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January 30, 2009

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

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

References:
1. Letter From NRC (M. Fields) to SCE (H. Ray) dated August 10, 1992," Acceptance of Topical Report SCE-9001, 'PWR Reactor Physics Methodology Using CASMO-3/SIMULATE-3' For Use at San Onofre Nuclear Generating Station, Units 1, 2, and 3 (TAC Nos. M77846, M77843, and M77844)"

2. Letter From NRC (S. Dembek) to SCE (H. Ray) dated June 2, 1999, "San Onofre Nuclear Generating Station, Units 2 and 3 – Evaluation of Reload Analysis Methodology Technology Transfer (TAC Nos. MA4289 and MA4290)"

Dear Sir or Madam:

Pursuant to 10 CFR 50.12, "Specific Exemptions," Southern California Edison Company (SCE) is requesting a temporary exemption from the requirements of 10 CFR 50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50, Appendix K, "ECCS Evaluation Models." In addition to the temporary exemption, pursuant to 10 CFR 50.90, SCE is requesting an amendment to Technical Specification 5.7.1.5, "Core Operating Limits Report." The change will allow the use of the CASMO-4 methodology to perform nuclear design calculations.

SCE is developing a lead fuel assemblies (LFAs, also known as lead test assemblies or lead use assemblies) program with AREVA NP.

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A001
NRR

Under this program, up to sixteen LFAs manufactured by AREVA NP may be inserted into the SONGS Unit 2 core or potentially into the SONGS Unit 3 core. Currently, eight AREVA LFAs are scheduled for installation in Unit 2 Cycle 16, with use for up to three operating cycles (Cycles 16, 17, and 18). Unlike current fuel assemblies, the AREVA LFAs will contain M5 alloy cladding material and Gadolinia (gadolinium oxide) burnable absorbers. As described below, SCE is requesting a temporary exemption to allow the use of M5 alloy cladding and a change to Technical Specification (TS) 5.7.1.5. Use of gadolinium oxide burnable absorbers is already authorized in TS 4.2.1.

Temporary Exemption Request

The temporary exemption is required to allow up to sixteen LFAs manufactured by AREVA NP with M5 alloy cladding fuel rods to be inserted into the Unit 2 or Unit 3 reactor core, beginning with the upcoming Unit 2 refueling outage (Cycle 16).

The use of M5 alloy cladding LFAs allows SCE to evaluate cladding for future fuel assemblies in order to eliminate grid to rod fretting fuel failures. Since the requirements in 10 CFR 50.46 specifically, and 10 CFR 50, Appendix K implicitly, refer to the use of Zircaloy or ZIRLO cladding, a temporary exemption is required to use fuel rods clad with an advanced zirconium-based alloy that is not Zircaloy or ZIRLO. The temporary exemption request and justification for the temporary exemption are described in detail in Enclosure 1 to this letter.

This temporary exemption is similar to the temporary exemption approved by the NRC for Calvert Cliffs (ADAMS Accession Number ML030640137), the temporary exemption for Palo Verde Unit 1 (ADAMS Accession Number ML082730003), and the exemption (ADAMS Accession Number ML011280063) and amendment (ADAMS Accession Number ML011300351) for Three Mile Island Unit 1.

A list of the regulatory commitments resulting from the exemption request is provided in Enclosure 2.

Request to Amend Technical Specification 5.7.1.5

The upgrade from CASMO-3 to CASMO-4 is needed to model the AREVA LFAs, as they will contain Gadolinia burnable absorbers.

The SCE reactor core physics methodology using CASMO-3 and SIMULATE-3 has been previously approved by the NRC (References 1 and 2). The proposed amendment would add the CASMO-4 computer code to the list of analytical methods that may be used to determine the core operating limits contained in TS 5.7.1.5. Similar to CASMO-3, CASMO-4 would be used in conjunction with SIMULATE-3. The amendment request is documented in Enclosure 3.

This request requires NRC review and approval of the enclosed Southern California Edison topical report, SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes."

SCE-0901 documents the applicability of the proposed CASMO-4 methodology to SONGS. This report also demonstrates SCE's proficiency to set up input decks, execute the programs, and properly interpret the results using CASMO-4.

SCE has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22 (c)(9).

Enclosure 3 to this letter contains the Description and No Significant Hazards Analysis for the proposed amendment.

The proposed amendment is neither exigent nor emergency. SCE requests approval of this license amendment request (LAR) with an allowance of 60 days for implementation of the approved amendment.

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

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

Executed on 1/30/2009
(Date)

Sincerely,

Michael P. Short

Enclosures:

Enclosure 1: Request for Temporary Exemption from the Provisions of 10 CFR 50.46 and 10 CFR 50, Appendix K for Lead Fuel Assemblies

Enclosure 2: Commitments

Enclosure 3: Description and No Significant Hazards Analysis for Proposed Change NPF-10/15-589 Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)" San Onofre Nuclear Generating Station Units 2 and 3

- Attachment A: Existing Technical Specification Page, Unit 2
- Attachment B: Existing Technical Specification Page, Unit 3
- Attachment C: Proposed Technical Specification Page (Redline and Strikeout), Unit 2
- Attachment D: Proposed Technical Specification Page (Redline and Strikeout), Unit 3
- Attachment E: Proposed Technical Specification Page, Unit 2
- Attachment F: Proposed Technical Specification Page, Unit 3

Enclosure 4: SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes," January 2009

cc: E. E. Collins, Regional Administrator, NRC Region IV
N. Kalyanam, 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 Public Health, Radiologic Health Branch

ENCLOSURE 1

Request for Temporary Exemption from
the Provisions of 10 CFR 50.46 and
10 CFR 50, Appendix K for Lead Fuel Assemblies

Introduction

The San Onofre Nuclear Generating Station (SONGS) Unit 2 or Unit 3 core consists of 217 fuel assemblies. Each fuel assembly consists of 236 fuel rods. The rods are arranged in a square 16 x 16 array. The fuel rods consist of slightly enriched uranium dioxide cylindrical ceramic pellets, encapsulated within a cylindrical Zircaloy or ZIRLO tube.

Part 50.46(a)(1)(i) of Title 10 of the Code of Federal Regulations [10 CFR 50.46(a)(1)(i)] states in part:

> *"Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated."*

10 CFR 50.46 continues on to delineate specifications for peak cladding temperature, maximum hydrogen generation, coolable geometry, and long-term cooling. Since 10 CFR 50.46 specifically refers to fuel with Zircaloy or ZIRLO cladding, the use of fuel clad with zirconium-based alloys that do not conform to either of these two designations requires a temporary exemption from this section of the regulations.

10 CFR 50, Appendix K, paragraph I.A.5, states:

> *"The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation."*

The Baker-Just equation presumes the use of Zircaloy or ZIRLO cladding. The use of fuel with zirconium-based alloys that do not conform to either of these two designations requires a temporary exemption from this section of the regulations.

Pursuant to 10 CFR 50.12, "Specific Exemptions," Southern California Edison Company (SCE) is requesting a temporary exemption from the requirements of 10 CFR 50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" and 10 CFR 50, Appendix K "ECCS Evaluation Models" for San Onofre Units 2 and 3.

The temporary exemption will allow a limited number (not to exceed 16) of lead fuel assemblies (LFAs, also known as lead test assemblies or lead use assemblies) manufactured by AREVA NP with M5 alloy clad fuel rods containing Gadolinia burnable

absorbers to be inserted into SONGS Unit 2 or 3 reactor core in nonlimiting core regions during the upcoming Cycle 16 refueling outage. The LFAs will remain in the core for up to three operating cycles. The use of M5 alloy cladding LFAs allows SCE to evaluate cladding for future fuel assemblies in order to eliminate grid to rod fretting fuel failures. The regulations specify standards and acceptance criteria only for fuel rods clad with Zircaloy or ZIRLO. Thus a temporary exemption is required to use fuel rods clad with an advanced alloy that is not Zircaloy or ZIRLO.

Currently, at least ten US plants have used M5 alloy cladding fuel assemblies either in full batch or LFA programs, including Arkansas Nuclear One Unit 1, Crystal River Unit 3, Three Mile Island Unit 1, Davis Besse, North Anna Units 1 and 2, Sequoyah Units 1 and 2, Oconee Unit 2, Ft. Calhoun, Palo Verde Unit 1, and Braidwood Unit 1. M5 alloy cladding fuel assemblies have also been used extensively in European plants.

10 CFR 50.12, Specific Exemption

The standards set forth in 10 CFR 50.12 provide that the Commission may grant exemptions from the requirements of the regulations of this part for reasons consistent with the following:

- The exemption is authorized by law;
- The exemption will not present an undue risk to the public health and safety;
- The exemption is consistent with the common defense and security; and
- Special circumstances are present.

This exemption is authorized by law. The remaining standards for the temporary exemption are also satisfied, as described in the following paragraphs.

The exemption will not present an undue risk to public health and safety. The NRC-approved M5 topical report (Reference 1) demonstrates that predicted chemical, mechanical, and material performance characteristics of the M5 alloy cladding are within those approved for Zircaloy under anticipated operational occurrences and postulated accidents. The LFAs will be placed in nonlimiting core regions as required by Technical Specification (TS) 4.2.1 "Fuel Assemblies." In the unlikely event that cladding failures occur in the LFAs, the environmental impact would be minimal and is bounded by previous accident analyses. Therefore, the use of the advanced zirconium-based cladding material, M5 alloy, will not present an undue risk to the public health and safety.

The exemption is consistent with the common defense and security. The use of M5 alloy clad LFAs allows SCE to evaluate this cladding material for use in future fuel assemblies, to provide a more robust design to eliminate grid to rod fretting fuel failures.

Special circumstances are present. As set forth in 10 CFR 50.12(a)(2)(ii), which states that special circumstances are present whenever *"Application of the regulation in*

the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule…".

10 CFR 50.46 identifies acceptance criteria for ECCS system performance at nuclear power facilities. The effectiveness of the ECCS in SONGS Units 2 and 3 will not be affected by the insertion of the LFAs. Due to the similarities in the material properties of the M5 alloy to Zircaloy or ZIRLO as identified in the AREVA M5 alloy topical report (Reference 1) and the placement of the LFAs in nonlimiting core regions, it can be concluded that the ECCS performance would not be adversely affected.

The intent of paragraph I.A.5 of Appendix K to 10 CFR 50 is to apply an equation for rates of energy release, hydrogen generation, and cladding oxidation from metal-water reaction that conservatively bounds all post-LOCA (Loss of Coolant Accident) scenarios. The supporting documentation for the AREVA M5 topical (Reference 1) shows that due to the similarities in the composition of the M5 alloy cladding and Zircaloy or ZIRLO, the application of the Baker-Just equation will continue to conservatively bound all post-LOCA scenarios.

A strict interpretation of 10 CFR 50.46 and 10 CFR 50, Appendix K, would not allow the use of M5 alloy clad fuel rods in lead fuel assemblies since the cladding material does not fall within the strict definition of Zircaloy or ZIRLO even though the AREVA M5 topical report shows that the intent of the regulations is met. Application of these regulations in this particular circumstance would not serve the underlying purpose of the rule and is not necessary to achieve the underlying purpose of the rule, so special circumstances exist.

Lead Fuel Assembly (LFA) Program Summary

SCE is requesting this temporary exemption in order to install LFAs manufactured by AREVA NP in the SONGS Unit 2 core or potentially into the SONGS Unit 3 core. The AREVA NP mechanical design for the SONGS LFAs is similar to the standard AREVA NP high thermal performance (HTP) fuel designed for Combustion Engineering (CE) 14x14 fuel pin lattice reload fuel, with the primary difference being the 16x16 fuel pin lattice. The M5 cladding will be used on the Calvert Cliffs 14x14 CE design, beginning in year 2010. The AREVA LFA design for SONGS is very similar to the AREVA LFAs being evaluated at Palo Verde (ADAMS Accession Number ML082730003). The mechanical design evaluations for the LFAs will be performed with the standard reload mechanical design methods using the M5 alloy cladding properties. The NRC has reviewed and approved (Reference 2) the M5 alloy cladding properties in topical report BAW-10227P-A (Reference 1).

The LFAs are currently scheduled for installation in Cycle 16 of SONGS Unit 2, for up to three cycles of irradiation (Cycles 16, 17, and 18). The burnup achieved after three cycles of irradiation will be less than the current NRC approved (Reference 3) San Onofre burnup limit of 60 MWd/kgU, which is also less than the approved AREVA NP methodology peak rod limit of 62 MWd/kgU as described in report ANF-88-133(P)(A)

(Reference 4). Prior to use of AREVA LFAs for a second or third fuel cycle of irradiation, poolside LFA examinations will be performed to evaluate assembly and cladding performance, and acceptability for continued use (Commitments 1 and 2). If the AREVA LFAs are inserted for a third fuel cycle of irradiation, then poolside LFA examinations will be performed after completion of the third fuel cycle of irradiation to evaluate assembly and cladding performance (Commitment 3).

The fuel management will place the LFAs in nonlimiting core regions. Since these assemblies will not be in the highest power density locations, the placement scheme will assure that the behavior of the LFAs is bounded by the safety analyses performed for the standard fuel rods.

SCE, AREVA, and Westinghouse evaluations will verify performance of the LFAs with respect to the safety analysis. The analyses will include thermal hydraulic compatibility, LOCA and non-LOCA criteria, mechanical design, seismic and core physics. The evaluations will make use of the fact that the LFAs will be operated in nonlimiting core regions and will verify that the reload analyses are not adversely impacted. In addition, an evaluation will be performed to verify the insertion of the AREVA LFAs does not adversely impact the fuel performance and mechanical integrity of the co-resident Westinghouse fuel.

LFA Mechanical Design Description

The LFAs for the SONGS Unit 2 or Unit 3 reactor will be the AREVA NP 16 x 16 CE design. The fuel bundle uses ten M5 spacer grids of the high thermal performance (HTP) design and one Alloy 718 spacer grid of the high mechanical performance (HMP) design. The lower end fitting is the FUELGUARD design, and the upper end fitting is a reconstitutable AREVA NP design for CE fuel. The HTP spacer grid was generically reviewed and accepted by the NRC and has been used in reload design for CE, Westinghouse, and Kraftwork Union reactors since 1991. The FUELGUARD lower end fitting has also been used in reload design for CE, Westinghouse, and General Electric reactors. The reconstitutable upper end fitting design has been used in reload design for plants with CE 14 x 14 fuel pin lattices.

Each fuel bundle contains four outer guide tubes, one center guide/instrument tube, and 236 fuel rods. The LFA fuel rods have the same pellet stack height and overall dimension as the co-resident Westinghouse fuel.

The primary differences between the Westinghouse fuel design currently used in the SONGS reactors and the AREVA fuel design include (1) the use of the different zirconium-based alloys for fuel rod cladding, guide tubes, and spacer grids; (2) the use of HTP grids; and (3) the use of a different burnable absorber (Gadolinia).

LFA in Nonlimiting Core Regions

SONGS TS 4.2.1 "Fuel Assemblies," states:

"The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Integral or Discrete Burnable Absorber Rods may be used. They may include: borosilicate glass – Na_2O-B_2O_3-SiO_2 components, boron carbide – B_4C, zirconium boride – ZrB_2, gadolinium oxide – Gd_2O_3, erbium oxide – Er_2O_3. Limited substitutions of zirconium alloy (such as ZIRLO™ or Zircaloy) or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions."

SCE is currently planning to place eight AREVA LFAs in nonlimiting core regions in Cycle 16. SCE has defined nonlimiting core regions as core locations where the peak integrated radial power peaking factor in the LFAs will be 0.95 or less of the core maximum integrated radial power peaking factor at all times in life (Commitment 4). Therefore, the LFAs will not contain the limiting rod in the core and will have margin relative to the bounding peaking factors used in safety analyses. This criterion is consistent with that used in the AREVA LFA program at Palo Verde (ADAMS Accession Number ML082620212). The AREVA LFAs will be explicitly modeled 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). SCE will design the LFA lattice, Cycle 16 core loading pattern, and perform reload physics analyses using both the Westinghouse (ABB-CE) reload methodology (Reference 5) and STUDSVIK CASMO and SIMULATE Code packages. The SONGS Cycle 16 core design will ensure that the LFA predicted peak rod power meets the 0.95 criterion.

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 Emergency Core Cooling System (ECCS) performance analyses. The 0.95 radial power peaking factor criterion applied to the LFA is a means of applying the 5% penalty to the LFAs. 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. Additionally, the maximum LFA integrated fuel rod burnup will be maintained less than or equal to 60 MWd/kgU, the SONGS limit in Updated Final Safety Analysis Report (UFSAR) section 4.2.3.2.12.4 "SONGS Burnup Extension to 60,000 MWD/MTU."

For subsequent cycles containing the LFAs (Cycle 17 and potentially Cycle 18), SCE will explicitly model and analyze the AREVA LFAs in the reload core physics analyses, in the same manner as is performed for the lead cycle, Cycle 16. In addition to being once or twice burned, the reload core design for these cycles will determine the physical

peaking factors lower than the 0.95 criterion, thereby ensuring that the LFAs remain in nonlimiting core regions (Commitment 4).

