080 1.076 1.076	1.011 1.006 1.005	1.003 0.995 0.996
					0.162 0.159 0.158	0.942 0.938 0.938	0.976 0.973 0.975
						0.965 0.958 0.959	0.978 0.964 0.966
							0.959 0.952 0.954

m APS SCE	m: measurements, B&W Core XIV APS: Arizona Public Service (Palo Verde) results SCE: SCE CASMO-4/SIMULATE-3 results

Figure 2. Pin Power Comparison for B&W Core XVIII CE 16x16, No Absorbers.

0.000	1.205	1.033	0.997	0.977	0.959	0.941	0.909
0.000	1.208	1.033	1.007	0.990	0.958	0.939	0.919
0.000	1.209	1.036	1.001	0.984	0.965	0.943	0.918
0.000	1.217	1.035	1.002	0.988	0.970	0.948	0.922
	1.076	1.021	1.012	1.010	0.982	0.946	0.912
	1.088	1.042	1.031	1.015	0.986	0.945	0.912
	1.086	1.034	1.032	1.017	0.980	0.947	0.916
	1.082	1.033	1.028	1.014	0.983	0.951	0.920
		1.065	1.228	1.203	1.043	0.957	0.928
		1.092	1.215	1.200	1.048	0.961	0.909
		1.086	1.219	1.204	1.043	0.955	0.914
		1.079	1.211	1.197	1.038	0.958	0.919
			0.000	0.000	1.183	0.974	0.924
			0.000	0.000	1.171	0.967	0.915
			0.000	0.000	1.175	0.969	0.910
			0.000	0.000	1.168	0.966	0.913
				0.000	1.170	0.970	0.909
				0.000	1.157	0.955	0.903
				0.000	1.161	0.957	0.899
				0.000	1.154	0.953	0.901
					0.995	0.924	0.886
					1.010	0.925	0.875
					1.004	0.920	0.881
					0.999	0.922	0.883
						0.893	0.866
						0.887	0.856
						0.888	0.860
						0.890	0.862
							0.833
							0.837
							0.837
							0.838

Measured	m: measurements, B&W Core XVIII
APS	APS: Arizona Public Service (Palo Verde) results
SCE	SCE: SCE CASMO-4/SIMULATE-3 results
C3/S3	C3/S3: SCE CASMO-3/SIMULATE-3 results

References:

1. American National Standard, ANSI/ANS-19.6.1-2005, "Reload Startup Physics Tests for Pressurized Water Reactors."

2. NRC Letter To Arizona Public Service Company, March 20, 2001, SUBJECT: PALO VERDE NUCLEAR GENERATING STATION (PVNGS), UNITS 1,2, AND 3- ISSUANCE OF AMENDMENTS ON CASMO-4/SIMULATE-3 (TAC NOS. MA9279, MA9280, AND MA9281)

RAI 5: Statistical Analysis: Biases and Uncertainties

Table 1.1 of Enclosure 4 of Reference 1 lists biases and uncertainties in parameters calculated using CASMO-4/SIMULATE-3.

Provide detailed statistical analyses methodology for biases, uncertainty and tolerance values for predicted (calculated) and measured values for parameters listed in Table 1.1 of Enclosure 4 of Reference 1. The parameters include, but are not limited to, reactivity, power coefficients, control rod worths, and various assembly and pin peaking factors.

RAI 5 Response

For each of the parameters with direct plant measurements available (e.g., boron concentration), the sample mean, standard deviation, and/or the root-mean-square (RMS) of the observed differences were calculated using standard statistical techniques, including the ANSI standard normality tests (Ref 1). Based on the sample mean, the standard deviation (S), and the sample size, a set of conservative one-sided 95/95 tolerance limits (bias ± reliability factor) was calculated using the methods of Reference 2. The reliability factor, or 95/95 uncertainty is calculated as:

$$\text{Reliability Factor (95/95 Uncertainty)} = K_{95/95} * S$$

$K_{95/95}$ is the critical factor obtained from Section 2.4, (values of k for $f = n - 1$ and $\gamma = .95$) of Reference 2 for the sample size of the comparison. The appropriate critical factor is from the column with $P = .95$.

