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Richard J. St. Onge
Director, Nuclear Regulatory Affairs and
Emergency Planning

September 14, 2012

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Docket Nos. 50-361 and 50-362**
Response to Request for Additional Information Regarding License Amendment Request for Permanent Use of AREVA Fuel and for Permanent Exemption to Use M5 Cladding (TAC Nos. ME6820, ME6821, ME6822, AND ME6823)
San Onofre Nuclear Generating Station, Units 2 and 3

Reference: Letter from N. Kalyanam (NRC) to P. T. Dietrich (SCE) dated August 1, 2012; Subject: San Onofre Nuclear Generating Station, Units 2 and 3 – License Amendment Request RE: Use of AREVA Fuel (TAC Nos. ME6820, ME6821, ME6822, AND ME6823)

Dear Sir or Madam:

By letter dated August 1, 2012, the Nuclear Regulatory Commission issued a Request for Additional Information (RAI) regarding use of unrestricted usage of AREVA fuel and permanent exemption to use M5 cladding.

The RAI letter requested a response within 30 days of receipt of the letter. NRC staff agreed by phone on September 4, 2012, that SCE may submit the response by September 14, 2012.

Enclosure 2 of this submittal contains information that is proprietary to SCE or AREVA. SCE requests that this proprietary Enclosure be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). Enclosure 1 provides notarized affidavits from SCE and AREVA which set forth the basis on which the information in Enclosure 2 may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed by paragraph (b)(4) of 10 CFR 2.390. Enclosure 3 provides the non-proprietary version of Enclosure 2.

A002
NRR

P.O. Box 128
San Clemente, CA 92672

There are no new regulatory commitments contained in this letter. If you have any questions or require additional information, please contact Ms. Linda T. Conklin, Licensing Manager, at (949) 368-9443.

Sincerely,

Enclosures:

1. NOTARIZED AFFIDAVITS

<u>Proprietary Enclosures</u>

2. Response to Request for Additional Information (RAI) regarding use of unrestricted usage of AREVA fuel and permanent exemption to use M5 cladding

<u>Non-Proprietary Enclosures</u>

3. Response to Request for Additional Information (RAI) regarding use of unrestricted usage of AREVA fuel and permanent exemption to use M5 cladding

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

ENCLOSURE 1

NOTARIZED AFFIDAVITS

AFFIDAVIT

STATE OF CALIFORNIA)
) ss.
CITY OF SAN CLEMENTE)

1. My name is Owen J. Thomsen. I am employed by Southern California Edison Company ("SCE"). My present capacity is Manager, Nuclear Fuel Management, for the San Onofre Nuclear Generating Station ("SONGS"), and in that capacity I am authorized to execute this Affidavit.

2. SCE is the operating agent for SONGS. I am familiar with the policies established by SCE to determine whether certain SCE information is proprietary and confidential, and to ensure the proper application of these policies.

3. I am familiar with SCE information in the document entitled "San Onofre Nuclear Generating Station, Units 2 and 3, Proposed Permanent Exemption Request and Proposed Change Number (PCN) 600, Amendment Application Numbers 261 and 247, Request for Unrestricted Use of AREVA Fuel," (referred to herein as "Document") submitted to the NRC in July 2011.

4. SCE has classified the information contained in the Document as proprietary and confidential in accordance with SCE's policies.

5. Specifically, SCE applied the following criteria to determine that the information contained in the Document should be classified as proprietary and confidential:

   (a) SCE has a Non-Disclosure Agreement (NDA) with Westinghouse Electric LLC ("Westinghouse") and AREVA NP ("AREVA") (the NDA is referred to as the "Westinghouse-AREVA-SCE NDA"), under which Westinghouse and AREVA have provided to SCE certain proprietary and confidential information contained in the Document.

(b) The information reveals details of Westinghouse's, SCE's, and/or AREVA's research and development plans and programs, or the results of these plans and programs.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive commercial advantage for Westinghouse, SCE, and/or AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive commercial advantage for Westinghouse, SCE, and/or AREVA on product optimization or marketability.

(e) The unauthorized use of the information by one of Westinghouse's, SCE's, and/or AREVA's competitors would permit the offending party to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(f) The information contained in the Document is vital to a competitive commercial advantage held by Westinghouse, SCE, and/or AREVA, would be helpful to their competitors, and would likely cause substantial harm to the competitive position of Westinghouse, SCE, and AREVA.

6. The information contained in the Document is considered proprietary and confidential for the reasons set forth in Paragraph 5. In addition, the information contained in the Document is of the type customarily held in confidence by AREVA, Westinghouse, and SCE, and not made available to the public. Based on my experience in the nuclear industry, I am aware that other companies also regard the type of information contained in the Document as proprietary and confidential.