AREVA LFA Analyses

AREVA will perform 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 core (Commitment 6). The analyses will make use of the fact that the LFAs will be operated in nonlimiting core regions and will verify the reload analyses are not adversely impacted.

These analyses will include tasks such as physical design, normal and faulted operations, growth calculations, pressure drop and flow testing, among others. In addition, AREVA will analyze the seismic performance of the LFAs by evaluating the seismic/LOCA time history with respect to the strength of the AREVA LFAs.

The AREVA LFA analyses require the resources and coordination of SCE, AREVA, and Westinghouse. SCE will maintain overall responsibility for the LFA project. SCE has entered a three-party proprietary information agreement with AREVA and Westinghouse which allows the exchange of technical information for this project to ensure the compatibility of the LFAs with the SONGS co-resident Westinghouse fuel and core internals. SCE acts as the intermediary between AREVA and Westinghouse to ensure that each organization has the necessary and appropriate inputs to perform the analyses.

Co-Resident Westinghouse Fuel Compatibility Analyses

Westinghouse will perform a compatibility analysis to ensure that insertion of the AREVA LFAs will not cause the remaining Westinghouse fuel to exceed its operating limits and ensure there is no adverse impact on the fuel performance or mechanical integrity (Commitment 7).

Poolside LFA Examinations

Poolside LFA examinations to assess key performance measures will include, as a minimum, 4-face inspections of the highest burn LFAs. Based on results of the inspection, the inspection scope may be expanded. Additional scope inspections could include but are not limited to additional visual inspections, oxide/crud lift-off measurements, fretting and diameter measurements, shoulder gap, assembly length and guide tube wear measurements.

Precedent

This temporary exemption request is similar to the temporary exemption approved by the NRC for Calvert Cliffs (ADAMS Accession Number ML030640137) and Palo Verde (ADAMS Accession Number ML082730003), and to the amendment and related exemption for Three Mile Island Unit 1 (ADAMS Accession Number ML011300351).

References

1. BAW-10227P-A, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, Framatome Cogema Fuels, February 2000 (proprietary) (ADAMS Ascension Number ML003686305).

2. NRC Revised Safety Evaluation (SE) for Topical Report BAW-10227P: "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel" (TAC NO. M99903) dated February 4, 2000 (ADAMS Ascension Numbers ML003681479 and ML003681490).

3. NRC Generic Approval Of C-E Topical Report CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel" (TAC NO. M82192) dated June 22, 1992 (proprietary).

4. ANF-88-133(P)(A) and Supplement 1, Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62GWd/MTU, Advanced Nuclear Fuels Corporation, December 1991 (proprietary).

5. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3", June 1999 (proprietary).

ENCLOSURE 2

Commitments

Commitments

1. Prior to use of AREVA Lead Fuel Assemblies (LFAs) for a second fuel cycle of irradiation, poolside LFA examinations will be performed to evaluate assembly and cladding performance, and acceptability for continued use.

2. Prior to use of AREVA LFAs for a third fuel cycle of irradiation, poolside LFA examinations will be performed to evaluate assembly and cladding performance, and acceptability for continued use.

3. If the AREVA LFAs are inserted for a third fuel cycle of irradiation, then poolside LFA examinations will be performed after completion of the third fuel cycle of irradiation to evaluate assembly and cladding performance.

4. The AREVA LFAs will be placed in core locations where the peak integrated radial power peaking factor in the LFAs will be 0.95 or less of the core maximum integrated radial power peaking factor at all times in life.

5. The AREVA LFAs will be modeled in the SONGS core physics models and their impact will be analyzed in the cycle-specific core physics calculations that support the reload analyses.

6. Analyses will be performed to verify the performance of the AREVA LFAs. These analyses include thermal-hydraulic compatibility, loss-of-coolant accident (LOCA) and non-LOCA criteria, mechanical design, thermal hydraulic, seismic, core physics, and neutronics compatibility of the AREVA LFAs in the SONGS reactor core. The analyses will make use of the fact that the LFAs will be operated in nonlimiting core regions and will verify the reload analyses are not adversely impacted.

7. A compatibility analysis will be performed to ensure that insertion of the AREVA LFAs will not cause the remaining Westinghouse fuel to exceed its operating limits and ensure there is no adverse impact on the fuel performance or mechanical integrity.

ENCLOSURE 3

Description and No Significant Hazards Analysis
for Proposed Change NPF-10/15-589
Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)"
San Onofre Nuclear Generating Station Units 2 and 3

Description and No Significant Hazards Analysis
for Proposed Change NPF-10/15-589
Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)"
San Onofre Nuclear Generating Station Units 2 and 3

EXISTING TECHNICAL SPECIFICATIONS

Unit 2: see Attachment A
Unit 3: see Attachment B

PROPOSED TECHNICAL SPECIFICATIONS
(highlight for additions, strikeout for deletions)

Unit 2: see Attachment C
Unit 3: see Attachment D

PROPOSED TECHNICAL SPECIFICATIONS
(with changes)

Unit 2: see Attachment E
Unit 3: see Attachment F

1.0 INTRODUCTION

Southern California Edison Company (SCE) requests NRC approval to incorporate use of the Studsvik-Scandpower computer program CASMO-4 (Reference 1) to support physics design analyses.

The currently approved SCE reload analysis methodology uses CASMO-3 (Reference 2) for cross-section generation and SIMULATE-3 (Reference 3) for core simulation. The current SCE reload analysis methodology is documented in SCE-9801-P-A (Reference 4) and SCE-9001-A (Reference 5).

The CASMO-3 and SIMULATE-3 methodologies have been used to evaluate reload designs at San Onofre for over fifteen years. SCE proposes to allow the use of CASMO-4 to support physics design analyses. CASMO-4 will be used in conjunction with SIMULATE-3. CASMO-4 streamlines the cross-section generation process while maintaining current peaking factor uncertainties. CASMO-4 was benchmarked against

CASMO-3 and it was determined that CASMO-4 does not reduce previously approved safety margins. SCE physics analysis methodology report SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes" (Reference 6), is enclosed for NRC review and approval.

2.0 PROPOSED CHANGE

SCE requests NRC approval to use the CASMO-4 computer program for San Onofre physics analyses to streamline the process of generating cross-sections and to reduce the potential for human error. SCE has prepared topical report SCE-0901 to support physics analyses using the CASMO-4 computer program.

TS 5.7.1.5, "Core Operating Limits Report (COLR)", will be revised by adding new Reference 9 to Section (b) as follows:

9. SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes"

3.0 BACKGROUND

Studsvik has developed a new cross-section generation program called CASMO-4 since the initial development of SCE's reload analysis methodology. CASMO-4, like CASMO-3, is a two-dimensional neutron transport theory lattice physics code with depletion capability and the ability to generate cross-sections and discontinuity factors for boiling water reactor (BWR) and pressurized water reactor (PWR) diffusion theory physics analysis. SCE will use CASMO-4 for cross-section generation. The cross-sections from CASMO-4 will be used in SIMULATE-3 to generate physics parameters used to perform safety analyses.

CASMO-4 incorporates the microscopic depletion of burnable absorbers into the main calculations, which replaces the MICBURN-3 auxiliary code needed by CASMO-3. A new feature of CASMO-4 is the use of the characteristics form for solving the transport equation, which allows for a true heterogeneous geometry in the two-dimensional calculation. Studsvik has also automated the cross-section generation process with CASMO-4 so that complete nuclear data for SIMULATE-3 can be generated in one execution, which results in reduced potential for human error in the model development process.

CASMO-4 is in widespread use in the nuclear industry. Refer to Section 7.0 for examples of where the NRC has specifically approved the use of CASMO-4.

4.0 TECHNICAL ANALYSIS

SCE has performed benchmarking of CASMO-4 against plant-specific measured data and against data calculated with CASMO-3 from Reference 5. Studsvik-Scandpower has performed benchmarking of CASMO-4 against critical experiments, including Gadolinia as the burnable absorber. The CASMO-4 benchmarking results are included in SCE-0901 (Reference 6).

The Critical Boron Concentrations, CEA worths, and Isothermal Temperature Coefficients generated with the CASMO-4 model show good general agreement with measurements. The peaking factors calculated with the CASMO-4 model also show good agreement with those calculated with the CASMO-3 model. Also, the peaking factor uncertainties are not changing as a result of CASMO-4. Thus, the newer version of CASMO does not reduce previously approved margins.

As discussed in SCE-0901, CASMO-4 can accurately model Gadolinia burnable absorbers. This capability is important to SCE since the AREVA lead fuel assemblies (LFAs) use gadolinia as the burnable absorber.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

SCE has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment", as discussed below.

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   SCE is adding the CASMO-4 computer program to its physics analysis methodology and will use the program for nuclear design analysis. This will allow the use of the CASMO-4 methodology to perform all steady-state pressurized water reactor (PWR) nuclear design analyses. The probability of occurrence of an accident previously evaluated will not be increased by the proposed change in the particular computer programs used for physics calculations for nuclear design analysis. The results of nuclear design analyses are used as inputs to the analysis of accidents that are evaluated in the Updated Final Safety Analysis Report (UFSAR). These inputs do not alter physical characteristics or modes of operation of any system, structure, or

component involved in the initiation of an accident. Thus, there is no significant increase in the probability of an accident previously evaluated as a result of this change.

The consequences of an accident evaluated in the UFSAR are affected by the values of the physics inputs to the safety analysis. An extensive benchmark of CASMO-4 was performed with both San Onofre measured and predicted data, and with critical experiments. The accuracy of the CASMO-4 model is similar to the accuracy of the CASMO-3 model. Furthermore, there is the potential for the value of the nuclear design parameters to change solely as a result of the new core reload full core loading pattern. Regardless of the source of a change, an assessment is made of changes to the nuclear design parameters with respect to their effects on the consequences of accidents previously evaluated in the UFSAR. Thus, the nuclear design parameters are intermediate results and by themselves will not result in an increase in the consequences of an accident evaluated in the UFSAR.

Therefore, the use of the CASMO-4 methodology, which will perform the same functions as the existing CASMO-3 methodology with similar accuracy, does not significantly increase the consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The possibility for a new or different kind of accident evaluated previously in the UFSAR will not be created by the change to the particular methodologies used for physics calculations for nuclear design analyses. The change involves adding CASMO-4 to the SCE physics analysis methodology. CASMO-4 is an update to the CASMO-3 methodology currently approved for use at San Onofre. The results of nuclear design analyses are used as inputs to the analysis of accidents that are evaluated in the UFSAR. These inputs do not alter the physical characteristics or modes of operation of any system, structure or component involved in the initiation of an accident. Therefore, the addition of CASMO-4, which will perform the same functions as CASMO-3 with similar accuracy, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change does not involve a significant reduction in a margin of safety. The margin of safety as defined in the basis for any technical specification will not be reduced by the proposed change to the computer programs used for physics calculations for nuclear design analyses.

The change involves the addition of CASMO-4 to the SCE physics analysis methodology for nuclear design analysis. Extensive benchmarking of CASMO-4 has demonstrated that the values of those parameters used in the safety analysis are not significantly changed relative to the values obtained using the NRC approved 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. Specifically, use of these code specific biases and uncertainties in safety analyses continues to provide the same statistical assurance that the values of the nuclear parameters used in the safety analysis are conservative with respect to the actual values on at least a 95/95 probability/confidence basis.

Based on the above, SCE concludes that the proposed amendment presents no significant hazards considerations under the standards set forth in I0 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

SCE's NRC approved physics analysis methodologies, described in Reference 4 and Reference 5, meet applicable regulatory requirements including the following 10 CFR 50 Appendix A general design criteria:

Criterion 10 - Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11 - Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12 - Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified

acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Including the CASMO-4 computer program in SCE's physics analysis methodology does not change SCE's compliance with applicable regulatory requirements.

5.3 Conclusion

SCE evaluated the CASMO-4 computer program and determined that calculations performed with CASMO-4 yield similar results to its predecessor CASMO-3. CASMO-4 also does not change uncertainties associated with the peaking factors. Thus, CASMO-4 does not reduce the previously approved margin of safety.

The change does not alter, degrade, or prevent actions described or assumed in any accident analysis in the UFSAR. It will not change any assumptions previously made in evaluating radiological consequences or affect any fission product barriers, nor does it increase or have any impact on the consequences of events described and evaluated in Chapter 15 of the UFSAR.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL EVALUATION

Based on the above considerations, the proposed amendment does not involve and will not result in a condition which significantly alters the impact of San Onofre on the environment. Thus, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51.22(c)(9), and pursuant to 10 CFR Part 51.22(b), no environmental assessment need be prepared.

7.0 PRECEDENT

Examples of where the NRC has specifically approved the use of CASMO-4 include McGuire (ADAMS Accession Number ML082820015), Catawba (ADAMS Accession Number ML082820047), Palo Verde (ADAMS Accession Number ML010860187), and North Anna and Surry (ADAMS Accession Number ML030700038) and Fort Calhoun (ADAMS Accession Number ML050750534).

8.0 REFERENCES

1. SSP-01/400 Rev 4, Studsvik-Scandpower Inc, December 2004, "CASMO-4 – A Fuel Assembly Burnup Program, User's Manual"

2. STUDSVIK/NFA-89/3, Studsvik Energiteknik AB, Sweden, November 1989, "CASMO-3 – A Fuel Assembly Burnup Program, User's Manual, Version 4.4"

3. STUDSVIK/SOA-92/01, Studsvik of America Inc, April 1992, "SIMULATE-3 – Advanced Three-Dimensional Two-Group Reactor Analysis Code, Version 4.02"

4. SCE-9801-P-A Rev 0, Southern California Edison, June 1999, "Reload Analysis Methodology For The San Onofre Nuclear Generating Station, Units 2 and 3"

5. SCE-9001-A, Southern California Edison "PWR Reactor Physics Methodology Using CASMO-3/SIMULATE-3," September 1992

6. SCE-0901, Southern California Edison "PWR Reactor Physics Methodology Using Studsvik Design Codes," January 2009

Attachment A

Existing Technical Specification Page, Unit 2

5.7 Reporting Requirements (continued)

5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"

2. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

3. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"

4. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"

5. CEN-635(S), "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology"

6. Letter, dated May 16, 1986, G. W. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 3 SER)

7. Letter, dated January 9, 1985, G. W. Knighton (NRC) to K. P. Baskin, "Issuance of Amendment No. 30 to Facility Operating License NPF-10 and Amendment No. 19 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 2 SER)

8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

Attachment B

Existing Technical Specification Page, Unit 3

5.7 Reporting Requirements (continued)

5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"

2. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

3. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"

4. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"

5. CEN-635(S), "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology"

6. Letter, dated May 16, 1986, G. W. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 3 SER)

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8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A

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5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

Attachment C

Proposed Technical Specification Page, Redline and Strikeout, Unit 2

5.7 Reporting Requirements (continued)

5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"

2. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

3. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"

4. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"

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7. Letter, dated January 9, 1985, G. W. Knighton (NRC) to K. P. Baskin, "Issuance of Amendment No. 30 to Facility Operating License NPF-10 and Amendment No. 19 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 2 SER)

8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A

9. SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes"

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

Attachment D

Proposed Technical Specification Page, Redline and Strikeout, Unit 3

5.7 Reporting Requirements (continued)

5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"

2. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

3. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"

4. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"

5. CEN-635(S), "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology"

6. Letter, dated May 16, 1986, G. W. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 3 SER)

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8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A

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c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

Attachment E

Proposed Technical Specification Page, Unit 2

5.7 Reporting Requirements (continued)

5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"

2. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

3. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"

4. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"

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8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A

9. SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes"

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d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

Attachment F

Proposed Technical Specification Page, Unit 3

5.7 Reporting Requirements (continued)

5.7.1.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model"

2. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"

3. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties"

4. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3"

5. CEN-635(S), "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology"

6. Letter, dated May 16, 1986, G. W. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 3 SER)

7. Letter, dated January 9, 1985, G. W. Knighton (NRC) to K. P. Baskin, "Issuance of Amendment No. 30 to Facility Operating License NPF-10 and Amendment No. 19 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 2 SER)

8. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A

9. SCE-0901, "PWR Reactor Physics Methodology Using Studsvik Design Codes"

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d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

ENCLOSURE 4

SCE-0901, PWR Reactor Physics Methodology
Using Studsvik Design Codes

SCE-0901

PWR REACTOR PHYSICS METHODOLOGY USING STUDSVIK DESIGN CODES

January 2009

SOUTHERN CALIFORNIA EDISON COMPANY

SCE-0901

PWR REACTOR PHYSICS METHODOLOGY USING STUDSVIK DESIGN CODES

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ABSTRACT

This report documents the validation and level of accuracy of the reactor core physics methodology used by Southern California Edison Company to perform steady-state analyses for for Pressurized Water Reactors (PWR). In June 1992, SCE obtained NRC approval to use CASMO-3 / SIMULATE-3 (Reference 1) for reactor core physics design activities. SCE is upgrading from CASMO-3 (and supporting programs) to CASMO-4 (and supporting programs). The CASMO-4 / SIMULATE-3 methodology has been validated by an in-house benchmarking effort against predictions with measured data, critical experiments, a higher order computer program (MCNP), and analyses of record (CASMO-3 / SIMULATE-3 results). Based on the results from this benchmarking effort, a set of 95/95 tolerance limits (uncertainties) has been calculated. Southern California Edison Company intends to use this methodology to perform PWR calculations including reload design, input to safety analyses, startup predictions, core physics databooks, and reactor protection system and monitoring system setpoint updates.