The tolerance limits are such that, when applied to the CASMO-4/SIMULATE-3 results, there is a 95 percent probability, with 95 percent confidence that the calculated values will conservatively bound the true values.

As shown in Table 1-1 of the submittal, for application to safety analyses, the CASMO-4/SIMULATE-3 calculated values will be conservatively adjusted as follows:

For those parameters with differences expressed in relative units:

$$\text{Predicted Value} = \text{Calculated Value} * [1.0 + (\text{Bias} \pm 95/95\ \text{Uncertainty})/100\%]$$

For those parameters with differences expressed in absolute units:

$$\text{Predicted Value} = \text{Calculated Value} + (\text{Bias} \pm 95/95\ \text{Uncertainty})$$

As an example, the following details the application of the aforementioned statistical method for the calculation of Critical Boron Concentration (CBC) bias and uncertainty.

The SCE CASMO-4/SIMULATE-3 model calculations of the Critical Boron concentrations (CBC) were compared with zero-power startup measurements and full power operating data for several cycles of SONGS-2 and SONGS-3.

Differences between calculated and measured boron ppm data are stated in absolute terms, measured minus calculated (m - c). The statistics from the zero power comparison quantify the model accuracy for predicting the CBC for beginning-of-cycle (BOC), xenon-free conditions. Measurements from eleven cycles of startup tests were included.

The full power comparison results are used as conservative estimates of the model uncertainty for all equilibrium power conditions with thermal feedback. A total of 111 measurements from six operating cycles were used.

Statistical analysis was performed on the measured versus CASMO-4/SIMULATE-3 calculated CBC differences for zero power and full power, respectively. First, the sample mean and standard deviation were calculated for CBC. The differences are due to CASMO-4/SIMULATE-3 calculational uncertainties, variations in B-10 isotopic concentrations, and measurement (titration) uncertainties. For conservatism, all differences are assumed due only to CASMO-4/SIMULATE-3 calculational uncertainties.

Second, the sample distributions from zero power and full power comparisons are tested for normality using ANSI Standard N15.15-1974 (Reference 1). The normality test is used since the 95/95 tolerance limit method (Reference 2) assumes that the population has a normal distribution. The test concludes that both zero power and full power distributions are normal. Finally, the bias, 95/95 uncertainty, and one-sided tolerance limit are calculated for zero power and full power, respectively. The calculation is summarized in the table below.

	Zero Power CBC	Full Power CBC
Sample Size	11	111
Mean (bias)	-1 ppm	30 ppm
Standard Deviation (S)	8 ppm	10.3 ppm
Normality Test Result	Normal	Normal
Critical Factor ($K_{95/95}$)	2.815	1.911
95/95 Uncertainty ($K_{95/95}$ * S)	23 ppm	20 ppm
Upper Tolerance Limit (bias + 95/95 Uncertainty)	22 ppm	50 ppm
Lower Tolerance Limit (bias - 95/95 Uncertainty)	-24 ppm	10 ppm

This method was also used in the calculation of biases and 95/95 uncertainties for Inverse Boron Worth (IBW), Power Coefficient (PC), Isothermal Temperature Coefficient (ITC), and Control Rod worths. For Power coefficient, an additional conservatism was applied by using the largest observed difference as the 95/95 uncertainty. This process is consistent with the process used in the SCE CASMO-3/SIMULATE-3 topical.

For parameters related to power peakings, which are inferred from incore detector responses, the approach was to compare the CASMO-4/SIMULATE-3 calculated parameters with CASMO-3/SIMULATE-3 calculations to demonstrate that the two codes (CASMO-4 and CASMO-3) provide consistent results and therefore, the CASMO-3/SIMULATE-3 uncertainties for these parameters remain applicable for CASMO-4/SIMULATE-3.

Derivation of the local pin power uncertainty is discussed in SCE response to RAI 4.

References:

1. ANSI N15.15-1974, "American National Standard: Assessment of the Assumption of Normality (Employing Individual Observed Values)," October 1973.

