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June 10, 2009

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Proposed License Amendment Request for Methodology Report DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3"
License Amendment Request No. 2009-02

References:
1. DPC-NE-1004-A, "Nuclear Design Methodology Using CASMO-3 / SIMULATE-3", Revision 1a, January, 2009.
2. DPC-NE-1005-PA, "Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 MOX", Revision 0, SER dated August 20, 2004.
3. DPC-NE-1005-PA, "Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 MOX", Revision 1, SER dated November 12, 2008.

Duke Energy Carolinas, LLC (Duke) hereby submits a license amendment request (LAR) for the Oconee Nuclear Station Renewed Facility Operating License (FOL). Specifically, Duke requests Nuclear Regulatory Commission (NRC) review and approval of methodology report DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3." Duke currently performs reload design analysis for Oconee Nuclear Station (ONS) using CASMO-3 / SIMULATE-3 (Reference 1). As part of a continuous effort to improve design methods, this methodology report is presented to seek NRC review and approval for the CASMO-4 / SIMULATE-3 nuclear design methodology. Methodology report DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3," describes the methodology for application to core designs containing low enriched uranium (LEU) fuel bearing lumped burnable and/or gadolinia integral absorbers and its associated technical justification. This methodology is consistent with that used for the McGuire and Catawba Nuclear Stations reload core designs (References 2 and 3).

This report presents an alternate model, based on CASMO-4 / SIMULATE-3 software package, for performing nuclear design calculations for the ONS units. This report provides CASMO-4 / SIMULATE-3 benchmark comparisons to operating data from fuel cycles using both B_4C lumped and gadolinia integral burnable absorbers, and from the B&W Urania Gadolinia critical experiments. From this benchmarking, a new set of power distribution uncertainties are developed that are used in different aspects of reactor core reload design. These uncertainties

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may be updated as necessary using the methodology described in this report as new operating data is collected from subsequent ONS fuel cycles.

This report contains information that is proprietary to Duke. In accordance with 10 CFR 2.390, Duke requests that this information be withheld from public disclosure. Enclosure 1 contains an Affidavit attesting to the proprietary nature of the information in the report. The proprietary information is owned by Duke and has substantial commercial value that provides a competitive advantage. Attachment 1 contains the proprietary report. The non-proprietary version of the report is also included in Attachment 2.

This LAR is seeking review and approval of the methodology change only. Future submittals are planned to address transitioning to gadolinia fuel in support of 24 month fuel cycles. All affected technical specifications will be submitted at that time. There are no commitments being made as a result of this LAR. Duke requests approval of the LAR by March 31, 2010.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, these proposed changes to the license have been reviewed and approved by the Plant Operations Review Committee and Nuclear Safety Review Board. Additionally, a copy of this license amendment request is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

Inquiries on this proposed amendment request should be directed to Reene' Gambrell of the Oconee Regulatory Compliance Group at (864) 873-3364.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 10, 2009.

Sincerely,

Dave Baxter, Vice President
Oconee Nuclear Site

Enclosures:
1. Notarized Affidavit of T. C. Geer
2. Evaluation of Proposed Changes

Attachments:
1. DPC-NE-1006-P – Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 – Proprietary Version
2. DPC-NE-1006 – Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 – Non-Proprietary Version

bc w/enclosures and attachments:

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ENCLOSURE 1

AFFIDAVIT OF T. C. GEER

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1. I am Vice President of Duke Energy Corporation, and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b) (4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390, it is to be received in confidence by the NRC.

   (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

   (v) The Duke proprietary information sought to be withheld in the submittal is that which is marked in the proprietary version of the Duke methodology report DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3." This information enables Duke to:

(Continued)

Thomas C. [signature]
T. C. Geer

(a) Support license amendment and Technical Specification revision request for its Oconee reactors.

(b) Perform nuclear design calculations on Oconee reactor cores.