TABLE OF CONTENTS

	Page
Disclaimer	ii
Abstract	iii
List of Tables	vi
List Of Figures	viii
List Of Acronyms	ix
SECTION 1 INTRODUCTION, OVERVIEW, AND SUMMARY	1
1.0 Introduction	
1.1 Overview	
1.2 Summary	
SECTION 2 DESCRIPTION OF METHODOLOGY	5
2.0 Introduction	
2.1 Computer Program Description	
2.2 Model Descriptions	
SECTION 3 DESCRIPTION OF REACTORS USED IN BENCHMARKING	12
3.0 Introduction	
3.1 San Onofre Units 2 And 3	
SECTION 4 BENCHMARK COMPARISONS	19
4.0 Introduction	
4.1 Critical Boron Concentration	
4.1.1 Zero Power Critical Boron Concentration	
4.1.2 Hot-Full-Power Critical Boron Concentration	
4.2 Isothermal Temperature Coefficient	
4.3 Power Coefficient	
4.4 Control Rod Worth	
4.5 Net (N-1) Control Rod Worth	
4.6 Inverse Boron Worth	

TABLE OF CONTENTS (Continued)

Page

SECTION 5 POWER DISTRIBUTION AND PIN PEAKING COMPARISONS 43
5.0 Introduction
5.1 Local Pin Power
5.2 Pin Peaking Factors (Fq, Fxy, Fr)
5.3 Core Radial And Axial Power Distributions
5.4 Axial Shape Index

SECTION 6 FUTURE BURNABLE ABSORBERS AND FUEL ROD CLADDING 53
6.0 Introduction
6.1 Criticals with Gadolinia
6.2 MCNP Comparisons for IFBA and Gadolinia Lattices
6.3 Conclusion For Future Burnable Absorbers
6.4 M5 Cladding

SECTION 7 CONCLUSION 62

SECTION 8 REFERENCES 63

LIST OF TABLES

Number	Title	Page
1.1	CASMO-4 / SIMULATE-3 Biases And Uncertainties	4
2.1	CASMO-3 to CASMO-4 Code Feature Comparison	10
3.1	Mechanical Design Parameters – SONGS 2&3	14
4.1	HZP Critical Boron Concentration	28
4.2	SONGS 2 Cycle 12 Critical Boron Concentration – HFP	29
4.3	SONGS 2 Cycle 13 Critical Boron Concentration – HFP	30
4.4	SONGS 2 Cycle 14 Critical Boron Concentration – HFP	31
4.5	SONGS 3 Cycle 12 Critical Boron Concentration – HFP	32
4.6	SONGS 3 Cycle 13 Critical Boron Concentration – HFP	33
4.7	SONGS 3 Cycle 14 Critical Boron Concentration – HFP	34
4.8	HZP Isothermal Temperature Coefficient – Cycles 10 – 15	35
4.9	HZP Isothermal Temperature Coefficient – Cycles 1 – 5	36
4.10	At-Power Isothermal Temperature Coefficient – Cycles 1 – 5	37
4.11	Power Coefficients	38
4.12	Control Rod Bank Worths – Unit 2	39
4.13	Control Rod Bank Worths – Unit 3	40
4.14	Net (N-1) Control Rod Worth	41

LIST OF TABLES (Continued)

Number	Title	Page
4.15	Inverse Boron Worth	42
5.1	Power Distribution Peaking Factor – Fxy	46
5.2	Power Distribution Peaking Factor – Fr	47
5.3	Power Distribution Peaking Factor – Fq	48

LIST OF FIGURES

Number	Title	Page
2.1	Computer Program Sequence Flow Chart	11
3.1	SONGS 2 And 3 Reactor Core With Control Rod Pattern	16
3.2	SONGS 2 And 3 Reactor Core With Incore Instrument Pattern	17
3.3	SONGS 2 And 3 16x16 Fuel Assembly	18
5.1	S2C14 Axially Integrated RPD Comparison 100% HFP -- ARO -- BOC	49
5.2	S2C14 Axially Integrated RPD Comparison 100% HFP -- ARO -- EOC	50
5.3	S2C14 Average Axial Power Distribution Comparison 100% HFP -- ARO -- BOC	51
5.4	S2C14 Average Axial Power Distribution Comparison 100% HFP -- ARO -- EOC	52

LIST OF ACRONYMS

ARO	All Rods Out
BOC	Beginning Of Cycle
CBC	Critical Boron Concentration
CEA	Control Element Assembly (Control Rod)
C-3	CASMO-3
C-4	CASMO-4
C-3/S-3	CASMO-3 / SIMULATE-3
C-4/S-3	CASMO-4 / SIMULATE-3
EARO	Essentially All Rods Out
EOC	End Of Cycle
Fr	Power Distribution Integrated Peaking Factor
Fq	Power Distribution Total peaking Factor
Fxy	Power Distribution Planar Peaking Factor
HFP	Hot Full Power
HZP	Hot Zero Power
PCM	Percent Mil (Unit of Reactivity = 10^{-5} $\Delta k/k$)
SCE	Southern California Edison Company
SONGS	San Onofre Nuclear Generating Station

SECTION 1

INTRODUCTION, OVERVIEW, AND SUMMARY

1.0 INTRODUCTION

This document describes the Southern California Edison Company (SCE) San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 reactor core physics methodology and model verification using the CASMO-4 and SIMULATE-3 computer programs. Studsvik-Scandpower Inc. is the developer of the CASMO-4 and SIMULATE-3 computer programs.

In June 1992, SCE obtained NRC approval to use CASMO-3 / SIMULATE-3 (Reference 1) for reactor core physics design activities. SCE is upgrading from CASMO-3 (and supporting programs) to CASMO-4 (and supporting programs). The CASMO-4 computer program package is superior in many respects to CASMO-3, which although extremely reliable, has become dated. This modernization initiative is part of the ongoing commitment of SCE to use best available well established methods for reactor analysis.

1.1 OVERVIEW

The data demonstrating the applicability of CASMO-4 for SONGS Units 2 and 3 core physics analyses are documented in Sections 2 through 7.

Section 2, "**Description Of Methodology,**" describes the CASMO-4 / SIMULATE-3 computer program package.

Section 3, "**Description Of Reactors Used in Benchmarking,**" describes the SONGS reactors (Units 2 and 3) and cycles (1 - 5 and 10 - 14) modeled with CASMO-4. San Onofre Unit 1 has been decommissioned.

Section 4, "**Benchmark Comparisons,**" provides the benchmarking of the core physics parameters calculated for SONGS 2&3. CASMO-4 / SIMULATE-3 results are compared with plant measurements. The sample mean (bias) and standard deviation are quantified. Finally, a 95/95 tolerance limit (Uncertainty) is determined.

Section 5, "**Power Distribution Comparisons,**" provides the benchmarking of the power distribution and pin peaking parameters listed in Table 1.1. It is shown that the power distributions calculated by CASMO-4 / SIMULATE-3 and CASMO-3 / SIMULATE-3 are essentially identical. Therefore, the biases and uncertainties for CASMO-3 also apply to CASMO-4.

Section 6, "**Future Burnable Absorbers And Fuel Rod Cladding,**" summarizes critical experiment benchmarking (including gadolinia cores) performed by Studsvik-Scandpower.

MCNP benchmarking results for Integral Fuel Burnable Absorber (IFBA) fuel are also presented. SONGS currently uses erbia burnable absorber but may in the future wish to use gadolinia or IFBA in Lead Fuel Assemblies (LFAs) and then subsequently in full reload batches. The CASMO-4/SIMULATE-3 methodology is applicable to these burnable absorbers.

SONGS currently uses Zircaloy-4 and Zirlo fuel rod cladding. AREVA LFAs will have M5 cladding. The CASMO-4 / SIMULATE-3 methodology is applicable to Zircaloy-4, Zirlo, and M5 cladding.

Section 7, "**Conclusions,**" presents the conclusions of this report and the range of application for which SCE will use this methodology.

Section 8, "**References,**" presents documents referenced in this report.

1.2 SUMMARY

Table 1.1 summarizes the bias and 95/95 tolerance limits for CASMO-4 / SIMULATE-3 calculated in Sections 4 and 5. The tolerance limits are such that, when applied to the CASMO-4 / SIMULATE-3 results, there is a 95 percent probability, with a 95 percent confidence that the calculated values will conservatively bound the "true" values. Based on the analyses and results contained in this report, the CASMO-4 / SIMULATE-3 methodology may be used for physics analyses of SONGS Units 2 and 3 with the same level of accuracy as CASMO-3.

SCE concludes that CASMO-4 / SIMULATE-3 is acceptable for performing all SONGS Units 2 and 3 steady-state core physics analyses including:

Reload Design
Safety Analyses Input
Startup Predictions
Plant Physics Data Books
Reactor Protection and Monitoring System Updates

TABLE 1.1

CASMO-4 / SIMULATE-3 Biases And Uncertainties

Parameter	Bias	Uncertainty	Units*
HZP PPM	-1 ppm	23 ppm	Absolute
HFP PPM	30 ppm	20 ppm	Absolute
IBW	-3.9 %	13.6 %	Relative
Power Coefficient	0 pcm/%P	2 pcm/%P	Absolute
ITC	-0.5 pcm/°F	2.5 pcm/°F	Absolute
CEA Bank Worth	1.3 %	7.6 %	Relative
Local Pin Power	0 %	1.78 %	Relative
Axial Shape Index	0.003	0.014	Absolute
Assembly Fq	0.0 %	4.17 %	Relative
Assembly Fxy	0.0 %	4.80 %	Relative
Assembly Fr	0.0 %	3.34 %	Relative
Fuel Rod Fq	0.0 %	4.62 %	Relative
Fuel Rod Fxy	0.0 %	5.20 %	Relative
Fuel Rod Fr	0.0 %	3.89 %	Relative

* For those parameters with differences expressed in relative units:

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

For those parameters with differences expressed in absolute units:

Predicted Value = Calculated Value + (Bias $\pm$ Uncertainty)

SECTION 2

DESCRIPTION OF METHODOLOGY

2.0 INTRODUCTION

This section provides a brief description of the CASMO-4 / SIMULATE-3 methodology. This computer program package has already received NRC approval for use in core physics calculations at other utilities (References 2, 3, 4, 5, and 6).

In general, the methodology is similar to the SCE CASMO-3 Topical Report (Reference 1). However, the Palo Verde Topical Report (Reference 2) is followed for Inverse Boron Worth and Local Pin Power biases and uncertainties.

2.1 COMPUTER PROGRAM DESCRIPTIONS

The CASMO-4 / SIMULATE-3 computer program package consists of the following computer programs developed by Studsvik-Scandpower Inc.:

INTERPIN-3
CASMO-4
CMSLINK
SIMULATE-3

The CASMO-4 program package is described below.

CASMO-4

CASMO-4 (Reference 7) is a multi-group, two-dimensional neutron transport theory lattice physics code with depletion capability and the ability to generate cross-sections and discontinuity factors for both boiling water reactor (BWR) and pressurized water reactor (PWR) diffusion theory core analysis. 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.

One new feature of CASMO-4 is the use of the Method of Characteristics for solving the Boltzmann neutron transport equation, which allows for modeling the true heterogeneous geometry directly in the two-dimensional transport calculation. The Method of Characteristics has been well established as a methodology particularly well suited to these types of problems and is now the solution method of choice for most next generation lattice physics codes: CASMO-5, HELIOS-2, APOLLO-2, LANCER-02, and WIMS-10.

The CASMO-4 transport method has been well validated in peer reviewed publications (e.g. References 16, 17).

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 performance error that may arise from the manual construction of the case matrix as was required with CASMO-3. Table 2.1 lists some of the major differences (and similarities) between CASMO-3 and CASMO-4.

As can be seen in Table 2.1, CASMO-4 shares some common heritage with CASMO-3 but also has some key methodological differences that are essentially transparent to the end user, e.g., the 2D transport method implemented. As both codes share the same base ENDF cross section evaluation, built-in biases due to library data should be similar between the two codes.

One of the key similarities between CASMO-3 and CASMO-4 is use of identical input and input preparation which means that training and skills relevant to proficiency at CASMO-3 directly translate into proficiency at CASMO-4. Many of the methodological improvements of CASMO-4 over CASMO-3 are transparent to the end user with the end result that CASMO-4 is intrinsically no more difficult to use than CASMO-3.

CMSLINK

CMSLINK (Reference 8) is a data processing program that links CASMO-4 to SIMULATE-3. The primary purpose of CMSLINK is to read the CASMO-4 ASCII card image file, functionalize key neutronic variables versus important independent variables and produce a binary master cross-section library for use with SIMULATE-3.

The CMSLINK program processes the following types of data from CASMO-4:

Two-group Macroscopic Cross-sections
Assembly Discontinuity Factors (ADF's)
Fission Product Data (fission yields and microscopic cross sections)
Incore Instrument Response Data (detector data)
Pin Power Reconstruction Data
Kinetics Data (beta's, lambda's and neutron velocities)
Spontaneous Fission Data/Alpha-n Sources
Decay Heat Data

Neutronic data is generated by performing CASMO-4 fuel assembly depletions (assuming zero leakage) and branches to off-nominal conditions. These calculations provide all the data needed to functionalize neutronic parameters versus:

- Fuel Burnup (EXP)
- Coolant Density or Temperature (DEN)
- Fuel Temperature (TFU)
- Boron Concentration (BOR)
- Control Rod Type (CRD)

- Historical Coolant Density (HDEN)
- Historical Rod Presence (HCRD)
- Historical Boron Concentration (HBOR)
- Historical Fuel Temperature (HTFU)

The data from CASMO-4 is then functionalized by CMSLINK into a series of 1-D, 2-D, and 3-D table sets (for each unique fuel type). This functionalization allows linear interpolation in SIMULATE to accurately evaluate neutronic data for each node in the core for the appropriate reactor conditions.

The functionalizations made in CMSLINK depend upon the amount of detailed branch and depletion data generated in CASMO-4. Normally, several branch cases are made for each independent variable, and simultaneous branches are made for certain variables such as coolant density and control rod. CASMO-4 contains an automated case matrix feature that allows this process to proceed in a standardized fashion.

INTERPIN

INTERPIN-3 (Reference 9) is used to calculate fuel pin temperatures. SCE uses this computer program to calculate the fuel temperature of the average rod as a function of burnup. Output from this computer program provides a single burnup independent fuel rod temperature for use in CASMO-4, and a power and burnup dependent fuel rod temperature for use in SIMULATE-3.

SIMULATE-3

SIMULATE-3 (Reference 10) is a two-group, three-dimensional (3-D), coarse mesh nodal diffusion theory reactor simulator computer program that employs both thermal-hydraulic and Doppler feedback. In the axial direction, 12-25 nodes are typically used to represent the active portion of each fuel assembly, and one node is used to represent the upper and lower reflectors.

SIMULATE-3 explicitly models the baffle/reflector region, eliminating the need to normalize to higher-order fine mesh calculations such as PDQ (a diffusion theory computer program). Homogenized cross-sections and discontinuity factors are applied to the coarse mesh nodal model to solve the two-group diffusion equation using the QPANDA neutronics model. QPANDA employs fourth order polynomial representations of the intra-nodal flux distributions in both the fast and thermal groups.

The nodal thermal-hydraulic properties are calculated based on the inlet temperature, RCS pressure, coolant mass flow rate, and the heat addition along the channel.

The pin-by-pin power distributions, on a 2-D or 3-D basis, are constructed from the inter- and intra-assembly information from the coarse mesh solution and the pin-wise assembly power distribution from CASMO-4.

SIMULATE-3 performs a macroscopic depletion and individual uranium, plutonium, and fission product isotope concentrations are not computed. However, microscopic depletion of iodine, xenon, promethium, and samarium is included to allow modeling of typical reactor transients.

2.2 MODEL DESCRIPTIONS

CASMO-4 Fuel Assembly And Reflector Models

Each unique SONGS fuel assembly type (defined by geometry, enrichment, and burnable poison) is separately modeled in CASMO-4 using octant symmetry. Enrichment zoning among fuel rods, burnable absorbers (separate rods and mixed with the UO_2 fuel) and guide tubes are explicitly modeled. Any water gap between fuel assemblies in the core is included in the CASMO-4 model. The spacer grids are also modeled. Design basis documents such as the SONGS Reload Ground Rules and as-built drawings provide the necessary data to develop the CASMO-4 assembly models.

Several depletions are typically needed to generate each fuel assembly type's cross-section data. First, the fuel assembly is depleted at hot full power, reactor average conditions (base depletion). Moderator temperature, fuel temperature, and soluble boron concentration are set to constant average values for the complete depletion. The average fuel temperature at hot full power conditions is calculated with INTERPIN-3. Next the fuel assembly is depleted at a lower moderator temperature, typically TMO-20 K, (moderator temperature history depletion). However, the fuel temperature and the soluble boron concentration are kept at the constant hot full power, reactor average values. The fuel assembly is again depleted at constant hot full power, reactor average conditions, but with a constant soluble boron concentration higher than is usually seen in normal operation, typically 2*BOR where BOR is the boron at base conditions, (boron history depletion). A fuel temperature history depletion is then performed at TFU=TMO-20 K, and finally an optional rodded depletion can be performed. Each fuel assembly is depleted up to 60 GWD/T assembly average burnup using the CASMO-4 default depletion steps.