2. Factors for One-Sided Tolerance Limits and for Variable Sampling Plans, D. B. Owen, Sandia Corporation Monograph, SCR-607, March 1963.

<u>RAI 6</u>: Mixed Core Fuel Design Analysis

With the proposed introduction of AREVA NP fuel assemblies in to the SONGS Unit 2 or Unit 3 core, the AREVA NP fuel assemblies will be co-resident with Westinghouse fuel in the core, which will be considered a mixed core.

(1) *How will the "mixed core" affect the compatibility analysis that will be performed "to ensure that the insertion of the AREVA LFAs will not cause the remaining Westinghouse fuel to exceed its operating limits and to ensure there is no adverse impact on the fuel performance or mechanical integrity?"*

(2) *How will the non-LOCA transient analyses be impacted by the mixed core, particularly the minimum DNBR calculations?*

When a reactor core consists of more than one type of fuel assembly, as in the proposed SONGS Unit 2 or Unit 3 core, the flow redistributions due to pressure drop differences in the fuel assemblies of different types might introduce a DNBR penalty with respect to the reference core consisting of only one fuel type. Discuss the impact, if any, of the mixed core on various thermal margin calculations for the proposed new core.

<u>RAI 6, Part (1) Response</u>

Compatibility analyses will be performed by both AREVA and Westinghouse using their approved analysis methods. These analyses consider the impact of the other vendor's fuel to determine if there are any operational issues with the vendor's fuel as a result of the two fuel types being in the core. Additional discussion is provided in the response to RAI 2.

<u>RAI 6, Part (2) Response</u>

The AREVA Lead Fuel Assemblies are being placed in non-limiting locations. SCE evaluations concluded that the thermal hydraulic performance of the AREVA design provides additional margin when compared with the co-resident fuel. No DNBR penalty is being applied for SONGS-2 Cycle 16 as discussed in the response to RAI 2, since the limited set of 8 LFAs is being explicitly evaluated by both Westinghouse and AREVA for impact.

RAI 7: Statements regarding the use of CASMO-3/SIMULATE-3 Methodology at SCE/SONGS

Clarification is required regarding the date of implementation of CASMO-3/SIMULATE-3 methodology for reactor physics design analyses at SONGS.

SCE obtained NRC approval to use CASMO-3/SIMULATE-3 methodology (SCE-9001-A) for their nuclear design reload analyses for Units 2 and 3 in August 1992.

In June 1999, SCE obtained NRC approval to use ROCS/MC for their nuclear design analysis as part of their reload analysis methodology (SCE-9801-P-A).

San Onofre Updated FSAR (June 2005), Section 4.3.3.5.1 indicates "Specifically, SCE intends to use CASMO-3/SIMULATE-3 programs in licensing applications, including PWR reload physics design, calculating startup predictions, generating physics input for safety analyses, generating core physics data books, and establishing setpoint updates for both reactor protection and monitoring systems."

Current San Onofre Unit-2 and Unit-3 Technical Specification 5.7.1.5 b. CORE OPERATING LIMITS REPORT does not list SCE 9001-P-A as a methodology for analytical methods used to determine the core operating limits. Instead the TS lists only SCE-9801-P-A as "Reload Analysis Methodology for SONGS" which permits SCE to use ROCS/MC methodology.

The LAR for ME0602/3 states (Page 1 of 7 of Enclosure 3) that "The CASMO-3 and SIMULATE-3 methodologies have been used to evaluate reload designs at San Onofre for over fifteen years". I have no way to verify this statement in the light of the above facts.

Please clarify the information regarding the date of implementation and duration of use of CASMO-3/SIMULATE-3 methodology at SONGS. Include the clarification in the final version of appropriate enclosures of the LAR,

RAI 7 Response

In August 1992, SCE obtained NRC approval to use CASMO-3/SIMULATE-3 methodology (SCE-9001-A) for nuclear design reload analyses for SONGS Units 2 and 3 (Reference 1 of Section 8 of the SCE topical). The SER states:

> "Based on the analyses and results presented in the topical report, the staff concludes that the CASMO-3/SIMUALTE-3 methodology as validated by SCE can be applied to steady-state PWR reactor physics calculations for the SONGS units reload applications as discussed in the above technical evaluation. The accuracy of this methodology has been demonstrated to be sufficient for use in licensing applications, including PWR reload physics analysis, generation of safety analysis inputs, startup predictions, core physics databooks, and reactor protection system and monitoring system setpoint updates."