7. In accordance with the Westinghouse-AREVA-SCE NDA, the Document has been made available to the NRC in confidence, with the request that the information contained in this Document be withheld from public disclosure. The request for withholding the information from public disclosure is made in accordance with 10 CFR 2.390. The information qualifies for withholding from public disclosure under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

8. In accordance with SCE's policies governing the protection and control of proprietary and confidential information, the information contained in the Document has been made available, on a limited basis, to others outside Westinghouse, SCE and AREVA only as required in accordance with the Westinghouse-AREVA-SCE NDA.

9. SCE's policies require that proprietary and confidential information be kept in a secured file or area and distributed on a need-to-know basis. The information contained in the Document has been kept in accordance with these policies.

10. The foregoing statements are true and correct to the best of my knowledge, information, and belief, and if called as a witness I would competently testify thereto. I declare under penalty of perjury under the laws of the State of California that the above is true and correct.

Owen J. Thomsen

State of California County of Orange

Subscribed and sworn to (or affirmed) before me on this 27th day of July, 2011, by Owen J. Thomsen, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Trupti Trivedi

(Seal)

TRUPTI TRIVEDI
Commission # 1936996
Notary Public - California
Orange County
My Comm. Expires Jun 14, 2015

SUBSCRIBED before me this ________

day of ____________________, 2011.

NOTARY PUBLIC, STATE OF CALIFORNIA
MY COMMISSION EXPIRES:
Reg. #:

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
COUNTY OF CAMPBELL)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in a letter from Rick Williamson (AREVA NP) to Owen Thomsen (Southern California Edison) with subject "Proposed Response to RAI on License Amendment Request for Use of AREVA VQP Fuel," FAB12-439, dated September 5, 2012 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in

accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 5th day of September 2012.

Kathleen A. Bennett

Kathleen A. Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2015
Reg. #110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2015

ENCLOSURE 3

Response to Request for Additional Information (RAI) regarding use of unrestricted usage of AREVA fuel and permanent exemption to use M5 cladding (Non-Proprietary)

SOUTHERN CALIFORNIA EDISON

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST FOR PERMANENT USE OF AREVA FUEL

AND FOR PERMANENT EXEMPTION TO USE M5 CLADDING

DOCKET NOS. 50-361 AND 50-362

TAC NOS. ME6820, ME6821, ME6822, AND ME6823

RAI #1

1. Page 3

(a) SCE states that, "The number of fuel assemblies in the initial batch will be between eight fuel assemblies and approximately a half core of fuel assemblies." Please specify the number of AREVA fuel assemblies that are to be inserted into the SONGS core during the next cycle. Also specify which components of the AREVA fuel assemblies will be of M5 alloy material.

(b) It is stated that the "exact reload core fuel management has not been defined" at the time of the LAR submittal. Please provide the details of the core reload management for the next cycle.

RESPONSE:

(a) SCE currently plans to introduce 36 AREVA CE-HTP fuel assemblies into the Unit 3 Cycle 17 core. The AREVA assembly design is described in Enclosure 2 Section 5.1.2 of PCN-600. The fuel assembly components made of M5™ alloy material are [ ].

(b) Due to the unscheduled shutdown of Unit 3 Cycle 16 by a steam generator tube leak, forcing an extended mid-cycle outage, the final core pattern for Unit 3 Cycle 17 has still not been finalized. The 36 AREVA CE-HTP assemblies will be implemented into the pattern with some assemblies placed in limiting or near limiting locations. The core loading patterns referenced in PCN-600 (Enclosure 2 Section 7.1, Figures 7.1.2 and 7.1.3) present potential Unit 3 Cycle 17 and Cycle 18 core designs to demonstrate and exercise all aspects of the methodology associated with transitioning to AREVA fuel. Using the SONGS fuel management guidelines, Cycle 17 was designed as a transition core with half core fresh AREVA fuel and the other half Westinghouse once burned fuel to fully exercise the mixed core process. Cycle 18 was subsequently designed with a full core of AREVA fuel (note the center assembly exception in Section 4.1.3) to reflect the end state AREVA assembly core.

RAI #2

2. Section 3.2.1

   Provide justification for continuing the use of CE methodologies, CENPD-382-P-A and CENPD-275-P-A in support of the fuel Safety Limits (SLs) in SONGS Technical Specification (TS) 2.1.1.2 in the specification of fuel centerline temperature variation with burnup and its adjustments with burnable poison.

RESPONSE:

PCN-600 (Enclosure 2, Section 4.10) provides justification for continuing the use of CE methodologies, CENPD-382-P-A and CENPD-275-P-A in support of the SONGS Units 2 and 3 fuel Safety Limit (SL) 2.1.1.2 in the specification of fuel centerline temperature variation with burnup and its adjustments with burnable poison.

Per PCN-600, the selection of the Westinghouse methodology to represent the fuel melting temperature is based on [

].