Branch cases are performed to calculate instantaneous effects. These instantaneous effects are individually calculated and added together later to recreate the proper fuel assembly cross-sections. The branch cases are executed from the hot full power reactor average conditions depletion case at typically 0, 5, 10, 20, 30, 40, 50, and 60 GWD/T. Branch cases are run for off-nominal moderator temperatures, fuel temperatures, soluble boron concentrations, and control rod insertions to encompass the full range of conditions expected during reactor startup and operation.

A single CASMO-4 case performs all of the above depletions and branch cases. Furthermore, if gadolinia is the burnable absorber, no auxiliary program is needed as the depletion of gadolinia is modeled directly in CASMO-4.

CASMO-4 also generates explicit top, bottom, and radial reflector cross-sections (which are important for PWR modeling). The radial reflector consists of the stainless steel core baffle

followed by about 15 centimeters (cm) of water. The top reflector extends from the top of the active fuel to the lower surface of the fuel assembly upper end fitting. The bottom reflector extends from the bottom of the active fuel to the lower surface of the core support plate. Reflector cross-sections are typically modeled as a function of soluble boron concentration and moderator temperature. CASMO-4 performs reflector calculations in 25 energy groups to accurately capture the effects of high energy leakage.

CMSLINK Model

The CMSLINK computer program generates two-group fuel assembly and reflector cross-section tables for SIMULATE-3. For each fuel assembly type and reflector type, data from the CASMO-4 card-image file are processed into a binary cross-section library for input to SIMULATE-3.

SIMULATE-3 Model

The SIMULATE-3 model typically divides the active fuel region into 20 axial nodes and four radial nodes per fuel assembly. A pseudo-assembly, consisting of reflector material, surrounds the core and is divided into one radial and 20 axial nodes. Axially, the fuel is divided into a single bottom reflector node, 20 active fuel nodes, and a single top reflector node.

Additional model input data are:

- Full Core Fuel Assembly Serial Number Map
- Quarter Core Fuel Assembly Type Map
- Fuel Assembly Axial Zone Definition, Including Reflectors
- Cross-Section Library Assignment To Fuel Assembly And Reflector Types
- Control Rod Locations
- Grouping of Control Rods Into Banks
- Axial Zone Definitions For Control Rods, especially Part Length Control Rods
- In-Core Instrument Locations
- Fuel Temperature versus Power and Burnup Correlation (INTERPIN-3 Program)
- Core Mw-Thermal Output At 100% Power
- Core Pressure And Coolant Mass Flow Rate At 100% Power
- Core Inlet Temperature Versus Power Level
- Input Restart File (Cycle N-1) And Cross-Section Library File
- Output Restart File

After the cycle base model is set up, the physics analyst can specify the percent power level, control rod bank positions (percent withdrawn), output and edit options, and the type of calculation: depletion, xenon transient, coefficient calculation (e.g. ITC, IBW, FTC, etc.).

TABLE 2.1

CASMO-3 to CASMO-4 Code Feature Comparison

	CASMO-3	CASMO-4
Differences between CASMO-3 and CASMO-4		
Library Data Evaluation	ENDF/B-IV	ENDF/B-IV
# of energy groups in neutron data library	40 or 70	70
# of nuclides in neutron data library	93	103
2D transport method	Transmission probabilities	Method of Characteristics
Geometry	homogeneous	heterogeneous
Gd depletion	External code	Internal depletion
Case Matrix	Manual Input	Automated Case Matrix
Default # 2D energy groups (non-Ref)	7	8
Default # 2D energy groups (Ref)	7	25
Similarities between CASMO-3 and CASMO-4		
Pin Cell Calculation	Collision Probabilities	
Resonance Calculation	Equivalence Theory	
Burnup Calculation	Predictor/Corrector	
Input	Simple Engineering Data –Identical for CASMO-3 and CASMO-4	
Thermal Expansion	Yes	
Generates 2 Group Data for SIMULATE-3	Yes	

FIGURE 2.1

Computer Program Sequence Flow Chart

INTERPIN

|
|
v

CASMO-4

|
|
v

CMSLINK

|
|
v

INTERPIN --------> SIMULATE-3

|
|
v

Assembly Power Distribution
Pin-By-Pin Power Distribution
Critical Boron Concentration
Cycle Length
Reactivity Coefficients
Etc.

SECTION 3

DESCRIPTION OF REACTORS USED IN BENCHMARKING

3.0 INTRODUCTION

This report compares the CASMO-4 / SIMULATE-3 predictions of key physics parameters with measured data and CASMO-3 / SIMULATE-3 results. The reactor plants are San Onofre Nuclear Generating Station Units 2 and 3. The measured data were obtained during plant startup and normal operation.

The following Section provides a brief description of San Onofre Units 2 And 3. Detailed information can be found in the SONGS Units 2 and 3 UFSAR (Reference 11).

3.1 SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3 (SONGS 2&3)

SONGS 2&3 are commercial nuclear power plants. SONGS 2 began commercial operation in 1983. SONGS 3 began commercial operation in 1984. SONGS 2 is in its 15th cycle of operation. SONGS 3 is in its 15th cycle of operation. SONGS 2&3 are Combustion Engineering (absorbed into Westinghouse) two-loop pressurized water reactors (PWRs). Each unit produces 3438 megawatts-thermal at 100% rated power. (The power has been uprated from 3390 Mw-t.)

Each reactor core contains 217 fuel assemblies arranged as shown in Figure 3.1. Both in-out and low-leakage fuel management patterns have been used. Each fuel assembly consists of a 16x16 array of 236 fuel rods and 5 control rod guide tubes (Figure 3.3). Core, fuel assembly, control rod, and burnable absorber data are summarized in Table 3.1.

The fuel rods are low enriched (< 4.8 w/o) UO_2 pellets clad in Zircaloy-4 or Zirlo. The control rod guide tubes are also Zircaloy-4. Ten Zircaloy-4 grids and one Inconel-718 grid are located along the length of the assembly.

The incore instrumentation system for power distribution measurement consists of 56 strings of fixed Rhodium detectors (Figure 3.2). Each detector string consists of five individual, 40 cm long, Rhodium detectors placed at about 15, 30, 50, 70, and 90 percent of active core height. The detector signals are processed of-line with the Combustion Engineering (now Westinghouse) CECOR (Reference 13) computer program to determine the power distribution in the core.

There are 83 full-length and eight part-length control rods, called control element assemblies (CEA's). Seventy-nine full length CEA's have 5 individual absorber rods. Four full-length CEA's located on the core periphery have 4 individual absorber rods. The full length CEA's have an Inconel nose cap, about 6" of AgInCd, and about 136" of B_4C pellets. The eight part length CEA's have five absorber rods consisting of approximately 75" of Inconel, 58" of water

filled Inconel tube, and 16" of B_4C pellets. The cladding material is Inconel-625. The CEA's are moved in nine symmetrical groups:

Regulating Groups 1 Through 6
Part Length Group
Shutdown Groups A and B

Burnable absorber rods, consisting of B_4C-Al_2O_3 pellets in Zircaloy-4 cladding, were used in Cycles 1 through 8 of both Units. Lead Fuel Assemblies (LFA's) with erbia (Er_2O_3) mixed with the UO_2 in the fuel rod were placed in the SONGS 2 reactor core during cycle 6. Cycle 9 of each Unit was a transition cycle consisting of a full, fresh reload batch with erbia as the burnable absorber, and a burned batch with depleted B_4C-Al_2O_3 absorber rods. Cycles 10 through the present have used erbia as the burnable absorber. Future cycles may employ:

(1) Gadolinia (Gd_2O_3) mixed with UO_2 in the fuel rod with Zircaloy-4, Zirlo, or M5 cladding, and/or,
(2) Integral Fuel Burnable Absorber (IFBA) – A ZrB_2 coating on the surface of the UO2 fuel pellet in Zircaloy-4, Zirlo, or M5 cladding.

The SONGS 2&3 reactors have several unique features. The outermost row of four fuel assemblies does not line up with the next interior row of fuel assemblies. The four-finger CEA inserted in the middle pair of these "off-set" assemblies has two fingers in one assembly and two fingers in the adjacent assembly. The B_4C-Al_2O_3 burnable absorber rods and erbia burnable absorber do not extend the full length of the active fuel region resulting in axially zoned fuel assemblies. [Future burnable absorbers such as Gadolinia or IFBA would also be axially zoned.] The cycle length of both Units has increased from 366 Effective Full Power Days (EFPD) for Cycle 1 to about 600 EFPD for Cycle 15. Finally, the five control rod guide tubes are large (displacing 4 fuel rods) compared to Westinghouse fuel (displacing 1 fuel rod) and Babcock & Wilcox fuel (displacing 1 fuel rod) fuel assembly designs.

TABLE 3.1

Mechanical Design Parameters
SONGS 2&3

Core Description

Power Level	3390 / 3438 Megawatts-Thermal
Number of Assemblies	217
Number Of Control Rods	91
Fuel Assembly Pitch	8.180 inches
Core Area	101.1 Square Feet
Core Equivalent Diameter	136 inches

Fuel Assembly Description

Fuel Rod Array	16 x 16
Fuel Rod Pitch	0.506 inches
Outside Dimension	7.972 inches
Number Of Guide Tubes	5
Guide Tube I.D.	0.90 inches
Guide Tube O.D.	0.98 inches
Guide Tube Material	Zircaloy-4

Fuel Rod Description

Material	UO_2
Maximum Enrichment	4.8 w/o U^{235}
Stack Height Density	10.06 -- 10.34 gm/cm^3
Pellet Diameter	0.3250 / 0.3255 inches
Clad Material	Zircaloy-4 and Zirlo
Clad I.D.	0.332 inches
Clad O.D.	0.382 inches
Clad Thickness	0.025 inches
Active Fuel Length	150 inches

Full-Length Control Rod Description

Number	83
5-Finger	79
4-Finger	4
Clad Material	Inconel-625
Clad Thickness	0.035 inches
Clad O.D.	0.816 inches

TABLE 3.1 (Continued)

Full-Length Control Rod (Continued)

Poison Material	B_4C / Ag-In-Cd / Inconel		
Poison Material Length			
5-Finger	136.0"	12.5"	0.6"
4-Finger	126.5"	12.5"	10.1"
B_4C Pellet Diameter	0.737 inches		
% T. D. Of 2.52 g/cm^3	73 %		
Weight % Boron, Minimum	77.5 %		

Part Length Control Rod Description

Number	8 (5-fingers)		
Clad Material	Inconel-625		
Clad Thickness	0.035 inches		
Clad O.D.	0.816 inches		
Poison Material	Inconel / Water / B_4C		
Poison Material Length	76.4"	55.0"	16.0"
B_4C Pellet Diameter	0.737 inches		
% T. D. Of 2.52 g/cm^3	73 %		
Weight % Boron, Minimum	77.5 %		

Burnable Poison Description

Absorber Material	Er_2O_3
$UO_2 - Er_2O_3$ T.D.	10.3 gm/cm^3
Number Per Fuel Assembly	0, 40, 48, 60, 72, and 80
Absorber Material	$Al_2O_3 - B_4C$
Pellet Diameter	0.310 inches
Pellet Length	0.50 inches min
Active Length	136 inches
Clad Material	Zircaloy-4
Clad I.D.	0.332 inches
Clad O.D.	0.382 inches

FIGURE 3.1

SONGS 2 And 3 Reactor Core With Control Rod Pattern

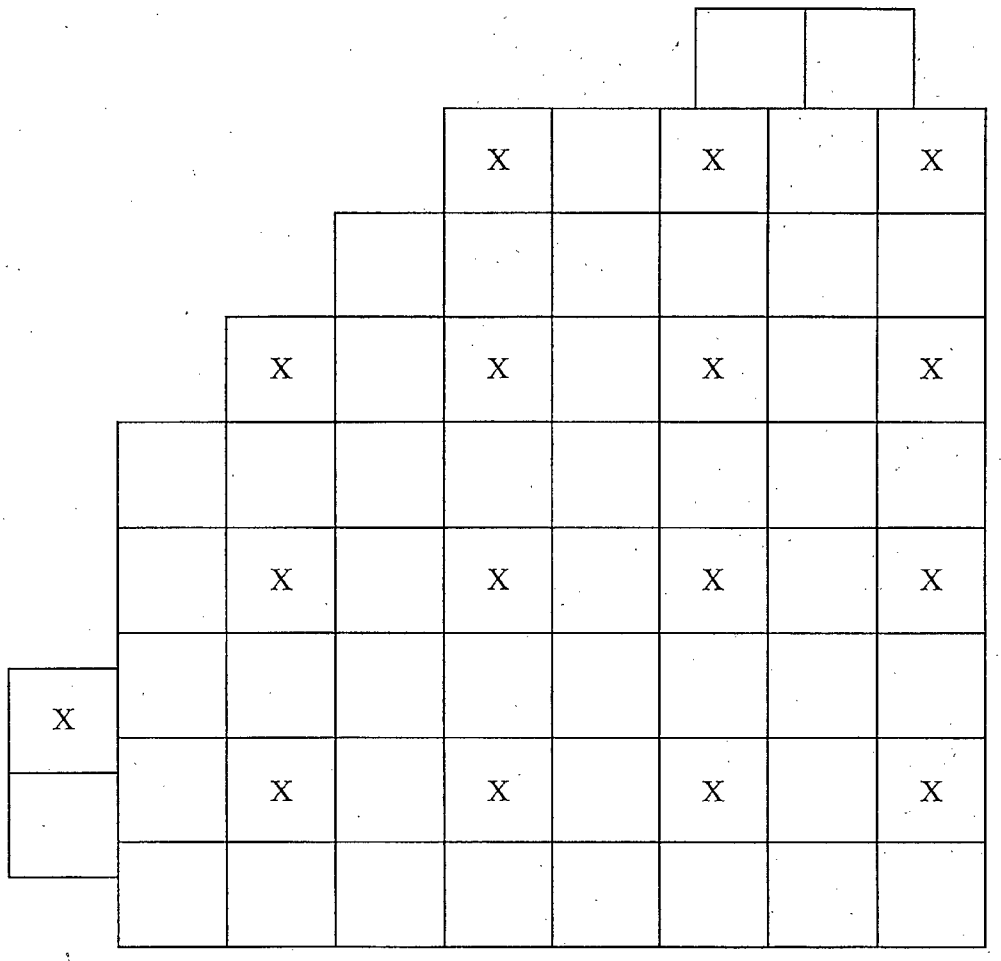
						A	A
					2		
			B		3		4
		1		A		B	
	B		5		P		6
		A		A		B	
	3		P		4		1
2		B		B		A *	
A A	4		6		1		2

Banks 1 - 6 are Regulating Banks.
Banks A and B are Shutdown Banks.
(AA Indicates a 4-Finger Control Rod spanning Two Assemblies on the core periphery.)
Bank P is the Part-Length Control Rods.

A*: If the quadrant shown is #1, this control rod is not present in quadrants #2 and #4.