(c) Perform transient and accident analysis calculations for Oconee.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) Duke uses this information to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke can sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

(Continued)

Thomas C. Geer

T. C. Geer

Thomas C. Geer affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

Thomas C. Geer
T. C. Geer

Subscribed and sworn to me: June 8, 2009
Date

Freda K. Crump
Notary Public

My Commission Expires: August 17, 2011

SEAL FREDA K CRUMP NOTARY PUBLIC LINCOLN COUNTY, NC

ENCLOSURE 2

EVALUATION OF PROPOSED CHANGE

Subject: Proposed License Amendment Request for Methodology Report DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE - 3"

1. Summary Description

2. Detailed Description

3. Technical Evaluation

    3.1 CASMO-4

    3.2 CMS-LINK

    3.3 SIMULATE-3

    3.4 Power Distribution Uncertainties

4. Regulatory Evaluation

    4.1 Applicable Regulatory Requirements/Criteria

    4.2 Precedent

    4.3 Significant Hazards Consideration

5. Environmental Consideration

6. References

1.0 SUMMARY DESCRIPTION

Duke Energy Carolinas, LLC (Duke) requests Nuclear Regulatory Commission (NRC) review and approval of methodology report DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3." Duke currently performs reload design analysis for Oconee Nuclear Station (ONS) using CASMO-3 / SIMULATE-3 (Reference 1). As part of a continuous effort to improve design methods, this methodology report is presented to seek review and approval for the CASMO-4 / SIMULATE-3 nuclear design methodology. Methodology report DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3," describes the methodology for application to core designs containing low enriched uranium (LEU) fuel bearing lumped burnable and/or gadolinia integral absorbers and its associated technical justification. It is included in Attachment 1. This methodology is consistent with that used for the McGuire and Catawba Nuclear Stations reload core designs (References 2 and 3).

This LAR is seeking review and approval of the methodology change only. Future submittals are planned to address transitioning to gadolinia fuel in support of 24 month fuel cycles. All affected technical specifications will be submitted at that time. There are no commitments being made as a result of this LAR. Duke requests approval of the LAR by March 31, 2010.

2.0 DETAILED DESCRIPTION

The design of a commercial pressurized water reactor (PWR) core determines the characteristics of fuel assemblies which are generally similar in design but differ in the amount of fissile material content. The refueling of a reactor core involves removing some of the fuel assemblies and replacing them with fresh fuel assemblies and possibly previously burned fuel assemblies. In a reload core the fuel enrichment, burnup, and burnable absorber content may be different for each fuel assembly in the core. In general, the neutronic and operating parameters of the new core are different from the previous core. The reload design analysis defines the characteristics of the new core and confirms that it can be operated safely while meeting design power generation requirements.

Neutronic analyses are performed to define the number of feed assemblies, their enrichment, burnable poison loading, and the arrangement of fuel and control components within the reactor core. Calculations are performed which verify core safety parameters, determine reactor protection system setpoints, and provide necessary startup and operational information. This report presents a state-of-the art package of analytical models which may be used to perform these analyses. The fidelity of the analytical models is demonstrated by comparison of calculated nuclear parameters to available measurements from power reactor operation and laboratory experiments.

Duke currently performs reload design analysis for the Oconee units using the CASMO-3 / SIMULATE-3 nuclear design methodology. As part of a continuous effort to improve design methods, this methodology report is presented to seek NRC approval of the CASMO-4 / SIMULATE-3 nuclear design methodology described herein for the Oconee units. This report describes the qualification of the proposed methodology for analyzing reactor cores using both lumped and integral burnable absorbers. Duke currently uses B_4C lumped burnable absorber rods in Oconee reactor core designs for reactivity and peaking control. Duke intends to use gadolinia integral burnable absorbers in future Oconee reactor core designs. The use of gadolinia in fuel assembly lattices is not new, and a significant amount of benchmarking of this absorber type has been performed by the industry. The CASMO-4 / SIMULATE-3 code system has been previously used by other utilities to model fuel designs containing gadolinia, and the benchmark calculations performed have shown that the code system is capable of accurately modeling gadolinia bearing fuel. However, the NRC Safety Evaluation for DPC-NE-1004-A requires that Duke perform additional benchmark calculations to assure that the nuclear uncertainties calculated remain applicable for significant changes in fuel design. The introduction of gadolinia bearing fuel is considered a significant fuel design change.