Complementing the justification presented in PCN-600 are the findings presented in the NRC Safety Evaluation in support of gadolinia burnable poison Topical Report CENPD-275-P-A, which specifies a fuel centerline temperature adjustment for the gadolinia burnable poison. CENPD-275-P-A (Revision 1-P, Supplement 1-P-A, Section 2.2.1) states that the slightly more conservative correlation for erbia additions may be used for gadolinia additions. The acceptability of the use of the erbia correlation for gadolinia is addressed in the NRC Staff Safety Evaluation for CENPD-275-P, Revision 1-P, Supplement 1-P. The Safety Evaluation acknowledges that the gadolinium and erbium are closely-related rare earth elements, which form oxides of the same structure. Because of the similarities of these oxides, the measured values of specific additions of gadolinia or erbia on urania properties would be similar. The Safety Evaluation states use of the slightly more conservative erbia correlation (as done in the SCE process) as the gadolinia correlation is acceptable.

The measured values of specific additions of gadolinia and erbia on urania properties are similar. The CE methodologies for burnable poison adjustments have been found to be conservative for gadolinia and erbia. The fuel melt correlation was developed using data which is independent of fuel manufacturer. Therefore, it is justified to use CE methodologies, CENPD-382-P-A and CENPD-275-P-A, in support of the fuel Safety Limits in SONGS Technical Specification 2.1.1.2 in the specification of fuel centerline temperature variation with burnup and its adjustments with burnable poison.

RAI #3

3. Section 4.1.3

(a) Explain why there may be a need to retain the center assembly from the old vendor.

(b) Explain why the core will [ ].

RESPONSE:

(a) The SONGS reactor core has an odd number of fuel assemblies (i.e., 217) and thus the center assembly is typically retained for a third cycle of operation or re-inserted from the spent fuel pool. For fuel economics, SCE would like to retain the ability to reinsert the center assembly from the old fuel vendor without having to invoke all of the mixed core processes or requirements: e.g. mixed core compatibility and LOCA analyses.

(b) In the SONGS checkerboard fuel management patterns, the high burnup center assembly is in a low power, high RCS flow location and thus not limiting from a power peaking or thermal hydraulic perspective. Table 1 below shows power peaking from the center assembly for the cycle 17 and 18 patterns described in PCN-600. As seen from the table, the center assembly is not, nor is it likely to be, at or near limiting for SONGS cores. Enclosure 2, Figure 7.2.4 shows the relatively high flow of the center assembly location.

[ ]

Table 1: VQP* Cycles 17 and 18 Center Assembly Power Peaking

Parameter	Cycle 17		Cycle 18	
	BOC	EOC	BOC	EOC
Center Assembly Relative Power	[			
Center Assembly Fr				
Center Assembly Fq				
Core Fr				
Core Fq				]

- VQP is an acronym for the Vendor Qualification Plan, i.e. the PCN-600 submittal for the unrestricted use of AREVA fuel.

RAI #4

4. Section 4.1.4

(a) Provide details of how the two commitments that AREVA made to the NRC in the Topical Report, XN-NF-85-92(P)(A), have been implemented. The commitments are: (i) [

].

(b) Provide details of how the [

]. (i.e., the details of the nuclear design analysis for a typical cycle).

RESPONSE:

(a) AREVA has committed to ensuring that the Gadolinia bearing fuel rod will not be the limiting rod in the core. On a cycle-specific basis, [

]. The commitment made by AREVA to the NRC is embedded in the reload process as an automatic check performed by the fuel rod analysis code. For each batch of fuel analyzed as part of the reload analysis, the code checks to make sure that the maximum gas pressure predicted for the Gadolinia rods is less than that predicted for the UO_2 rods. If this criterion is violated, then the code flags this occurrence as a failure to meet the gas pressure criterion. Typically, the maximum predicted gas pressure for gadolinia rods is less than that predicted for the UO_2 rods. In case this criterion is failed, modifications will have to be made to the core design such that the criterion can be met for the upcoming cycle.

In the SCE fuel management guidelines SCE has adopted the standard enrichment cutback used by AREVA for gadolina assemblies. [

]. By reducing the enrichment in accordance with this formula, the power in the gadolina rod will always be less than the peak power in the assembly and consequently always less that the peak power in the core. As an example for PCN-600 cycle 17 demonstration analysis, the peak enrichment in PCN-600 Figure 7.1.2 is 3.5 w/o U^{235}. In this assembly the 8% gadolinia rod has been to 2.1 w/o U^{235}. In this way the gadolinia rod will always remain non-limiting for power peaking.

(b) Based on the process described in response (a) above, the gadolinia bearing fuel rod will always be [ ] if the fuel enrichment cutback is employed. However, in the situation that SCE does not employ the requisite gadolina enrichment cutback, the following step by step process is employed to ensure that the peak power of any gadolinia rod will not be the peak rod power in the core:

i. The SIMULATE-3 physics model for a reload cycle is depleted in steps of 1 GWD/T from BOC to EOC. SIMULATE-3 calculates a pin-by-pin power distribution at each 1 GWD/T burnup point. The SIMULATE-3 output contains a summary of maximum peaking factors and the assembly in which the maximum occurs.

ii. At each SIMULATE-3 burnup point, a utility computer program (e.g. MCEDIT) produces a pin-by-pin power distribution for every fuel assembly in the reactor core. In each assembly pin-by-pin power picture, the maximum fuel rod power and the maximum fuel rod location in that assembly are identified. Therefore, the location of the peak fuel rod in the core is identified.

iii. Final verification that the peak fuel rod does not contain Gadolinia is done by manual comparison to the enrichment zone pattern for the assembly in which the peak rod occurs.