FIGURE 3.2

SONGS 2 And 3 Reactor Core With Incore Instrument Pattern



X = Incore Instrument Location

FIGURE 3.3

SONGS 2 And 3 16x16 Fuel Assembly

```
X X X X X X X X X X X X X X X X     X X X X X X X X X X X X X X X X
X X X X X X X X X X X X X X X X     X X X X X X X X X X X X X X X X
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X X X       X X X X X X       X X X     X X X       X X X X X X       X X X
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```

<---------- > <---------- >

7.972" 0.208" 7.972"

<---------- >

8.180"

X = Fuel Rod

There are 5 control rod guide tubes. Each guide tube displaces 4 fuel rods.

Pellet Diameter	= 0.3255	inches
Clad I.D.	= 0.322	inches
Clad O.D.	= 0.382	inches
Fuel Rod Pitch	= 0.506	inches
Active Fuel Length	= 150	inches
Guide Tube I.D.	= 0.90	inches
Guide Tube O.D.	= 0.98	inches

SECTION 4

BENCHMARK COMPARISONS

4.0 INTRODUCTION

This section compares CASMO-4 / SIMULATE-3 calculated parameters with measured plant data. The measured data are from zero power startup testing, at-power Isothermal Temperature Coefficient measurements, and normal operations at SONGS 2&3. For each parameter compared, the sample mean and standard deviation of the observed differences are calculated. Based on the mean, standard deviation, and sample size, a bias and conservative 95/95 tolerance limit (bias ± uncertainty) are calculated.

Typically for this benchmarking, SONGS 2 and 3 cycles that have erbia as the burnable absorber and are most representative of the future and current fuel management strategy (checkerboard) and cycle length (~ 600 EFPD) will be used. These cycles are SONGS 2 Cycles 10 – 15 and SONGS 3 Cycles 10 – 14. However, the power coefficient has not been measured in these recent cycles. The power coefficient was measured in Cycles 1 - 3. Therefore, the Cycles 1 – 3 measurements will be analyzed with CASMO-4. The ITC measurement from Cycles 10 – 15 generally occurred around 2000 ppm. To obtain a greater range of soluble boron for the ITC measurements, Cycles 1-5 ITC measurements will be analyzed with CASMO-4.

Section 4.1 presents the Critical Boron Concentration (CBC) comparisons for zero power and hot full power conditions. Differences between calculated and measured data are represented in absolute terms (Measured – Calculated).

Section 4.2 presents the Isothermal Temperature Coefficient (ITC) comparisons for zero power and hot full power conditions. Differences between calculated and measured data are represented in absolute terms (Measured – Calculated).

Section 4.3 presents the Power Coefficient (PC) comparisons. Differences between calculated and measured data are represented in absolute terms (Measured – Calculated).

Section 4.4 presents the Control Rod Worth comparisons. Differences between calculated and measured data are represented in relative terms – 100% * (Measured – Calculated) / Calculated.

Section 4.5 verifies the ability of CASMO-4 / SIMULATE-3 to predict the net (N-1) control rod worth. This section is not a comparison against measured data. CASMO-4 / SIMULATE-3 results are compared against CASMO-3 / SIMULATE-3 results.

Section 4.6 presents the Inverse Boron Worth (IBW) comparisons. Differences are represented in relative terms.

4.1 CRITICAL BORON CONCENTRATION

CASMO-4 / SIMULATE-3 Critical Boron Concentrations (CBC) were compared to zero-power startup measurements and full power operating data. The most reliable measurements are the zero-power startup tests. These measurements are made under well controlled conditions without significant thermal and xenon feedbacks, and no boron depletion effects. The zero-power comparison statistics quantify SIMULATE-3's accuracy in predicting CBC for Beginning-Of-Cycle (BOC), zero-power conditions without xenon in the core.

The full-power operation boron concentration data are from titration of reactor coolant samples. All measurements are adjusted for control rod insertions and deviations from full-power equilibrium conditions. These full power comparisons serve as conservative estimates of the SIMULATE-3 uncertainties for at-power equilibrium conditions with thermal feedback.

Sections 4.1.1 and 4.1.2 present the comparisons for zero-power and full-power CBC.

4.1.1 ZERO-POWER CRITICAL BORON CONCENTRATION

Table 4.1 lists the measured and SIMULATE-3 predicted values for BOC, zero-power, xenon free Critical Boron Concentrations (CBC). Eleven measurements from the most recent eleven cycles of startup tests have been included. All measurements are at about 545 degrees F and essentially all control rods out.

A three-step statistical analysis was performed on the measured and SIMULATE-3 calculated CBC differences. First, the sample mean (x-bar), and standard deviation (S) were calculated for the CBC differences. The differences are due to SIMULATE-3 calculational uncertainties, variations in B-10 isotopic concentrations, and measurement (titration) uncertainties. For example, boron concentration measurement errors can be as high as 5 ppm. For conservatism, all differences are assumed due only to SIMULATE-3 calculational uncertainties.

Second, the sample distribution of differences is tested for normality using ANSI N15.15-1974 (Reference 14). The normality test is needed because the 95/95 tolerance limit assumes that the sample differences are normally distributed.

Finally the bias and 95/95 tolerance limit (Uncertainty) are calculated. The bias is equal to the sample mean (x-bar). The 95/95 tolerance limit is calculated by $K_{95/95} * S$, where S is the standard deviation and $K_{95/95}$ is determined from Reference 15.

Calculation of the CASMO-4 / SIMULATE-3 zero power CBC bias and 95/95 tolerance limit is shown in Table 4.1:

-1 ppm $\pm$ 23 ppm

4.1.2 HOT-FULL-POWER CRITICAL BORON CONCENTRATION

For this benchmarking effort, the most recent six cycles (SONGS 2 and 3 Cycles 12 – 14) were determined to be sufficient. There are a total of 111 comparisons. Therefore, Cycles 10 and 11 were not analyzed.

Tables 4.2 through 4.7 compare Cycles 12 - 14 CASMO-4 / SIMULATE-3 calculated Hot-Full-Power (HFP) CBCs with measured data taken during cycle operation.

The SIMULATE-3 HFP CBC and 95/95 tolerance limits were determined using the statistical methods outlined in Section 4.1.1.

Calculation of the CASMO-4 / SIMULATE-3 full power CBC bias and 95/95 tolerance limit is shown in Table 4.7:

30 ppm $\pm$ 20 ppm

4.2 ISOTHERMAL TEMPERATURE COEFFICIENT

The Isothermal Temperature Coefficient (ITC) is the change in the reactivity due to a 1 degree F change in the core average moderator and fuel temperature. Tables 4.8, 4.9, and 4.10 list the comparisons of zero-power and at-power calculated ITC's with measurements.

A statistical analysis has been performed on the ITC differences using the process outlined in Section 4.1.1.

4.2.1 HZP ITC

Calculation of the CASMO-4 / SIMULATE-3 zero-power ITC bias and uncertainty for the HZP ITC measured in SONGS 2 Cycles 10 – 15 and SONGS 3 Cycles 10-14 is shown in Table 4.8:

$$-0.15 \text{ pcm/}^{\circ}\text{F} \pm 0.36 \text{ pcm/}^{\circ}\text{F}$$

4.2.2 HZP AND AT-POWER ITC – CYCLES 1 - 5

The ITC measurement from Cycles 10 – 15 generally occurred around 2000 ppm. To obtain a greater range of soluble boron for the ITC measurements, Cycles 1-5 ITC measurements will be analyzed with CASMO-4.

Calculation of the CASMO-4 / SIMULATE-3 bias and uncertainty for the HZP and at-power ITC's measured in Cycles 1 – 5 is shown in Tables 4.9 and 4.10:

$$-0.46 \text{ pcm/}^{\circ}\text{F} \pm 2.50 \text{ pcm/}^{\circ}\text{F}$$

4.2.3 FINAL HZP AND HFP ITC

In order to expand the range of applicability, results from Cycles 1 – 5 and Cycles 10 – 15 were combined. The combined Cycles 1 - 5 and Cycles 10 - 15 results are not a normal distribution. However, the Cycles 1 – 5 results provide a conservative bias and uncertainty which covers all HZP and HFP conditions at a wide range of soluble boron concentrations.

Therefore, the final bias and uncertainty are: -0.5 ± 2.5 pcm / °F

4.3 POWER COEFFICIENT

The power coefficient is defined as the change in reactivity due to a change in the core power level. Since the power coefficient has not been measured for recent cycles, CASMO-4 / SIMULATE-3 power coefficient predictions were compared to measurements from Cycles 1, 2, and 3, and summarized in Table 4.11.

Due to the limited size of the database, a meaningful 95/95 tolerance limit could not be derived. However, all of the differences are within 2 pcm/%P, and the sample mean and standard deviation are -0.55 pcm/%P and 0.78 pcm/%P, respectively. Since the differences include both the calculational and the measurement uncertainties, a conservative 95/95 tolerance limit of 2 pcm/%P can be assumed based on sound engineering judgment.

Therefore, the CASMO-4 / SIMULATE-3 Power Coefficient bias and 95/95 tolerance limit is:

$$0 \text{ pcm/\%P} \pm 2 \text{ pcm/\%P}$$

4.4 CONTROL ROD WORTH

Tables 4.12 and 4.13 list the measured and the calculated control rod bank worths. The bank worths were measured during startup testing at BOC at zero-power conditions. Fifty-two (52) measurements from 10 cycles are shown. The STAR (Startup Test Activity Reduction) program (Reference 24) was implemented beginning with Unit 2 Cycle 15. Therefore, the Control Rod Bank worths were not measured in Cycle 15.

A statistical analysis has been performed on the control rod bank worth differences. The analysis determined the bias (mean), standard deviation, and normality of the difference distribution. The mean is 1.3% and the standard deviation is 4.67%.

The measurement uncertainty has two components: the measurement uncertainty and the calculational uncertainty. These two components are related to the observed uncertainty by:

$$S_{OBS}^2 = S_M^2 + S_C^2$$

The measurement uncertainty has been quantified in Reference 1. Since SONGS 2&3 Cycle 1 were duplicate plants (identical fuel management, enrichments, burnable absorber worth, etc) one would expect the measured control rod worths at the beginning of the first cycle to be exactly the same. Therefore, the observed differences in Cycle 1 measurements is attributable to the measurement uncertainty. In Reference 1, Section 4.4, the standard deviation (S_D) of the differences in SONGS 2&3 Cycle 1 measured control rod worths was determined to be 4.0 %. Therefore, the net measurement uncertainty is:

$$\text{Measurement Uncty} = S_M^2 = ½ * S_D^2 = 8.0\%$$

Once the measurement uncertainty is quantified, the control rod worth calculational uncertainty is:

$$S_C = [\, S_{OBS}^2 - S_M^2 \,]^{1/2}$$

$$= [\, (4.67)^2 - (8.00) \,]^{1/2}$$

$$= 3.72\ \%$$

Finally, the 95/95 tolerance limit is: 95/95 Tolerance Limits = $K_{95/95} * S_C$

Substituting the appropriate values into the above formula as shown in Table 4.13, the CASMO-4 / SIMULATE-3 control rod worth bias and 95/95 tolerance limit is:

$$1.3\ \% \pm 7.6\ \%$$

The bias and tolerance limit will be applied to the SIMULATE-3 calculation of control rod worths at all power and moderator temperature conditions.

4.5 NET (N-1) CONTROL ROD WORTH

The net (N-1) control rod worth is defined as the reactivity worth of the insertion of all of the control rods except the most reactive control rod, which remains stuck out. Due to the high peaking in the assembly in which the stuck control rod is located, this configuration represents one of the most severe challenges to any reactor physics analysis method.

CASMO-3 / SIMULATE-3 capabilities in predicting the net control rod worth and the worst stuck rod worth was verified by predicting the measurement performed during the initial startup of Arkansas Nuclear One – Unit 2 (ANO-2) (Reference 1, Section 4.5).
The CASMO-3 / SIMULATE-3 prediction compared well with the measured data. It was concluded that the bias and 95/95 tolerance limit for control rod bank worth (Section 4.4) was applicable to the net (N-1) worth also.

In Table 4.14, CASMO-3 /SIMULATE-3 and CASMO-4 /SIMULATE-3 results for total worth, stuck rod worth, and (N-1) worth for SONGS Unit 2 Cycle 14 are compared. Differences are less than 1.4% and well within the 95/95 tolerance limit for control rod bank worth (Section 4.4).

Therefore, it is concluded that the bias and 95/95 tolerance limit for control rod bank worth (Section 4.4) is applicable to the net (N-1) worth also.

4.6 INVERSE BORON WORTH

The Inverse Boron Worth (IBW) is defined as one over (the inverse) the reactivity worth of the soluble boron dissolved in the reactor coolant – ppm / % $\Delta\rho$.

The measured IBW is calculated using:

$$IBW = -(CBC_1 - CBC_2) / (\%\Delta\rho \text{ from Control Rod Insertion / Withdrawl})$$

where CBC_1 , CBC_2 , and $\%\Delta\rho_{Control\ Rods}$ are measured.

The measurement uncertainty includes boron titration errors and control rod worth measurement errors.

Calculated and measured Inverse Boron Worths are compared in Table 4.15.
A mean (bias) and standard deviation have been calculated based on all the data.
The distribution is normal. Finally, a 95/95 tolerance factor was determined.

The CASMO-4 / SIMULATE-3 IBW bias and uncertainty are:

-3.9 % $\pm$ 13.6 %

NOTES: (1) The IBW measurement in Unit 3 Cycle 12 was identified as bad and thus not included in the benchmarking.

(2) With implementation of the STAR program (Reference 24), the IBW was not measured in Cycle 15.

TABLE 4.1

HZP Critical Boron Concentration
Beginning Of Cycle , Essentially All Rods Out

Unit	Cycle	Measurement (ppm)	C-4/S-3 Prediction (ppm)	Difference (ppm)*
2	10	1992	2005	-13
2	11	1936	1935	1
2	12	2043	2034	9
2	13	2065	2061	4
2	14	2063	2071	-8
2	15	2148	2139	9
3	10	2018	2021	-3
3	11	2052	2044	8
3	12	2115	2127	-12
3	13	2100	2105	-5
3	14	2093	2091	2
			Bias =	-1

* Measurement - Prediction

CASMO-4 Uncertainty -- Normality Test Result = Normal
Sample Size = 11
Degrees Of Freedom = 10
$K_{95/95} = 2.815$
Standard Deviation (S) = 8 ppm
95/95 Uncertainty ($K_{95/95}$ * S) = 23 ppm

TABLE 4.2

SONGS 2 Cycle 12 Critical Boron Concentration -- HFP

	Power	EFPD	Burnup (GWD/T)	MEASURED PPM	SIMULATE-3 PPM	M - S3 Delta-PPM
S2C12	100	20.35	0.738	1505	1477	28
	100	47.33	1.716	1463	1432	31
	100	75.13	2.724	1418	1384	34
	100	103.90	3.767	1360	1325	35