The CASMO-3/SIMULATE-3 methodology is currently used to perform core management, startup predictions, and core physics data books. This use of CASMO-3/SIMULATE-3 methodology is consistent with the approval stated above.

In June 1999, SCE obtained NRC approval of topical report SCE-9801-P-A (Reference 1) authorizing SCE to perform reload design and Non-Loss of Coolant Accident analyses for SONGS Units 2 and 3 using ABB/CE (Asea Brown Boveri / Combustion Engineering, now Westinghouse) reload analysis methodology. Currently the ROCS/MC methodology (Reference 2) interfaces directly with the Non-LOCA safety analyses and Setpoint analyses within the Westinghouse/CE code packages and as such is used in this portion of the reload process. In addition, Section 3.1.1 Note 1 (Physics Analysis) of SCE-9801-P-A states:

> "ROCS/MC may be replaced in the future by SIMULATE which has been approved by the NRC in Reference 26 [which is SCE-9001-A] for use by SCE in licensing applications, including PWR reload physics analysis, generation of safety analysis inputs, startup predictions, core physics databooks, and reactor protection system and monitoring system setpoint updates."

In summary, SCE currently uses both the CASMO-3/SIMULATE-3 methodology and ROCS/MC methodology for reload physics licensing analyses. The CASMO-3/SIMULATE-3 methodology is used in the initial fuel management design and to generate startup predictions and core physics data books. The ROCS/MC methodology interfaces more directly with the NON-LOCA safety analyses and Setpoint analyses within the Westinghouse/CE code packages and as such is used in this portion of the core design. Per this request, SCE intends to replace CASMO-3 with CASMO-4 for physics reload design activities.

References:

1. Letter from S. Dembek (NRC) to H. B. Ray (SCE), "San Onofre Nuclear Generating Stations Units 2 and 3 – Evaluation of Reload Analysis Methodology Technology Transfer (TAC NOS. MA4289 and MA4290)," June 2, 1999.

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

RAI 8: Correction(s) needed

On page 18 of SCE-0901, correct the Clad ID of 0.322 inches to 0.332 inches.

RAI 8 Response

On page 18, the clad ID has been corrected. Three other typographical errors were also corrected on pages 34, 37, and 56. Enclosure 1 contains the correct pages 18, 34, 37, and 56. These pages will be incorporated into the final version of SCE-0901.

RAI 9: Margin of Safety

Page 5 of 7 of Enclosure 3 of Reference 1 states that, "Extensive benchmarking of CASMO-4 has demonstrated that the values of those parameters used in the safety analyses are not significantly changed relative to the values obtained using the CASMO-3 methodology. For any changes in the calculated values that do occur, the application of appropriate biases and uncertainties ensures that the current margin of safety is maintained." Also, this section states that, "The proposed change does not involve a significant reduction in a margin of safety."
Provide details of the calculations to support the above statements regarding margin of safety in the physics parameters from CASMO-4/SIMULATE-3, over those calculated from CASMO-3/SIMULATE-3 methodology.

RAI 9 Response

As indicated in Table 1-1 of the submittal, and further clarified in SCE response to RAI 5, for application to safety analyses, the CASMO-4/SIMULATE-3 calculated values will be conservatively adjusted by the bias and 95/95 uncertainty. The adjustment ensures that there is a 95 percent probability, with 95 percent confidence that the adjusted values will conservatively bound the true values. The application of 95/95 probability/confidence is consistent with the approach in the SCE CASMO-3/SIMULATE-3 topical (Reference 1 of Section 8 of the SCE topical) and Westinghouse (ABB/CE) ROCS topical (Reference 1). Therefore, margin of safety in physics parameters is not affected.

References:

1. ABB/CE "The ROCS and DIT Computer Codes for Nuclear Design," Topical Report CENPD-266-P-A, April 1983.