RAI #5

5. Section 4.2

[

].

RESPONSE:

The SCE TORC computer code [ ] was implemented, validated, and tested in accordance with SCE computer code control procedures. Test cases have been run and the output checked to verify that the correlation has been implemented correctly and to ensure that the installation of the CHF correlation has not affected the existing code capabilities, analyses or output. The results have been validated by comparison to [

]

The SCE formal software update process defined in Section 4.5.2 of Reference 8.4 of PCN-600 was applied to [

]

The TORC modeling scheme at SONGS as discussed in SCE-9801-P-A [

].

During a review of calculations performed in support of PCN-600, [

]

RAI #6

6. Section 4.2.1

   Provide typical calculations where the system parameter uncertainties and state parameter uncertainties are statistically combined to obtain the minimum DNBR limit [ ]. Provide a list of all parameters with uncertainties that are used in the calculation which leads to the minimum DNBR limit.

RESPONSE:

The SONGS design Specified Acceptable Fuel Design DNBR limit (SAFDL DNBR) for SONGS is 1.31 (Technical Specification LCO 2.1.1.1). This SAFDL DNBR Limit was calculated based on the NRC-approved Modified Statistical Combination of Uncertainties (MSCU) methods.

[ ]

The MSCU process considers two groups of uncertainties: [ ]

The first group, [ ], includes the following uncertainties: [ ]

The second group, [

.]

[

.] The remaining uncertainties are potentially impacted and are addressed below.

Engineering & Systematic Factors Uncertainties

[

]. The results are compared to the existing SONGS values in Table 4.2.

Table 4.2

System Parameter Uncertainties Used As Inputs in SONGS MSCU Analysis [

.]

The standard reload process at SCE does not involve recalculating the DNBR limit each cycle. Instead, the 1.31 DNBR limit is maintained and verified as conservative for each cycle. This is done by evaluating the DNBR change for a given perturbation of input parameters. In the MSCU AOR, the DNBR limit is dominated by the DNBR changes primarily due to perturbation of two factors:[

]

In the normal SCE reload process, the assemblies considered DNBR limiting for the specific cycle are put through the same perturbation as was done in the AOR for Factors #1 and #2 above. If the DNBR change for the cycle specific cases are less than the DNBR change for the equivalent AOR cases, then the cycle specific DNBR limit would be bounded by the AOR limit of 1.31 DNBR.

[

]

To demonstrate the process, an MSCU analysis was performed for the PCN-600 cycle 17 mixed core. The resultant probability distribution function (pdf) for both a CE16 (CE-1) limiting assembly and an AREVA (BHTP) limiting assembly versus the AOR (1.31 DNBR limit is shown in Figure 6-1.

To provide easier comparison, Figure 6-2 shows the pdfs all shifted to the same mean. [

]

Figure 6-1

Figure 6-2

RAI #7

7. Section 4.2.2

Section 4.2.2 states that [

] This section makes reference to Westinghouse methodology, AREVA methodology, and SCE methodology. Table 4.2 lists the SONGS rod bow penalty for AREVA fuel. Does this mean that the AREVA fuel type is the most limiting based on the thermal-hydraulics analysis? The staff would like the licensee to clearly provide the details about which fuel type is most limiting, the appropriate methodology used to calculate the rod bow penalty, and how the penalty is applied [ ].

RESPONSE:

The SCE thermal-hydraulic analysis process considers ALL assemblies in the core to determine which assemblies are potentially DNBR limiting. A wide range of operating conditions and power profiles are considered. Therefore, depending on the specific core design, it may be possible for a Westinghouse fuel assembly to be DNBR limiting at one condition and an AREVA fuel assembly to be DNBR limiting at some other condition during the cycle.

The SCE methodology provides for the rod bow penalty [

]

RAI #8

8. Section 4.3.2

The licensee states that [

]

(a) Due to the fact that [ ] please explain how the results will be consistent.

(b) The first paragraph of Section 4.3.2 states, "Per Reference 8.34 (CEN 193(B) Supplement 2-P), currently SCE uses FATES3B to provide predictions of the steady state response of fuel rods, and to model internal conditions of the fuel rods within the core from insertion to discharge. With the appropriate modeling of mechanical design data, power levels, and power distributions, these [ ] This statement appears to conflict with the last paragraph on page 28, which states, [ ] Please clarify the apparently ambiguous or conflicting statements.