	100	127.49	4.622	1293	1271	22
	100	157.34	5.704	1220	1198	22
	100	183.27	6.644	1160	1131	29
	100	234.39	8.498	996	991	5
	100	265.25	9.616	914	903	11
	100	292.11	10.590	835	825	10
	100	319.97	11.600	759	742	17
	100	373.40	13.537	614	580	34
	100	401.26	14.547	523	496	27
	100	429.09	15.556	440	410	30
	100	456.88	16.564	359	325	34
	100	512.83	18.592	193	154	39
	100	540.78	19.605	106	70	36
	100	568.73	20.619	22	-14	36

TABLE 4.3

SONGS 2 Cycle 13 Critical Boron Concentration -- HFP

	Power	EFPD	Burnup (GWD/T)	MEASURED PPM	SIMULATE-3 PPM	M - S3 Delta-PPM
S2C13	100	26.0	0.942	1499	1466	33
	100	53.8	1.949	1444	1419	25
	100	84.7	3.068	1386	1360	26
	100	114.2	4.136	1330	1297	33
	100	145.3	5.263	1253	1223	30
	100	179.3	6.494	1169	1136	33
	100	206.2	7.469	1087	1064	23
	100	230.0	8.331	1032	998	34
	100	264.7	9.587	928	899	29
	100	290.7	10.529	857	824	33
	100	320.0	11.590	756	738	18
	100	348.0	12.605	685	654	31
	100	373.5	13.528	608	577	31
	100	401.2	14.531	530	493	37
	100	428.9	15.535	443	408	35
	100	456.7	16.542	358	323	35
	100	484.4	17.545	274	239	35
	100	512.0	18.545	188	156	32

TABLE 4.4

SONGS 2 Cycle 14 Critical Boron Concentration -- HFP

	Power	EFPD	Burnup (GWD/T)	MEASURED PPM	SIMULATE-3 PPM	M - S3 Delta-PPM
S2C14	100	44.07	1.595	1439	1430	9
	100	64.73	2.343	1406	1396	10
	100	92.33	3.342	1357	1341	16
	100	99.23	3.592	1344	1326	18
	100	127.01	4.598	1281	1263	18
	100	154.73	5.601	1211	1195	16
	100	182.46	6.605	1142	1123	19
	100	216.52	7.838	1053	1031	22
	100	244.97	8.868	975	951	24
	100	279.34	10.112	879	853	26
	100	307.12	11.118	786	771	15
	100	334.86	12.122	709	689	20
	100	362.65	13.128	629	605	24
	100	397.26	14.381	524	501	23
	100	415.75	15.050	470	445	25
	100	441.03	15.965	394	368	26
	100	475.40	17.209	290	263	27
	100	503.10	18.212	212	179	33
	100	523.91	18.966	145	117	28
	100	556.80	20.156	47	19	28

TABLE 4.5

SONGS 3 Cycle 12 Critical Boron Concentration -- HFP

	Power	EFPD	Burnup (GWD/T)	MEASURED PPM	SIMULATE-3 PPM	M - S3 Delta-PPM
S3C12	100	43.23	1.567	1499	1487	12
	100	77.12	2.796	1447	1427	20
	100	104.00	3.771	1395	1371	24
	100	134.83	4.889	1326	1298	28
	100	166.90	6.051	1241	1217	24
	100	194.87	7.065	1158	1143	15
	100	222.83	8.079	1089	1065	24
	100	252.90	9.169	1011	979	32
	100	281.55	10.208	923	895	28
	100	314.29	11.395	815	797	18
	100	334.27	12.120	756	737	19
	100	369.49	13.397	659	628	31
	100	406.41	14.735	556	514	42
	100	434.39	15.750	474	427	47
	100	464.79	16.852	374	333	41
	100	493.78	17.903	299	243	56
	100	522.71	18.952	213	154	59
	100	549.55	19.925	133	73	60

TABLE 4.6

SONGS 3 Cycle 13 Critical Boron Concentration -- HFP

	Power	EFPD	Burnup (GWD/T)	MEASURED PPM	SIMULATE-3 PPM	M - S3 Delta-PPM
S3C13	100	16.2	0.587	1554	1524	30
	100	43.3	1.570	1501	1475	26
	100	71.3	2.585	1461	1425	36
	100	99.1	3.593	1398	1368	30
	100	127.7	4.629	1322	1301	21
	100	148.7	5.391	1287	1249	38
	100	176.6	6.402	1213	1177	36
	100	204.8	7.424	1132	1100	32
	100	232.5	8.429	1060	1023	37
	100	260.5	9.444	980	942	38
	100	288.5	10.459	901	860	41
	100	316.3	11.467	820	777	43
	100	343.8	12.463	740	694	46
	100	371.8	13.478	651	608	43
	100	399.8	14.494	568	523	45
	100	427.7	15.505	482	437	45
	100	455.3	16.506	397	352	45

TABLE 4.7

SONGS 2 Cycle 14 Critical Boron Concentration -- HFP

	Power	EFPD	Burnup (GWD/T	MEASURED PPM	SIMULATE-3 PPM	M - S3 Delta-PPM
S3C14	100	7.86	0.285	1549	1509	40
	100	39.95	1.448	1466	1444	22
	100	67.94	2.463	1421	1397	24
	100	95.83	3.474	1370	1341	29
	100	123.76	4.486	1311	1279	32
	100	156.57	5.676	1228	1199	29
	100	184.73	6.696	1159	1127	32
	100	212.54	7.704	1089	1052	37
	100	247.43	8.969	993	956	37
	100	275.39	9.983	917	876	41
	100	306.49	11.110	821	786	35
	100	334.47	12.124	737	703	34
	100	369.40	13.390	634	599	35
	100	396.01	14.355	556	519	37
	100	423.89	15.366	474	434	40
	100	437.90	15.873	429	392	37
	100	456.57	16.550	384	336	48
	100	484.22	17.552	290	252	38
	100	519.15	18.819	188	147	41
	100	546.00	19.792	107	67	40

Bias from 111 measurements = 30 ppm

CASMO-4 Uncertainty -- Normality Test Result = Normal
Sample Size = 111
Degrees Of Freedom = 110
$K_{95/95}$ = 1.911
Standard Deviation (S) = 10.3 ppm
95/95 Uncertainty ($K_{95/95}$ * S) = 20 ppm

TABLE 4.8

HZP Isothermal Temperature Coefficient – Cycles 10 - 15
Beginning of Cycle, Essentially All Rods Out

Unit	Cycle	Boron (ppm)	Measurement (pcm/°F)	C-4/S-3 Prediction (pcm/°F)	M – S3 (pcm/°F)
2	10	1992	-1.84	-1.81	-0.03
2	11	1931	-2.16	-2.06	-0.10
2	12	2035	-1.69	-1.63	-0.06
2	13	2060	-1.44	-1.53	0.09
2	14	2063	-1.41	-1.10	-0.31
2	15	2144	-1.16	-0.84	-0.32
3	10	2020	-1.98	-1.88	-0.10
3	11	2047	-1.79	-1.66	-0.13
3	12	2118	-1.09	-0.91	-0.18
3	13	2100	-1.42	-1.19	-0.23
3	14	2090	-1.72	-1.44	-0.28
				Bias =	-0.15

CASMO-4 Uncertainty -- Normality Test Result = Normal
Sample Size = 11
Degrees Of Freedom = 10
$K_{95/95} = 2.815$
Standard Deviation (S) = 0.13 pcm/°F
95/95 Uncertainty ($K_{95/95}$ * S) = 0.36 pcm/°F

TABLE 4.9

HZP Isothermal Temperature Coefficient – Cycles 1 - 5
Beginning of Cycle

Unit	Cycle	CBC (PPM)	Temp (Deg F)	CEA Position	MEAS ITC (pcm/°F)	CASMO4 ITC (pcm/°F)	M-C4 ITC (pcm/°F)
2	1	869	320	ARO	-1.43	-1.90	0.47
		797	320	Bk 6-4 In	-3.46	-4.26	0.80
		833	545	ARO	-3.80	-4.21	0.41
2	2	1198	545	ARO	0.75	2.65	-1.90
		883	545	Bk 6-1 In	-9.14	-7.48	-1.66
2	3	1580	545	ARO	0.50	1.97	-1.47
		1382	545	Bk B In	-5.88	-5.51	-0.37
2	4	1803	545	ARO	0.77	1.49	-0.72
		1563	545	Bk B In	-3.64	-3.95	0.31
2	5	1620	545	ARO	-0.82	-0.49	-0.33
		1208	545	Bk 6-1 In	-8.60	-9.39	0.79
3	1	823	545	ARO	-4.50	-4.35	-0.15
		484	545	Bk 6-1 In	-15.12	-15.34	0.22
3	2	1175	545	ARO	0.52	2.34	-1.82
		968	545	Bk B In	-5.70	-4.88	-0.82
3	3	1550	545	ARO	0.43	1.57	-1.14
		1369	545	Bk B In	-6.13	-5.75	-0.38
3	4	1822	545	ARO	1.13	1.81	-0.68
		1403	545	Bk 6-1 In	-6.60	-6.80	0.20

These HZP results are combined with the HFP results in the following Table 4.10 for a total of 41 measurements for the statistical analysis.

TABLE 4.10

At-Power Isothermal Temperature Coefficient – Cycles 1 - 5

Unit	Cycle	Power (%)	Burnup (GWD/T)	CBC (PPM)	MEAS ITC (pcm/°F)	CASMO-4 ITC (pcm/°F)	M-C4 ITC (pcm/°F)
2	1	20	0.103	660	-6.28	-6.39	0.11
		50	0.539	559	-8.24	-8.13	-0.11
		80	1.250	512	-9.42	-9.43	0.01
		100	2.050	483	-10.37	-10.75	0.38
		100	9.180	287	-16.47	-15.16	-1.31
2	2	98	0.208	818	-7.30	-7.41	0.11
		100	1.466	693	-12.50	-10.11	-2.39
		100	6.650	268	-22.30	-19.81	-2.49
		100	8.123	145	-25.42	-22.69	-2.73
2	3	100	0.380	818	-7.81	-7.54	-0.27
		100	1.336	693	-9.23	-8.66	-0.57
		100	10.202	268	-19.20	-21.05	1.85
		100	12.762	145	-23.00	-25.15	2.15
2	5	100	1.464	1063	-9.83	-10.55	0.72
3	1	50	0.288	540	-8.26	-8.89	0.63
		98	1.360	471	-10.72	-11.31	0.59
		100	9.067	277	-14.78	-15.11	0.33
3	2	50	0.150	893	-5.59	-3.62	-1.97
		89	0.378	758	-10.84	-7.80	-3.04
3	3	100	1.447	991	-9.64	-9.19	-0.45
		100	9.867	367	-22.20	-20.72	-1.48
3	4	100	1.520	1255	-8.23	-7.38	-0.85

Bias from 41 measurements = -0.46 pcm / °F

CASMO-4 Uncertainty -- Normality Test Result = Normal
Sample Size = 41
Degrees Of Freedom = 40
$K_{95/95} = 2.118$
Standard Deviation (S) = 1.18 pcm / °F
95/95 Uncertainty ($K_{95/95}$ * S) = 2.50 pcm / °F

TABLE 4.11

Power Coefficients
SONGS Units 2&3 – Cycles 1, 2, And 3

Unit	Cycle	Power %	Burnup MWD/T	Boron Ppm	Measured Pow Coeff (pcm/%ΔP)	CASMO-4 Pow Coeff (pcm/%ΔP)	(M - C4) (pcm/%ΔP)
2	1	50	539	559	-11.04	-10.06	-0.98
2	1	80	1250	512	-9.46	-9.61	0.15
2	1	100	2050	483	-9.47	-9.06	-0.41
2	2	98	208	818	-9.90	-9.38	-0.52
2	3	100	380	1095	-11.03	-9.01	-2.02
3	1	50	288	540	-10.41	-10.04	-0.37
3	1	100	1360	471	-8.93	-9.25	0.32
						Bias =	-0.55
						Standard Deviation =	0.78

TABLE 4.12

Control Rod Bank Worths – Unit 2
BOC, HZP, Measured By Boration/Dilution

Unit	Cycle	CEA Group	Measured Worth (pcm)	C-4/S-3 Predicted Worth (pcm)	100*(M-P) / P (%)
2	10	5	316	334	-5.4
		4	615	667	-7.9
		3	752	720	4.4
		2	367	369	-0.6
2	11	5	385	380	1.2
		4	634	641	-1.1
		3	681	639	6.6
		2	389	372	4.5
2	12	5	394	381	3.3
		4	759	754	0.7
		3	863	808	6.7
		1	551	559	-1.5
		6	401	413	-3.0
		A	1369	1333	2.7
2	13	5	305	298	2.4
		4	626	651	-3.9
		3	893	841	6.2
		2	409	381	7.2
		6	359	372	-3.4
		B	1812	1798	0.8
2	14	5	356	340	4.7
		4	709	727	-2.5
		3	786	717	9.6
		1	708	711	-0.4
		6	385	403	-4.6
		A	1414	1356	4.3

TABLE 4.13

Control Rod Bank Worths – Unit 3
BOC, HZP, Measured By Boration/Dilution

Unit	Cycle	CEA Group	Measured Worth (pcm)	C-4/S-3 Predicted Worth (pcm)	100*(M-P) / P (%)
3	10	5	401	407	-1.4
		4	640	703	-8.9
		3	786	762	3.2
		2	393	377	4.1
3	11	5	399	390	2.4
		4	686	708	-3.1
		3	815	761	7.1
		2	378	362	4.3
3	12	5	317	306	3.6
		4	712	734	-3.0
		3	863	777	11.1
		1	527	562	-6.2
		6	386	399	-3.4
		A	1323	1291	2.5
3	13	5	337	327	3.0
		4	684	687	-0.4
		3	812	738	10.0
		2	434	396	9.7
		6	369	371	-0.4
		B	1698	1675	1.3
3	14	5	311	307	1.3
		4	670	687	-2.4
		3	824	781	5.5
		1	538	550	-2.2
		6	383	397	-3.6
		A	1451	1403	3.4

For All Unit 2 and Unit 3 CEA Bank Worth Measurements:
CASMO-4 Uncertainty -- Normality Test Result = Normal
Sample Size = 52
Degrees Of Freedom = 51
$K_{95/95} = 2.055$
Bias (Mean) = 1.3%
Standard Deviation (S) = 4.67 %
Measurement Uncty: $S_M^2 = 8.00\%$
$S_C = [\,(4.67)^2 - 8.0\,]^{1/2} = 3.72\%$
95/95 Uncertainty ($K_{95/95} * S_C$) = 7.6 %

TABLE 4.14

Net (N-1) Control Rod Worth
CASMO-3 And CASMO-4 Comparison For Unit 2 Cycle 14

	CASMO-3 k-eff	CASMO-3 Worth (pcm)	CASMO-4 k-eff	CASMO-4 Worth (pcm)	
ARO	0.99780	--	1.00000	--	
ARI	0.92439	--	0.92543	--	
Rod 1	0.92550	130	0.92659	135	
Rod 2	0.92656	253	0.92770	264	
Rod 3	0.92658	256	0.92770	264	
Rod 4	0.92584	169	0.92693	175	
Rod 5	0.92489	58	0.92586	50	
Rod 6	0.92613	203	0.92724	211	
Rod 7	0.92680	281	0.92793	291	
Rod 8	0.92686	288	0.92794	292	
Rod 9	0.92573	157	0.92665	142	
Rod 10	0.92656	253	0.92769	263	
Rod 11	0.92765	380	0.92858	367	
Rod 12	0.92794	414	0.92909	426	
Rod 13	0.92799	420	0.92895	409	
Rod 14	0.92637	231	0.92746	237	
Rod 15	0.92687	289	0.92773	268	
Rod 16	0.92529	105	0.92631	103	
					% Difference
Total	Worth =	7959		8058	-1.2
Worst Stuck	Worth =	420		426	-1.4
(N-1)	Worth =	7539		7632	-1.2

TABLE 4.15

Inverse Boron Worth
BOC, Hot Zero Power

Unit	Cycle		MEAS IBW (ppm/%Rho)	CASMO4 IBW (ppm/%Rho)	M-C4 IBW (PCT Diff)
2	10	HZP	135.6	140.1	-3.2
2	11	HZP	129.0	137.2	-6.0
2	12	HZP	138.7	141.0	-1.7
2	13	HZP	149.0	142.5	4.6
2	14	HZP	138.7	143.1	-3.0
3	10	HZP	139.8	141.4	-1.2
3	11	HZP	127.3	142.5	-10.6
3	13	HZP	136.0	143.7	-5.3
3	14	HZP	132.0	144.9	-8.9
				Bias =	-3.9 %

CASMO-4 Uncertainty -- Normality Test Result = Normal
Sample Size = 9
Degrees Of Freedom = 8
$K_{95/95} = 3.031$
Standard Deviation (S) = 4.5 %
95/95 Uncertainty ($K_{95/95}$ * S) = 13.6 %

SECTION 5

POWER DISTRIBUTION AND PIN PEAKING COMPARISONS

5.0 INTRODUCTION

The ability of CASMO-4 / SIMULATE-3 (C-4/S-3) to accurately calculate core radial and axial power distributions and pin peaking factors (Fq, Fxy, Fr) is verified by comparison with CASMO-3 / SIMULATE-3 (C-3/S-3) results.

If C-4/S-3 and C-3/S-3 core radial and axial power distributions, and pin peaking factors (Fq, Fxy, Fr) are essentially the same, the C-3/S-3 assembly and fuel pin peaking biases and uncertainties from Table 1.2 of Reference 1 also apply to C-4/S-3.

Section 5.1 compares local pin power.

Section 5.2 compares pin peaking factors (Fq, Fxy, Fr).

Section 5.3 compares core radial and axial power distributions.

Section 5.4 compares the axial shape index.

5.1 LOCAL PIN POWER

The local pin power bias and uncertainty is determined by modeling industry standard critical experiments in which the individual fuel rod power distribution has been measured.

This bias and uncertainty were determined by analysis of the RPI critical experiments (Reference 2) with and without erbia and B&W critical experiments (Reference 2) with and without gadolinia.

The RPI core configurations calculated with CASMO-4 / DOT had 0, 20, 44, and 56 erbia pins. A mean and standard deviation were calculated for each core configuration, and for the pooled Data. The mean was negligibly small and negative. Therefore it was assumed to be 0.

From Reference 2, the Local Pin Power bias and uncertainty to be used for SONGS 2 and 3 are:

0 % $\pm$ 1.78 %

5.2 PIN PEAKING FACTORS (Fq, Fxy, Fr)

This section compares CASMO-4/SIMULATE-3 (C-4/S-3) calculated pin peaking factors (Fq, Fxy, and Fr) with CASMO-3/SIMULATE-3 (C-3/S-3) calculated results.

SONGS 2&3 Cycles 12, 13, and 14 pin peaking factor comparisons are shown in Tables 5.1, 5.2, and 5.3. The C-3/S-3 and C-4/S-3 pin peaking factors are all less than about 1.5%.

Since the pin peaking factors are essentially the same, the C-3/S-3 biases and uncertainties also apply to C-4/S-3.

Therefore the C-3/S-3 fuel pin peaking biases and uncertainties from Table 1.2 of Reference 1 also apply to C-4/S-3:

	Bias	±	Uncertainty
Fuel Rod Fq	0.0 %	±	4.62 %
Fuel Rod Fxy	0.0 %	±	5.20 %
Fuel Rod Fr	0.0 %	±	3.89 %

These pin peaking factor comparisons demonstrate the continuing accuracy of SCE's neutronics models and methodology.

5.3 CORE RADIAL AND AXIAL POWER DISTRIBUTIONS

This section compares CASMO-4/SIMULATE-3 (C-4/S-3) calculated radial and axial power distributions with CASMO-3/SIMULATE-3 (C-3/S-3) calculated results.

SONGS 2 Cycle 14 BOC and EOC radial and axial power distributions are shown in Figures 5.1, 5.2, 5.3, and 5.4. The C-3/S-3 and C-4/S-3 power distributions are negligibly different.

Since the power distributions are essentially the same, the C-3/S-3 biases and uncertainties also apply to C-4/S-3.

Therefore the C-3/S-3 assembly biases and uncertainties from Table 1.2 of Reference 1 also apply to C-4/S-3:

	Bias	±	Uncertainty
Assembly Fq	0.0 %	±	4.17 %
Assembly Fxy	0.0 %	±	4.80 %
Assembly Fr	0.0 %	±	3.34 %

These power distribution comparisons demonstrate the continuing accuracy of SCE's neutronics models and methodology.

5.4 AXIAL SHAPE INDEX

Axial Shape Index (ASI) is defined as: $(P_B - P_T) / (P_T + P_B)$

where P_T = Power in the top half or the reactor core
P_B = Power in the bottom half or the reactor core

CASMO-3 / SIMULATE-3 and CASMO-4 / SIMULATE-3 BOC and EOC axial power shapes are compared in Figures 5.3 and 5.4. The axial shapes from the two computer program systems are negligibly different..

The CASMO-3 and CASMO-4 Axial Shape Index was compared for Cycle 14 of Units 2 and 3. The average difference is 0.002, which is much smaller than the CASMO-3 Axial Shape Index uncertainty of 0.014.