Benchmark calculations were performed using measured data from twenty-one recent Oconee fuel cycles, four recent TMI fuel cycles, and the B&W Urania Gadolinia critical experiments. Oconee and TMI are both B&W plants and are similar in design. More detailed descriptions of Oconee and TMI are presented in Sections 3.1.1 and 3.2.1 of the report, respectively. The Oconee benchmark calculations include core designs containing AREVA NP's Mk-B fuel product with B_4C lumped burnable absorber rods. The TMI benchmark calculations include core designs containing AREVA NP's Mk-B fuel product with both B_4C lumped burnable absorber rods and gadolinia integral burnable absorbers. The TMI core designs contain a large range of gadolinia concentrations and absorber patterns, including some designs with both B_4C lumped and gadolinia burnable absorbers. Because of the diverse loading patterns and fuel assembly designs evaluated in the TMI benchmark, the results and conclusions drawn from the benchmark calculations are considered applicable to future Oconee core designs, which plan to transition to 24-month cycles, using both B_4C lumped and/or gadolinia burnable absorbers.

Section 2 of the report briefly describes the CASMO-4 / SIMULATE-3 analytical models used in this nuclear design methodology.

Section 3 presents the power reactor operation benchmark of the nuclear design methodology against the Oconee and TMI measured data and demonstrates the ability of the methodology to predict core physics parameters. Comparisons between predicted and measured critical boron concentrations, control rod bank worths, isothermal temperature coefficients (ITC) and assembly power distributions are presented. A statistical analysis of predicted to measured power distributions was also performed to determine the fuel assembly $F_{\Delta H}$, Fq and Fz peaking factor uncertainties.

Section 4 presents a benchmark of the nuclear design methodology against the B&W Urania Gadolinia critical experimental data for fuel designs containing low-enriched uranium (LEU) fuel pins (fuel pins containing only UO_2 pellets) and gadolinia fuel pins (fuel pins containing UO_2 fuel pellets bearing gadolinia integral absorber), and demonstrates the ability of the methodology to predict relative fuel pin power for the LEU and gadolinia fuel pins. A statistical analysis of predicted to measured pin power distributions was also performed to determine the LEU and gadolinia fuel pin power uncertainties.

Section 5 develops the statistically combined uncertainties (SCUs) associated with the fuel pin FΔH, Fq and Fz peaking factors to be used in reload design power distribution analyses for LEU fuel pins and for gadolinia fuel pins.

Section 6 summarizes the key results of this report.

3.0 TECHNICAL EVALUATION

3.1 CASMO-4

CASMO-4 is a multi-group, two-dimensional transport theory model for burnup calculations on fuel assemblies or fuel pin cells. The code accommodates a geometry consisting of cylindrical rods of varying composition in a square pitch array. CASMO-4 can model fuel pins, lumped and integral burnable absorbers, control rods, guide tubes, in-core instruments, water gaps, and reflectors. The nuclear data library input to CASMO-4 is based mainly on data from ENDF/B-IV. It contains cross sections for more than 100 materials commonly found in light water reactors. The cross sections are collected into 70 energy groups covering neutron energies from 0-10 million electron volts (Me V). At this time, CASMO-4 supports NRC-approved methodologies at McGuire, Catawba, Palo Verde, Prairie Island, Surry, and North Anna Nuclear Stations.

Important new features of CASMO-4 over CASMO-3 are the incorporation of microscopic depletion of burnable absorbers into the main calculation, use of a geometrically heterogenous model for the entire calculation, and use of the "method of characteristics" for solving the transport equation.

The use of the "method of characteristics" to solve the neutron transport equation in heterogeneous geometry, and a microscopic depletion model for burnable absorbers, allows for the direct computation of nuclear data in CASMO-4 for fuel bearing gadolinia. The code internally adjusts the detail of the depletion calculation to appropriately account for the depletion of gadolinia.

CASMO-4 cases are executed for each unique fuel assembly lattice configuration for fuel designs with and without gadolinia. A typical case set characterizes the effect of fuel burnup, moderator temperature, fuel temperature, soluble boron concentration, shutdown cooling times, and control rod or lumped burnable absorber presence. For core reflector regions the impact of changes in moderator temperature and soluble boron concentration are typically modeled. The exact case set is dependent on the proposed CMS-LINK setup, as described in the following section.

3.2 CMS-LINK

CMS-LINK processes data generated by CASMO-4 and produces a nuclear data library for input to the SIMULATE-3 core model. The code collects the following data for each unique fuel assembly lattice configuration:

- Macroscopic cross sections in two energy groups
- Discontinuity factors at fuel assembly boundaries in two energy groups
- Yields and microscopic cross sections for important fission products
- Incore detector constants
- Kinetics data
- Pin-by-pin power distributions

The data is collected into multi-dimensional tables that characterize the effect of both instantaneous and integrated perturbations to local core conditions. The precise functionalization of the data varies depending on the type of data and the amount that a given data type changes as core conditions change.