RESPONSE:

(a) Although the fuel performance data used in the fuel mechanical design, LOCA, non-LOCA and setpoints analyses are not originated by the same code or method, these analyses are originated using codes and methods that have been NRC approved for their intended purposes.

[

.]

There is no inconsistency introduced into the reload analysis effort because [

] In all cases, regardless of how the fuel performance data is generated, the reload analysis effort will be performed using codes and methods that have been NRC approved for their intended purposes.

(b) The first paragraph of Section 4.3.2 was intended to describe the current SCE reload process (i.e. before PCN-600 submittal). Subsequent paragraphs of Section 4.3.2 describe the division of fuel rod behavior scope between SCE and AREVA as it relates to the proposed reload licensing applications. As noted in the fourth paragraph of Section 4.3.2, AREVA will be performing all fuel mechanical design and LOCA analyses, including the fuel rod initial conditions for the AREVA LOCA analyses, and SCE will be generating fuel rod behavior analysis data to support the non-LOCA safety analyses and calculations that support SCE setpoints analyses.

RAI #9

9. Section 4.3.3

The licensee has used the CE/Westinghouse legacy code, FATES3B for their fuel rod behavior analyses for generating input to non-LOCA transient and setpoint analyses. Specifically, the FATES3B code has been used to model [

]

Provide the details of the results from the [

]

RESPONSE:

The FATES3B code [

] is used to generate input to non-LOCA transient and setpoints analyses.
[

]

AREVA developed M5™ cladding materials and correlations that were approved by the NRC in AREVA M5™ topical report BAW-10240(P)-A. Similarly, Westinghouse developed ZIRLO™ cladding materials and correlations that were approved by the NRC in Westinghouse ZIRLO™ Topical Report CENPD-404-P-A. [

]

The FATES3B code is used for thermal performance evaluations under normal operation considering steady-state and anticipated transient conditions. Per ZIRLO™ Topical Report CENPD-404-P-A (response to Question 7),[

]

As discussed in PCN-600 (Enclosure 2, Section 4.3.3.2), verification and validation (V&V) testing of [ ] was performed through a combination of code modification review and test case evaluations. [ ] All reference material property data, models, assumptions and required modifications were also reviewed and checked for correct implementation.

As discussed in PCN-600 (Enclosure 2, Section 4.3.3.2), detailed reviews were conducted to ensure the accuracy of [.]

Additionally, [ ]

The fuel temperature, power-to-centerline melt, and rod internal pressure history [ ] are shown in PCN-600 Figures 4.3.6, 4.3.7 and 4.3.8, respectively. [ ]

The use of the [ ]

RAI #10

10. Fuel Thermal Conductivity (Section 5.1.4)

An outstanding issue related to the mechanical and material design of UO2 fuel is the thermal conductivity of irradiated UO2 fuel considering the effects of burnup. The thermal conductivity of irradiated UO2 fuel is affected by changes that take place in the fuel during irradiation: solid fission product buildup (both in solution and as precipitates), porosity and fission gas-bubble formation.

NRC Information Notice 2009-23 dated October 8, 2009, notified licensees of nuclear power reactors of the thermal conductivity degradation (TCD) of uranium fuel pellets with increasing burnup. The significance of this effect was not included in the fuel thermal-mechanical performance codes approved prior to 1999.

NRC Information Notice 2011-21 notified the licensees of the impact of irradiation on fuel thermal conductivity and its potential to cause errors (specifically, in predicted peak clad temperature) in realistic ECCS evaluation models.

10 CFR Part 50, Appendix K, "ECCS Evaluation Models", Section I.A.1, stipulates that, "The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.) To accomplish this, the thermal conductivity of the UO2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep."

The CE/Westinghouse FATES3B code has been used for the evaluation of fuel thermal-mechanical performance at SONGS Units 2 and 3. [.]

Thermal conductivity of UO2 fuel degrades with burnup, and as such, the staff believes that each fuel vendor must have an explicit model to generate burnup dependent fuel thermal conductivity in their analyses to simulate transients and accidents.

(a) Explain how the licensee applied the TCD with burnup in the FATES3B code for the fuel performance evaluation, addressing factors such as, fission gas release, power-to-melt evaluation, and clad strain and fatigue. Please provide details of the fuel temperature calculations that are dependent on the effects of burnup as described above.

(b) Explain how the impact of TCD with burnup has been addressed in the analyses of non-LOCA transients and postulated accidents, specifically but not limited to, the spectrum of control rod ejection accident analyses.

RESPONSE:

a) [

] the selected applications of the FATES3B code are not significantly impacted by the effects of TCD. As discussed in PCN-600 Enclosure 2, Section 4.3.2, AREVA will be performing all fuel mechanical design and LOCA analyses, including the fuel rod initial conditions for the AREVA LOCA analyses, based on AREVA's approved computer codes RODEX2 & RODEX3A. SCE will be generating fuel rod behavior analysis data to support the non-LOCA safety analyses (i.e., CEA Ejection analysis) and calculations that support SCE setpoints analyses.