Therefore the CASMO-3 Axial Shape Index bias and uncertainty apply to CASMO-4:

0.003 ± 0.014

TABLE 5.1

Power Distribution Peaking Factor -- Fxy
100% Hot Full power, All Rods Out

Unit	Cycle	Burnup (GWD/MTU)	C-3/S-3 Fxy Prediction	C-4/S-3 Fxy Prediction	Difference (C4 – C3)*	Difference (%)**
2	12	0.25	1.435	1.430	-0.005	-0.35
		1	1.424	1.421	-0.003	-0.21
		2	1.420	1.416	-0.004	-0.28
		4	1.421	1.416	-0.005	-0.35
		6	1.414	1.409	-0.005	-0.35
		8	1.402	1.401	-0.001	-0.07
		10	1.393	1.399	0.006	0.43
		12	1.379	1.392	0.013	0.94
		14	1.363	1.379	0.016	1.17
		16	1.349	1.364	0.015	1.11
		18	1.336	1.347	0.011	0.82
		20	1.320	1.331	0.011	0.83
2	13	0.25	1.398	1.407	0.009	0.64
		1	1.380	1.390	0.010	0.72
		2	1.368	1.377	0.009	0.66
		4	1.374	1.369	-0.005	-0.36
		6	1.375	1.370	-0.005	-0.36
		8	1.368	1.373	0.005	0.37
		10	1.360	1.375	0.015	1.10
		12	1.354	1.369	0.015	1.11
		14	1.344	1.360	0.016	1.19
		16	1.336	1.350	0.014	1.05
		18	1.327	1.338	0.011	0.83
		20	1.312	1.324	0.012	0.91
2	14	0.25	1.414	1.430	0.016	1.13
		1	1.397	1.412	0.015	1.07
		2	1.388	1.398	0.010	0.72
		4	1.396	1.391	-0.005	-0.36
		6	1.391	1.387	-0.004	-0.29
		8	1.380	1.387	0.007	0.51
		10	1.367	1.385	0.018	1.32
		12	1.359	1.377	0.018	1.32
		14	1.351	1.366	0.015	1.11
		16	1.341	1.356	0.015	1.12
		18	1.329	1.343	0.014	1.05
		20	1.316	1.328	0.012	0.91

* CASMO4 – CASMO3

** 100% x (CASMO4 - CASMO3)/(CASMO3)

TABLE 5.2

Power Distribution Peaking Factor -- Fr
100% Hot Full power, All Rods Out

Unit	Cycle	Burnup (GWD/MTU)	C-3/S-3 Fr Prediction	C-4/S-3 Fr Prediction	Difference (C4 – C3)*	Difference (%)**
2	12	0.25	1.402	1.399	-0.003	-0.21
		1	1.403	1.401	-0.002	-0.14
		2	1.408	1.405	-0.003	-0.21
		4	1.405	1.404	-0.001	-0.07
		6	1.399	1.398	-0.001	-0.07
		8	1.389	1.389	0.000	0.00
		10	1.377	1.384	0.007	0.51
		12	1.366	1.376	0.010	0.73
		14	1.353	1.365	0.012	0.89
		16	1.340	1.352	0.012	0.90
		18	1.327	1.338	0.011	0.83
		20	1.313	1.322	0.009	0.69
2	13	0.25	1.352	1.352	0.000	0.00
		1	1.358	1.354	-0.004	-0.29
		2	1.365	1.361	-0.004	-0.29
		4	1.371	1.367	-0.004	-0.29
		6	1.370	1.369	-0.001	-0.07
		8	1.364	1.365	0.001	0.07
		10	1.354	1.364	0.010	0.74
		12	1.343	1.358	0.015	1.12
		14	1.332	1.349	0.017	1.28
		16	1.319	1.336	0.017	1.29
		18	1.309	1.323	0.014	1.07
		20	1.298	1.312	0.014	1.08
2	14	0.25	1.371	1.370	-0.001	-0.07
		1	1.378	1.374	-0.004	-0.29
		2	1.387	1.383	-0.004	-0.29
		4	1.391	1.389	-0.002	-0.14
		6	1.385	1.385	0.000	0.00
		8	1.375	1.378	0.003	0.22
		10	1.362	1.374	0.012	0.88
		12	1.348	1.365	0.017	1.26
		14	1.335	1.351	0.016	1.20
		16	1.323	1.338	0.015	1.13
		18	1.314	1.328	0.014	1.07
		20	1.302	1.316	0.014	1.08

* CASMO4 - CASMO3
** 100% x (CASMO4 - CASMO3)/(CASMO3)

TABLE 5.3

Power Distribution Peaking Factor -- Fq
100% Hot Full power, All Rods Out

Unit	Cycle	Burnup (GWD/MTU)	C-3/S-3 Fq Prediction	C-4/S-3 Fq Prediction	Difference (C4 – C3)*	Difference (%)**
2	12	0.25	1.640	1.639	-0.001	-0.06
		1	1.626	1.628	0.002	0.12
		2	1.619	1.621	0.002	0.12
		4	1.569	1.576	0.007	0.45
		6	1.530	1.535	0.005	0.33
		8	1.515	1.518	0.003	0.20
		10	1.506	1.516	0.010	0.66
		12	1.499	1.515	0.016	1.07
		14	1.484	1.503	0.019	1.28
		16	1.470	1.488	0.018	1.22
		18	1.461	1.475	0.014	0.96
		20	1.447	1.459	0.012	0.83
2	13	0.25	1.571	1.586	0.015	0.95
		1	1.555	1.560	0.005	0.32
		2	1.553	1.553	0.000	0.00
		4	1.516	1.519	0.003	0.20
		6	1.482	1.486	0.004	0.27
		8	1.475	1.485	0.010	0.68
		10	1.471	1.488	0.017	1.16
		12	1.471	1.490	0.019	1.29
		14	1.462	1.481	0.019	1.30
		16	1.455	1.470	0.015	1.03
		18	1.449	1.463	0.014	0.97
		20	1.436	1.450	0.014	0.97
2	14	0.25	1.593	1.615	0.022	1.38
		1	1.587	1.589	0.002	0.13
		2	1.587	1.586	-0.001	-0.06
		4	1.545	1.550	0.005	0.32
		6	1.502	1.508	0.006	0.40
		8	1.487	1.500	0.013	0.87
		10	1.478	1.500	0.022	1.49
		12	1.477	1.498	0.021	1.42
		14	1.470	1.488	0.018	1.22
		16	1.460	1.477	0.017	1.16
		18	1.452	1.469	0.017	1.17
		20	1.439	1.455	0.016	1.11

* CASMO4 - CASMO4
** 100% x (CASMO4 - CASMO3)/(CASMO3)

FIGURE 5.1

S2C14 Axially Integrated RPD Comparison
100% HFP -- ARO -- BOC

CASMO3							0.381	0.754
CASMO4							0.377	0.748
CAS4 - CAS3							-0.004	-0.006
Max Abs Diff =	0.005			0.324	0.579	0.919	0.996	0.852
Min Abs Diff =	-0.006			0.324	0.578	0.915	0.992	0.852
				0.000	-0.001	-0.004	-0.004	0.000
RMS Diff =	0.003							
			0.436	1.058	0.985	1.196	1.031	1.093
			0.435	1.061	0.989	1.195	1.034	1.097
			-0.001	0.003	0.004	-0.001	0.003	0.004
		0.441	0.881	1.200	1.259	1.065	1.263	1.070
		0.441	0.886	1.201	1.259	1.068	1.261	1.072
		0.000	0.005	0.001	0.000	0.003	-0.002	0.002
	0.327	1.067	1.209	1.038	1.083	1.284	1.089	1.286
	0.326	1.071	1.211	1.041	1.086	1.283	1.090	1.284
	-0.001	0.004	0.002	0.003	0.003	-0.001	0.001	-0.002
	0.580	0.988	1.263	1.084	1.282	1.101	1.287	1.088
	0.580	0.993	1.263	1.087	1.281	1.103	1.284	1.089
	0.000	0.005	0.000	0.003	-0.001	0.002	-0.003	0.001
	0.920	1.197	1.066	1.285	1.101	1.285	1.065	1.249
	0.916	1.196	1.069	1.283	1.103	1.283	1.067	1.246
	-0.004	-0.001	0.003	-0.002	0.002	-0.002	0.002	-0.003
0.381	0.994	1.030	1.262	1.089	1.287	1.065	1.219	0.968
0.377	0.990	1.032	1.261	1.090	1.285	1.067	1.217	0.969
-0.004	-0.004	0.002	-0.001	0.001	-0.002	0.002	-0.002	0.001
0.754	0.852	1.093	1.070	1.286	1.088	1.249	0.968	0.789
0.748	0.852	1.097	1.072	1.284	1.089	1.246	0.969	0.785
-0.006	0.000	0.004	0.002	-0.002	0.001	-0.003	0.001	-0.004

FIGURE 5.2

S2C14 Axially Integrated RPD Comparison
100% HFP -- ARO -- EOC

CASMO3							0.494	0.844
CASMO4							0.482	0.835
CAS4 - CAS3							-0.012	-0.009
Max Abs Diff =	0.006			0.390	0.662	1.023	1.133	0.913
Min Abs Diff =	-0.012			0.384	0.654	1.016	1.130	0.912
				-0.006	-0.008	-0.007	-0.003	-0.001
RMS Diff =	0.005							
			0.503	1.013	0.975	1.231	1.019	1.049
			0.494	1.010	0.974	1.231	1.021	1.052
			-0.009	-0.003	-0.001	0.000	0.002	0.003
		0.505	0.894	1.209	1.253	1.018	1.245	1.003
		0.497	0.892	1.209	1.255	1.021	1.248	1.007
		-0.008	-0.002	0.000	0.002	0.003	0.003	0.004
	0.391	1.015	1.212	0.993	1.016	1.248	1.014	1.245
	0.384	1.012	1.211	0.994	1.020	1.252	1.018	1.249
	-0.007	-0.003	-0.001	0.001	0.004	0.004	0.004	0.004
	0.662	0.975	1.254	1.016	1.237	1.016	1.242	1.009
	0.654	0.975	1.255	1.020	1.241	1.021	1.247	1.014
	-0.008	0.000	0.001	0.004	0.004	0.005	0.005	0.005
	1.022	1.231	1.018	1.248	1.016	1.239	0.997	1.227
	1.015	1.231	1.021	1.252	1.021	1.245	1.002	1.232
	-0.007	0.000	0.003	0.004	0.005	0.006	0.005	0.005
0.494	1.132	1.019	1.244	1.014	1.242	0.997	1.201	0.942
0.482	1.130	1.020	1.248	1.018	1.247	1.002	1.207	0.947
-0.012	-0.002	0.001	0.004	0.004	0.005	0.005	0.006	0.005
0.843	0.913	1.049	1.003	1.245	1.009	1.227	0.942	0.811
0.834	0.912	1.052	1.007	1.249	1.014	1.232	0.947	0.815
-0.009	-0.001	0.003	0.004	0.004	0.005	0.005	0.005	0.004

FIGURE 5.3

S2C14 Average Axial Power Distribution Comparison
100% HFP -- ARO -- BOC

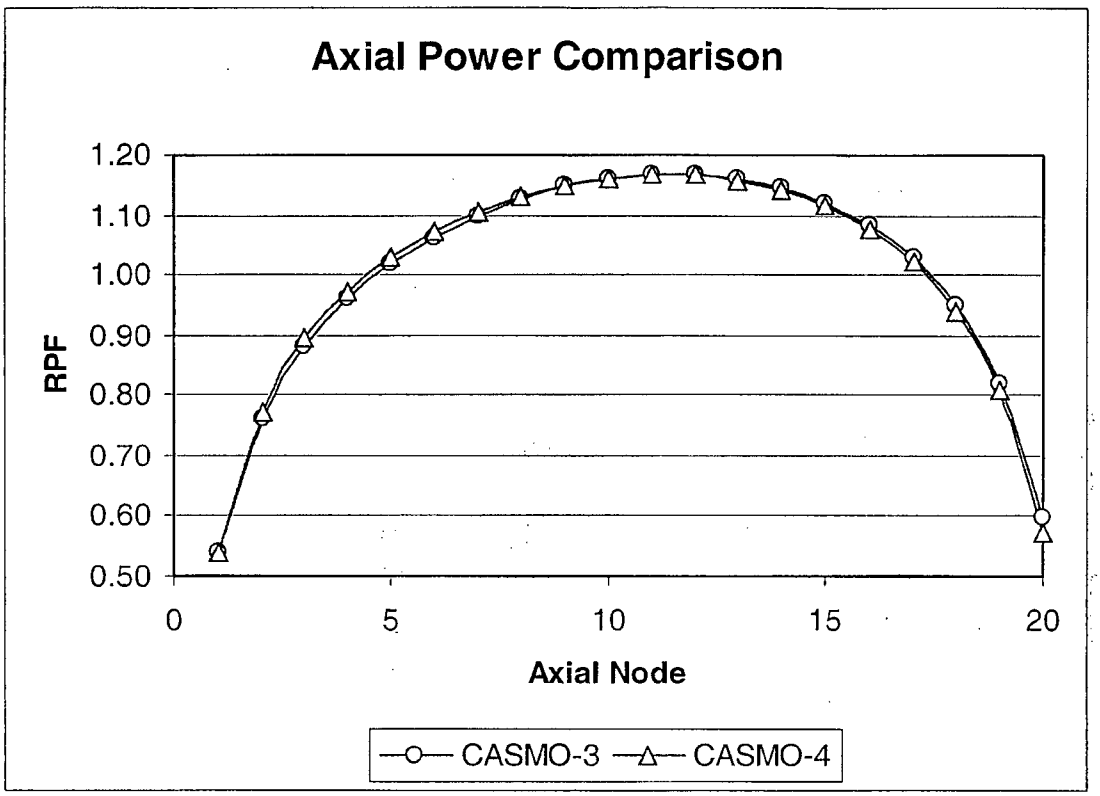
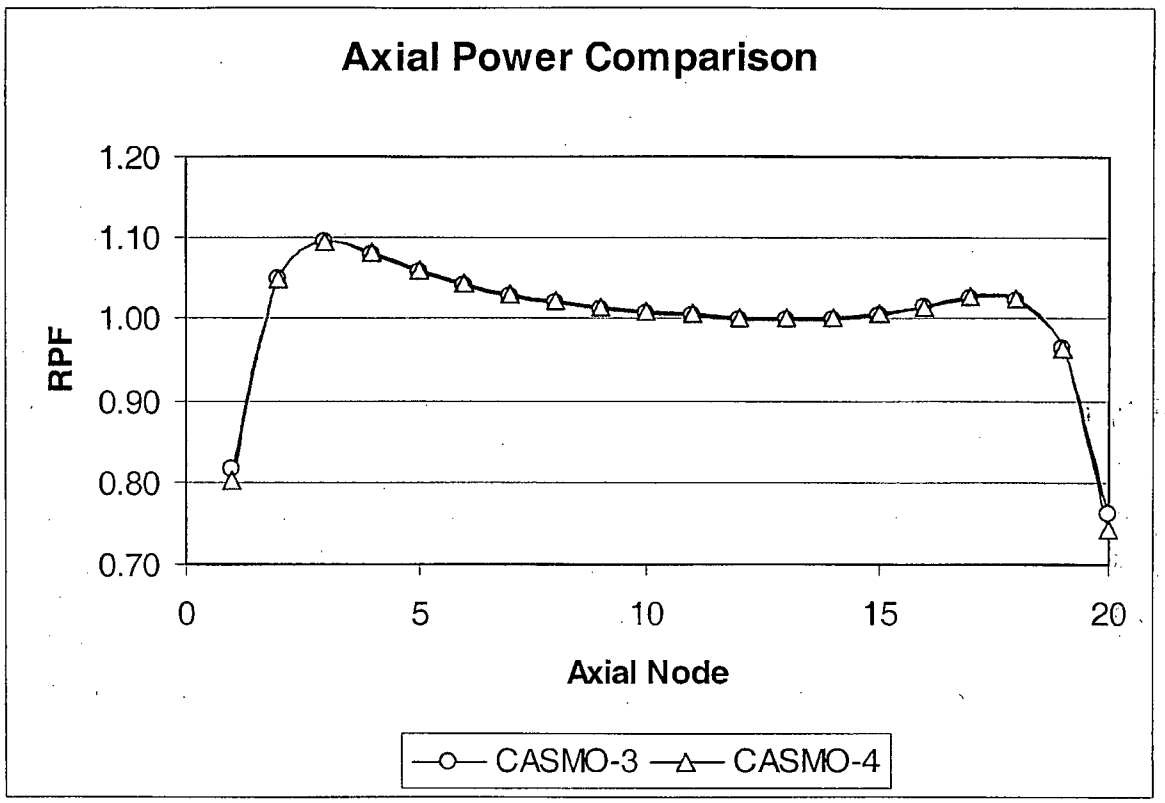


FIGURE 5.4

S2C14 Average Axial Power Distribution Comparison
100% HFP -- ARO -- EOC



SECTION 6

FUTURE BURNABLE ABSORBERS AND FUEL ROD CLADDING

6.0 INTRODUCTION

The CASMO-4 / SIMULATE-3 methodology has also been extensively used for two other widely utilized, common burnable absorber types:

1) Gadolinia where (Gd_2O_3) burnable absorber is mixed directly with the UO_2 in the fuel rod,

2) Integral Fuel Burnable Absorbers (IFBA) where zirconium diboride (ZrB_2) burnable absorber is applied in a thin layer directly on the surface of the UO_2 fuel pellet.

The use of these two modern burnable absorber types permits many more options for cycle reactivity management. Of the two types of burnable absorbers, gadolinia is far more challenging to model properly (IFBA with B-10 is fairly benign). CASMO-4 with its internal Gd-depletion capability was specifically designed to model gadolinia bearing fuel where resonance self-shielding is extremely important along with properly modeling the spatially heterogeneous ("onion-like") depletion of gadolinia.

References 12 and 20 provide benchmark results against critical experiments for gadolinia and MCNP comparisons for both gadolinia and IFBA burnable absorbers. References 18 and 19 provide further benchmarking of the CASMO-4 gadolinia depletion model.

6.1 CRITICALS WITH GADOLINIA

Reference 12 documents the benchmarking of CASMO-4 against several sets of critical experiments. These sets include the Kritz critical experiments performed at Studsvik in Sweden on PWR 3x3 and BWR 4x4 assembly configurations, and the B&W critical experiments performed using PWR ~5x5 assembly mock-ups.

The Kritz series of critical experiments are of special interest because many of the criticals were performed "at temperature," e.g., not just at cold conditions.

KRITZ-3

In the Kritz-3 PWR series or criticals four cores were analyzed - two UO2 cores and two MOX cores - where measurements were performed at temperatures ranging from 20 C to 245 C (i.e., cold conditions to hot-zero-power conditions). The cores are relatively small, but at the same time exhibit low radial leakage.

Summary of Results from Kritz-3 PWR Analysis

Core Design	k-eff	1 sigma
UO2 Cold Criticals	0.99874	0.00130
UO2 Hot Criticals	0.99911	0.00127
MOX Cold Criticals	1.00055	0.00110
MOX Hot Criticals	0.99960	0.00075

The documented uncertainty in the measured boron concentrations was ~1% which is substantial because the boron concentrations were typically ~1000 ppm which translates into ~200 pcm in reactivity. From this, the CASMO-4 results may be considered to be well within the uncertainty of the measurements.

Although the KRITZ-3 PWR cores did not contain gadolinia (or IFBA), comparisons to this set of criticals provides a good estimate of the accuracy of the CASMO-4 isothermal temperature coefficient calculation.

KRITZ-4

The Kritz-4 series of criticals do contain gadolinia and are of direct relevance to CASMO-4 benchmarking for this type of burnable absorber.

In the Kritz-4 series of criticals, two separate core configurations were modeled at both cold and hot-zero-power operating conditions again providing a good estimate of the CASMO-4 isothermal temperature calculation. The configurations below represent a variety of cores some with Gd (of various amounts) and some with control rods.

Core	Condition	Gd	k-eff	% RMS[1]	%RMS[2]	%RMS[3]
2:1	Cold	No	1.00060	0.89	0.73	1.02
2:2	Cold / Rodded	No	1.00103			
2:3	Cold	Yes	1.00079			
2:3	Hot	Yes	1.00021	0.90	0.63	1.33
2:4	Cold / Rodded	Yes	1.00060			
2:5	Cold	Yes	1.00103			
3:1	Cold	No	1.00132			
3:1	Hot	No	1.00008	1.17	0.73	1.38
3:2	Cold	Yes	1.00152			
3:2	Hot	Yes	0.99949	1.14	1.07	1.18

3:3	Cold	Yes	1.00256			
3:3	Hot	Yes	1.00006	1.02	1.04	1.45
3:4	Cold	Yes	1.00279			
3:4	Hot	Yes	1.00031	1.09	0.66	1.32
3:5	Cold	Yes	1.00182			
3:5	Hot	Yes	1.00017	1.29	0.86	1.32

RMS[1] Assy (2,2)
RMS[2] Assy (4,2)
RMS[3] Assy (1,1)

Summary of Results from KRITZ-4Analysis

Core Design	**k-eff**	**1 sigma**
Core 2 Cold Criticals	1.00081	0.00019
Core 2 Hot Criticals	1.00021	N/A
Core 3 Cold Criticals	1.00184	0.00063
Core 3 Hot Criticals	0.99985	0.00046

From these criticals it can be seen that CASMO-4 does a very good job at predicting not only overall global reactivity level, but Gd-worth as well (which is extremely sensitive).

B&W 1810 Series

The B&W 1810 series set of critical experiments (Reference 12) was meant to represent realistic reactor configurations and consisted of ~5x5 array of either PWR 15x15 type assemblies (Babcock & Wilcox design) or PWR 14x14 type assemblies (Combustion Engineering design). The central assembly was altered from one critical experiment to the next. Some cores contained gadolinia bearing fuel pins, some cores AgInCd or B_4C control rods, some cores hollow rods, etc. From this set of criticals, the following cores were analyzed with these results:

	Description	**k-eff**	**RMS[1]**	**RMS[2]**
I	Base B&W, no absorbers	1.00014	0.49	
II	16 AgInCd rods in center assy	0.99986		
V	28 Gd rods, 12 in center assy	0.99952	0.47	0.75
XII	Split enrichment B&W, No absorbers	1.00153	0.74	
XIII	16 B_4C rods in center assy	1.00134		
XIV	28 Gd rods, 12 in center assy	1.00069	0.83	1.26

XVII	Base CE, No absorbers	1.00208	0.71	
XX	32 Gd rods, 16 in center assy	1.00151	0.83	0.81

1 Fission rate RMS (%) in central assembly
2 Fission rate RMS (%) along core diagonal

OVERALL RESULTS FOR CRITICALS WITH CASMO-4

The overall reactivity, for all BWR and PWR cores reported in Reference 12, is 1.00032 with a standard deviation of 123 pcm. In these terms, two-thirds of all cores analyzed fall within an eigenvalue of 0.99900 and 1.00150. This encompasses hot and cold conditions for BWRs and PWRs, with both UO_2 and MOX fuel, with and without gadolinia, cruciform control rods, AgInCd and B_4C control rods, borated and non-borated coolant, and circular and square geometries. In addition, all calculated fission rate RMS's were within the uncertainty of the measurements.

In the CASMO-3 Topical Report (Reference 1), CASMO-3 / SIMULATE-3 pin power distribution results are reported for three B&W cores (Table 5.1):

Case	CASMO-3 RMS (%)	CASMO-4 RMS (%)
Core I	0.631	0.49
Core XII	0.872	0.74
Core XVIII	0.925	(Not Analyzed)

The CASMO-4 results (% RMS) are consistent with and generally better than the above CASMO-3 results.

6.2 MCNP COMPARISONS FOR IFBA AND GADOLONIA LATTICES

Reference 20 provides comparisons between MCNP and CASMO-4 for several IFBA configurations and several containing gadolinia. All MCNP calculations used 1 million histories, split into 50 batches of 20000 histories each. For each lattice design, the fission rate distribution calculated by CASMO-4 is compared against that calculated by MCNP. The accuracy of the fission rate calculation is emphasized since it reflects the accuracy of the overall calculational scheme used in CASMO-4.

WESTINGHOUSE 15x15 28 IFBA PINS
Model: 15x15 Westinghouse design containing 28 IFBA pins (out of 204 fuel pins) with three different IFBA strengths: (i) 1x (nominal loading), (ii) 3x (three times nominal loading), (iii) 5x (five times nominal loading).

Fission Rate RMS(%)

Case	RMS(%)
No IFBA	0.424
28 IFBA (1x)	0.505
28 IFBA (3x)	0.374
28 IFBA (5x)	0.547

IFBA Worth (pcm)

Case	MCNP	CASMO-4	% Difference
1x	-3086	-3117	1.00
3x	-7989	-7997	0.10
5x	-11882	-11848	-0.29

No IFBA's Fission Rate Distribution (C4-MCNP)/MCNP*100%