Effect of TCD on Fission Gas Release for Rod Internal Pressure Calculations (i.e., no-clad liftoff)

The FATES3B code is not used in the AREVA no-clad liftoff calculations, which will be done by AREVA based on their approved RODEX2 computer code.

Effect of TCD on Power to Fuel Centerline Melt

The FATES3B code [

] conservatism in the fuel behavior analysis results is accomplished by the means described in the NRC Question 3.A response as presented in CEN-193(B)-P Supplement 2-P. The response states that the code results [

] it is acceptable to not consider the effects of TCD on power to fuel centerline melt.

Effect of TCD on Clad Strain and Fatigue

The FATES3B code is not used in the clad strain and fatigue analyses of AREVA fuel, which will be performed by AREVA using their RODEX2 computer code as part of their fuel rod mechanical design.

b) Effect of TCD on Non-LOCA Transient Analyses

SCE will use FATES3B [
] for input to the transient analyses, and [
] the CEA Ejection transient analysis.

Depending on the non-LOCA transient being evaluated, [

[ ]

]

The CEA ejection transient analyst selects [

]

Therefore, it is acceptable to [

] At BOL, the onset of TCD has not yet materialized. Therefore, TCD does not affect the CEA ejection transient analysis [

]

RAI #11

11. The NRC staff intends to run FRAPCON-3.4 (Reference 2) benchmark calculations of the resident CE 16x16 fuel rod design and the new AREVA HTP fuel rod design. Please provide the following input for both co-resident fuels at SONGS, Units 2 and 3.

   A. Rod Power History, KW/ft as a function of GWd/MTU

      1. Bounding thermal-mechanical operating envelope (e.g., radial falloff curve)
      2. Discuss any application of rod power uncertainties
      3. Include power histories for different pellet designs (UO_2, Gadolinium).

   B. Axial Power Distribution (Fz at each axial node)

      1. Include axial power distributions (AXPDs) for different axial blanket configurations.

   C. Fuel Rod Design Specifications and Manufacturing Tolerances

      1. Outer diameter
      2. Inside diameter
      3. Pellet diameter
      4. Stack length
      5. Plenum length
      6. Pellet height
      7. Dish radius
      8. Dish depth
      9. Spring outside diameter
      10. Spring wire diameter
      11. Number of spring turns
      12. Maximum U-235 enrichment (%)
      13. Average U-235 enrichment (%)
      14. Maximum gadolinia content (%)
      15. Water in pellet (ppm)
      16. Nitrogen in pellet (ppm)
      17. Pellet density (%TD)
      18. Open porosity (%)
      19. Pellet surface roughness (microns)
      20. Expected density increase (gms/cc)
      21. Sintering temperature (^{o}F)
      22. Cladding Alloy = (Material name)
      23. Final thermal treatment = (RXA or ?)
      24. Cladding surface roughness (microns)
      25. Cladding texture factor
      26. Cladding Hydrogen content (ppm)
      27. Fill gas pressure
      28. Fill gas composition
      29. Rate of CRUD accumulation factor (mils/hr)
      30. CRUD thermal conductivity

D. Coolant conditions

1. Coolant inlet temperature (°F)
2. Coolant mass flux (lbm/hr-ft^2)
3. System pressure (psia)

RESPONSE:

As agreed, a response to this item will be provided by September 30, 2012.

RAI #12

12. The NRC Standard Review Plan, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," (ADAMS Accession Number ML003734190) states: "The analysis methods and assumptions used by the licensee in determining the core inventory should be reviewed to ensure that they are based on current licensing basis rated thermal power, enrichment, and burnup."

    Enclosure 2 to the submittal states that AREVA fuel is approved for Combustion Engineering pressurized water reactor's for a maximum peak burnup of 62,000 Megawatt-Day(s) per Metric Ton Uranium (MWD/MTU) (Section 4.4.2). The current fuel is designed to ensure the fuel does not exceed 60,000 MWD/MTU (Section 4.4.1).

    A modification to the licensing basis fuel type can have the potential to change the core isotopic distribution and inventory assumed in post-accident conditions. The impacts regarding the core inventory due to changes other than the cladding (i.e. burnup) are not discussed in the proposed amendment. Please provide a justification to support that changes in the fuel design parameters do not significantly change the core isotopic distribution and magnitude (source term) for the design basis accidents analyzed.

RESPONSE:

The SONGS fission product inventory is calculated using the guidance in Section 3.1 of Alternative Source Term (AST) Regulatory Guide 1.183. Table 4.1-1 of the SONGS AST License Amendment Request (LAR) (ADAMS Accession No. ML043650403) summarizes the parameters modeled in the evaluation of the reactor core activity inventory. As detailed in Section 4.1.1 of the LAR, the core inventory of fission products is based on the maximum full-power operation of the core with, as a minimum, currently licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the emergency core cooling system evaluation uncertainty. These parameters were examined parametrically to maximize the fission product inventory. The ORIGEN-S code was executed for the various combinations of core average burnups (0, 10, 20, 30 and 40 Gigawatt-Days per Metric Ton Uranium (GWD/MTU)) and enrichments. For each isotope, the maximum curie value from the ORIGEN-S code runs was chosen to represent the inventory of that isotope in the composite fuel assembly. The SONGS AST License Amendment, based on this bounding source term, was issued in December 2006 (ADAMS Accession No. ML063400359).

The core average burnup range of 0 to 40 GWD/MTU conservatively bounds fuel management scenarios up to 24-month operating cycles irrespective of the peak pin burnup limit. As such, an increase in maximum peak burnup from 60 to 62 GWD/MTU will not increase the current bounding source term.

Section 4.9 of PCN-600 acknowledges the current reload analysis process for verifying that the current bounding source term and current radiological dose analyses are applicable to the new fuel cycle. If the current source terms or radiological dose analyses are invalidated, then the current reload analysis process addresses the need for new cycle-specific source terms to be generated for use in the accident radiological dose analyses. The methodology to calculate the bounding source terms will remain unchanged during and after the transition to AREVA fuel.

RAI #13

13. Enclosure 2, Section 7.4.2.3, Table 7.4.9, and Attachment C (Table C.1) of the submittal provide text and tables describing events analyzed in the Updated Final Safety Analysis Report (UFSAR), the acceptance criteria for these events, and the impact of the use of AREVA fuel on these analyzed events. The NRC staff has the following questions concerning this information.

    A. The NRC staff compared this information to the current UFSAR discussion and noted several differences. For example, some of the events described in the submittal have different acceptance criteria from those stated in UFSAR Table 15.0-8 (i.e., 10 CFR Part 100 limits vs. 10 CFR 50.67 limits). Explain and justify why the acceptance criteria for certain events described in the submittal differ from those in the UFSAR.

    B. For some events, Attachment C states that the event is bounded by another event. The UFSAR is not consistent with some of these statements in Attachment C. For example, Attachment C states that the UFSAR Section 15.1.2.1 event is bounded by the Section 15.1.2.3 doses. UFSAR Section 15.1.2.1.5 states that the doses for this event are bounded not by Section 15.1.2.1, but by Section 15.1.2.4 events. Please explain why Attachment C is inconsistent with the descriptions of the bounding events provided in the UFSAR and state which is correct.

    C. In the column labeled "Impact of AREVA Fuel" of Attachment C to Enclosure 2 (for UFSAR Sections 15.7.3.4, and 15.7.3.9) it states: "As all pins in both the dropped and impacted assemblies are assumed to fail, there is no difference with use of AREVA fuel." A review of these UFSAR sections shows that the UFSAR analysis assumes 226 fuel pins fail which is less than all the fuel pins in 2 assemblies (472 fuel pins). Please resolve this inconsistency.

    D. Many of the evaluations of the impact of the AREVA fuel only address the impact of the change on fuel failure (source term). Per Appendices E-H of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," dated July 2000 (ADAMS Accession No. ML003716792), the dose consequences of several accidents are dependent on both the source term and how the radioactivity is transported to the environment. A change in fuel has the potential for changing the release rate and the total amount of steam needed to cool down the plant after an accident. For accident analyses this steam is assumed to transport radioactivity to the environment. For those analyses that consider more than the source term (i.e. the main steamline break, steam generator tube rupture, locked rotor and rod ejection accidents) please address any impact of the AREVA fuel on the transport of radioactivity to the environment.

RESPONSE:

A. Per UFSAR Section 15.0, UFSAR Section 15.10 was added to present updated fuel cycle and unit specific data and consequences for the events presented in Sections 15.1 through 15.9. When comparing UFSAR Section 15.10 to UFSAR Sections 15.1 through 15.9 the following should be noted:

(a) UFSAR Sections 15.1 through 15.9 are consistent with the latest information that has been reviewed and approved by the NRC. These sections are intended to be updated when information has been submitted to and approved by the NRC.

(b) Section 15.10 presents the current plant configuration. This section includes data that has been added in accordance with 10 CFR 50.59 since the last approval of the event information by the NRC. This section is intended to be updated under the requirements of 10 CFR 50.59.

UFSAR Section 15.10.0.5 summarizes the assumptions, parameters, and calculational methods used to determine the doses that result from postulated accidents. As discussed in this section, UFSAR Appendix 15G provides a list of the accidents modeled using Alternative Source Term methodology (based on 10 CFR 50.67 limits), and UFSAR Appendix 15B.1 provides a list of the accidents modeled using pre-AST methodology (based on 10 CFR Part 100 limits).