```
  ---
 0.20  0.63
 0.31  0.42 -0.40
 0.58 -0.40 -0.09  ---
  --- -0.46 -0.37 -0.55 0.00
-0.66 -0.19   --- -0.75 0.10   ---
 0.42  0.21 -0.10  0.31 0.10 -0.20 -0.21
 0.75  0.53  0.32  0.85 0.32 -0.21 -0.53 -0.53
```

28 IFBA's (1x) Fission Rate Distribution (C4-MCNP)/MCNP*100%

```
  ---
  0.30  0.00
  0.11  0.10 -1.38
 -1.05 -0.39  0.00  ---
   --- -0.09  0.10 -0.37  1.06
 -0.75  0.19   ---  0.19 -0.28  ---
  0.00  0.31 -0.58  0.77  1.03  0.59  0.22
 -0.42  0.00 -0.31  0.10  0.73 -0.31 -0.31 -0.72
```

28 IFBA's (3x) Fission Rate Distribution (C4-MCNP)/MCNP*100%

-0.10	1.02						
0.36	0.31	0.69					
-0.76	-0.10	-0.37	---				
---	-0.09	-0.44	0.55	-0.19			
-0.56	0.00	---	-0.28	-0.28	---		
0.36	-0.40	0.00	0.00	-0.30	0.29	0.85	
-0.62	0.31	0.30	0.21	0.10	-0.20	0.10	-0.50

28 IFBA's (5x) Fission Rate Distribution (C4-MCNP)/MCNP*100%

0.29	-0.20						
0.26	0.20	1.28					
-0.75	-0.29	0.47	---				
---	-0.18	0.00	0.46	0.74			
-1.10	-0.36	---	0.66	-0.72	---		
0.13	0.20	0.10	0.00	-0.10	-0.48	0.52	
0.93	-0.10	0.20	0.82	0.40	-0.89	-0.51	-1.47

Combustion Engineering 14x14 64 IFBA PINS
Model: 14x14 Combustion Engineering design containing 64 IFBA pins (out of 192 fuel pins) with IFBA strength: (i) 1x (nominal loading)

Fission Rate RMS(%)

Case	RMS(%)
64 IFBA	0.408
64 IFBA + B4C ROD	0.795

IFBA Worth (pcm)

Case	MCNP	CASMO-4	% Difference
64 IFBA	-12014	-11996	-0.15

B4C Control Rod Worth (pcm) with 64 IFBA present

Coefficient	MCNP	CASMO-4	% Difference
B4C Rod Worth	-23208	-23035	-0.75

No IFBA's Fission Rate Distribution (C4-MCNP)/MCNP*100% (800 ppm, no B4C rod)

-0.47	-1.05					
-0.20	0.41	-0.38				
-0.53	-0.41	0.09	---			
-0.43	-0.51	0.19	---	---		
-0.22	-0.21	0.78	-0.37	0.92	0.47	
0.11	0.11	-0.10	0.68	0.19	-0.11	0.53

64 IFBA's Fission Rate Distribution (C4-MCNP)/MCNP*100% (800 ppm, no B4C rod)

-0.84	-0.09					
0.78	0.58	0.76				
0.10	0.29	0.00	---			
0.00	-0.30	-0.67	---	---		
0.21	-0.10	-0.11	0.09	-0.09	-0.68	
0.23	0.41	0.50	-0.11	-0.30	-0.44	0.00

64 IFBA's Fission Rate Distribution (C4-MCNP)/MCNP*100% (800 ppm, B4C rod)

0.56	0.82					
0.59	0.89	0.39				
1.34	0.38	-0.72	---			
0.45	-1.34	-0.74	---	---		
0.00	-1.14	-0.23	-0.76	0.00	-1.85	
0.10	0.75	0.59	-0.78	-1.15	-0.44	0.83

WESTINGHOUSE 15x15 16 Gd PINS

Model: 15x15 Westinghouse design containing 16 gadolinia pins (out of 204 fuel pins) with three different IFBA strengths: (i) 1x (nominal loading), (ii) 3x (three times nominal loading), (iii) 5x (five times nominal loading).

Gd Worth (pcm)

Case	MCNP	CASMO-4	% Difference
16 Gd	-12874	-13070	1.52

No Gd Fission Rate Distribution (C4-MCNP)/MCNP*100% (800 ppm)

-0.19	-0.40						
-0.10	-0.20	-0.78					
-0.67	-0.19	-0.28	---				
---	-0.36	0.00	-0.62	-0.27			
0.47	-0.09	---	0.18	0.94	---		
0.51	-0.49	-0.27	0.50	0.10	-0.19	0.10	
0.42	0.72	0.30	0.00	-0.41	-0.21	0.11	1.29

16 Gd Fission Rate Distribution (C4-MCNP)/MCNP*100% (800 ppm)

0.90	0.56						
0.71	0.58	0.19					
0.41	0.59	-0.70	---				
---	-0.70	-0.52	-0.45	0.41			
-0.09	0.09	---	-0.18	-0.47	---		
0.48	0.30	0.83	0.10	-0.93	0.00	0.55	
0.20	0.71	-0.11	-0.30	0.38	-0.28	0.00	0.55

Combustion Engineering 14x14 16 Gd PINS
Model: 14x14 Combustion Engineering design containing 16 Gd pins (out of 192 fuel pins)
Fission Rate RMS(%)

Case	RMS(%)
No Gd	0.404
16 Gd	0.713

Gd Worth (pcm)

Case	MCNP	CASMO-4	% Difference
16 Gd	-15777	-15914	0.87

No Gd Fission Rate Distribution (C4-MCNP)/MCNP*100% (800 ppm, no B4C rod)

0.27	0.10					
-0.30	-0.10	-0.29				
-0.21	-0.30	0.35	---			
-0.11	-0.31	-0.35	---	---		
0.11	-0.51	0.88	-0.17	0.00	0.57	
0.00	0.00	0.61	0.69	0.29	-0.78	-0.49

16 Gd Fission Rate Distribution (C4-MCNP)/MCNP*100% (800 ppm, no B4C rod)

-0.79	-0.09					
-0.28	0.62	0.49				
-0.91	-0.52	-1.07	---			
-0.11	0.49	-0.60	---	---		
-0.60	0.31	0.27	0.16	0.67	0.33	
-0.85	-0.19	1.46	0.53	-0.18	1.69	0.09

6.3 CONCLUSION FOR FUTURE BURNABLE ABSORBERS

Therefore, based on the benchmarks presented above, CASMO-4 / SIMULATE-3 will also accurately model both gadolinia and IFBA based burnable absorbers in the San Onofre Units 2 and 3 reactor cores.

The above IFBA and gadolinia pin power distribution RMS differences are consistent with and generally better than the CASMO-3 results reported in Reference 1. Therefore, the CASMO-3 assembly and fuel rod biases and uncertainties are applicable to IFBA and gadolinia fuel designs.

6.4 M5 CLADDING

SONGS 2 and 3 fuel assemblies currently use Zircaloy-4 and Zirlo cladding. Lead Fuel Assemblies (LFAs) from AREVA will use M5 cladding (Reference 21). References 21 and 22 provide the chemical composition of Zircaloy-4, Zirlo, and M5 cladding materials. Neutronically, these three cladding materials are insignificantly different and all may be modeled as Zircaloy-4 in CASMO-4.

The NRC safety evaluation (Reference 23) for ZIRLO states:
"The change in the cladding material from OPTIN to ZIRLO will have negligible effect on nuclear fuel performance since the primary change in physics properties is a small increase in neutron absorption attributable to the addition of niobium. An increase in neutron absorption of this magnitude has no effect on nuclear performance. Thus, no modifications were made to the nuclear engineering methodologies or computer codes. This is the same approach that was used for the previous application of ZIRLO. The staff agrees that the change would be negligible and, thus, finds this approach acceptable."

The same discussion would apply to M5 cladding.

Therefore, Zircaloy-4, ZIRLO and M5 cladding have similar neutronic properties and can all be modeled as Zircaloy-4 in CASMO-4.

SECTION 7

CONCLUSION

SCE has performed extensive benchmarking using the CASMO-4 / SIMULATE-3 methodology. This effort consisted of comparisons of calculated physics parameters to SONGS measurements, comparison to another approved, benchmarked system (CASMO-3), and Studsvik-Scandpower critical experiment benchmarking and MCNP comparisons. The benchmarking includes erbia, gadolinia, and IFBA burnable absorbers. The CASMO-4 / SIMULATE-3 methodology also applies to Zircaloy-4, Zirlo, and M5 cladding. A set of biases and 95/95 (probability/confidence) tolerance limits for key physics parameters have been determined.

As CASMO-4 is the direct evolutionary descendent of CASMO-3 which SCE has previously licensed, the experience and proficiency that the SCE engineering staff has demonstrated with CASMO-3 is directly applicable to CASMO-4.

Based on the analyses and results contained in this report, SCE concludes that the CASMO-4 / SIMULATE-3 methodology applies to steady-state SONGS reactor physics calculations. The accuracy of this methodology is sufficient for use in licensing applications, SONGS reload physics analyses, safety analyses inputs, startup predictions, core physics databooks, and, reactor protection system and monitoring system setpoint updates.

Based on the analyses and benchmark results contained in this report, SCE concludes that the CASMO-4 / SIMULATE-3 methodology may also be used for fuel assembly types containing gadolinia burnable absorber and Integral Fuel Burnable Absorber (IFBA). The biases and uncertainties listed in this report also apply to fuel assembly types containing gadolinia and IFBA's.

This effort has also successfully demonstrated SCE's ability and proficiency to use the CASMO-4 / SIMULATE-3 computer program package.

SECTION 8

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