B. UFSAR Section 15.1.2.1.5 contained an inconsistency which has been corrected per the SONGS corrective action program. The noted UFSAR text stated that the radiological consequences of this UFSAR Section 15.1.2.1 event are less severe than the results of the increased main steam flow event with a concurrent loss of offsite power discussed in paragraph 15.1.2.4.5. However, the increased main steam flow event with a concurrent loss of offsite power event is discussed in paragraph 15.1.2.3.5 (not 15.1.2.4.5).

As discussed in the response to Part "A", UFSAR Section 15.10 was added to present updated fuel cycle and unit specific data and consequences for the events presented in Sections 15.1 through 15.9. The discussion as to which events are bounded by which other events is addressed in the UFSAR Section 15.10 subsections. Consistent with Attachment C, UFSAR Section 15.10.1.2.1 correctly states that the UFSAR Section 15.10.1.2.1 event doses are bounded by the UFSAR Section 15.10.1.2.3 doses.

C. As discussed in the response to Part "A", UFSAR Section 15.10 was added to present updated fuel cycle and unit specific data and consequences for the events presented in Sections 15.1 through 15.9. Per UFSAR Sections 15.10.7.3.4 and 15.10.7.3.9, the number of fuel pins that fail during a fuel handling accident is 472 (i.e., all the fuel pins in two assemblies).

D. The transport of radioactive material to the environment is dependent on:

- Cladding integrity (Fuel Failure)
- Primary to secondary leakage
- Containment leakage
- Reactor Coolant System (RCS) leakage
- Engineered Safety Feature (ESF) leakage
- The steaming rate from secondary (Mass Release)

The cladding integrity (Fuel Failure) portion of the analysis remains unaffected. As discussed in Section 4.5.2 (page 61 of 166, Enclosure 2 to SONGS PCN 600) of the submittal, the NRC staffs is quoted as stating "*the statistical convolution technique is conservative and acceptable provided that the probability distribution for DNB is acceptable*".

As discussed in Section 4.2.1.1 (page 26 of 166, Enclosure 2 to SONGS PCN 600) of the submittal, the Modified Statistical Combination of Uncertainties (MSCU) analysis [

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For CEA ejection, the STRIKIN code [

]

The primary to secondary leakage, containment leakage, RCS leakage and ESF leakage portions of the analysis remain unaffected since they are independent of fuel type and cladding material. Therefore, [

]

The steaming rate (Mass Release) portion of the analysis remains unaffected. The mass release is dependent on the core sensible heat, the RCS sensible heat, the core decay heat, and the heat removal systems. As discussed in Section 4.5.3 (page 62 of 166, Enclosure 2 to SONGS PCN 600) of the submittal, the M5™ cladding thermal conductivity, h_{gap}, and the cladding specific heat [

]

RSB RAI #1

(Section 5.2) Please confirm that the Steam Generator 8% tube plugging assumption will remain bounding with respect to the number of tubes expected to be plugged.

RESPONSE:

The replacement steam generators (RSG), currently installed in both SONGS units, were designed and analyzed (including LOCA and non-LOCA events) assuming up to 8% plugged tubes per steam generator. Consistent with the RSG design basis, the LOCA and non-LOCA events preformed for and presented in PCN 600 were analyzed with an input value of up to 8% plugged tubes per steam generator. These are benchmark analyses to demonstrate the methodology to transition to AREVA fuel.

For reload analyses, the number of plugged tubes per steam generator is a procedurally controlled input into LOCA and non-LOCA analyses. The Reload Groundrules (RGR) documents the number of plugged tubes per steam generator (currently RGR Item IV.005) to be used for LOCA and non-LOCA events. The RGR is reviewed and updated for each SONGS unit and cycle reload analysis campaign per procedure "Reload Groundrules (RGR) Control Methodology." This procedure requires that all plant parameters used in the safety analyses be reviewed and updated to reflect the current or planned plant conditions applicable for the SONGS unit and cycle of interest. The RGR process is discussed in PCN 600 Section 6.3 and is unchanged from SONGS established process described in SONGS Reload Analysis Methodology Topical Report SCE-9801-P-A, Section 4.3.

Due to steam generator inspections during a refueling outage, the actual number of plugged tubes could change and must be confirmed to be in compliance with the value in the RGR prior to startup. The "Core Reload Analysis and Activities Checklist" procedure performed every cycle (Step 6.1.7 and documented in the procedure's Attachment 3, Table 3.5) requires that this value be confirmed prior to startup. Should the actual number of plugged tubes exceed the value in the RGR, LOCA and non-LOCA events would be reanalyzed/evaluated using a new bounding input value prior to startup. This process is identical to that used previously by SONGS for the Original Steam Generators.

Since Unit 3 steam generator inspections have not been completed at the time of this response, we cannot confirm that the 8% plugged tubes per steam generator input value used in the PCN 600 benchmark analyses will remain bounding. However, we can confirm that if the number of plugged tubes exceeds 8% plugged tubes per steam generator, the Unit 3 LOCA and non-LOCA events will be reanalyzed/evaluated in accordance with SONGS' procedures prior to unit startup.