CMS-LINK is the modern version of TABLES-3, the linking code included in the currently approved methodology. The usage of these two codes is functionally equivalent.

3.3 SIMULATE-3

SIMULATE-3 is a three-dimensional diffusion theory reactor core simulator. The program calculates core wide power distribution and fuel depletion with macroscopic cross sections in two energy groups. Homogenized cross sections and discontinuity factors are used on a coarse mesh nodal basis to solve the two group diffusion equation using the QPANDA neutronics model, which uses the fourth-order polynomial representations of the intra-nodal flux distribution in both fast and thermal neutron energy groups. A nodal thermal-hydraulics model is also incorporated to provide both fuel and moderator temperature feedback.

The nodal solution is performed on a geometric mesh of either one or four nodes per assembly in the radial plane and an appropriate axial mesh in the active fuel column. Explicit models of top, bottom, and radial reflector regions allow analytic solutions for flux and leakage at the core boundary. Pin power distributions are constructed by synthesizing results of the nodal mesh solution with heterogeneous lattice solutions extracted from CASMO-4. A microscopic depletion model is used to track iodine, xenon, promethium, and samarium during anticipated core transients. The macroscopic effects of fission product isotopic decay after shutdown can also be included on a nodal basis. No modifications to the nodal solution or pin power reconstruction routines were necessary to model reactor cores containing fuel assemblies bearing gadolinia.

The fuel temperature data used within SIMULATE-3 is developed from data derived from the fuel rod thermal model within SIMULATE3-K.

SIMULATE-3 is fundamentally the same computer code as the earlier versions used in the current approved methodologies.

3.4 Power Distribution Uncertainties

The power distribution uncertainties are applied in the design of reload cores. The uncertainties are applied to the pin power distribution peaking factors to ensure a conservative comparison to the thermal design limits on fuel pin performance. Because a direct measurement of individual pin power distribution is not available from power reactor operation, the complete uncertainty in the core model's ability to predict pin power distribution must be constructed from a synthesis of power reactor and B&W critical experiments benchmark results.

This report presents the pin peaking factor statistically combined uncertainties for application in reload design safety-related analyses for reactor cores using LEU fuel bearing lumped burnable and/or gadolinia integral absorbers.

The CASMO-4 / SIMULATE-3 nuclear design methodology is acceptable for use in performing reactor core reactivity and power distribution calculations for input into safety-related reload design analyses for the Oconee reactor cores containing LEU fuel bearing lumped burnable and/or gadolinia integral absorbers.

The UFSAR will be updated to include the revised methodologies and the new analysis results following NRC approval of the revisions. This will be addressed through normal processes.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The principal design criteria for Oconee 1, 2 and 3 were developed in consideration of the seventy General Design Criteria for Nuclear Power Plant Construction Permits proposed by the AEC in a proposed rule-making published for 10CFR Part 50 in the Federal Register of July 11, 1967. The applicable regulatory requirements for Reactor Core Design are defined in the Oconee Updated Final Safety Analysis Report (UFSAR), Chapter 3, Criterion 6. This LAR is being submitted in accordance with 10 CFR 50.90.

4.2 Precedent

The CASMO–4 / SIMULATE–3 code system has been previously approved by the NRC for analyzing reactor cores with LEU fuel bearing lumped burnable absorbers in analysis for :

- McGuire and Catawba Nuclear Stations (Reference 2)
- Palo Verde Nuclear Station
- North Anna and Surry Nuclear Stations

The CASMO-4 / SIMULATE-3 code system has been previously approved by the NRC for analyzing reactor cores with LEU fuel bearing gadolinia integral absorbers in analysis for:

- McGuire and Catawba Nuclear Stations (Reference 3)
- Prairie Island Nuclear Station

4.3 Significant Hazards Consideration

Pursuant to 10 CFR 50.91, Duke has made the determination that this amendment request does not involve a significant hazards consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to allow the use of the CASMO–4/SIMULATE–3 reload design software to analyze reactor cores with fuel containing lumped burnable and/or gadolinia integral absorbers does not involve a significant increase in the probability or consequences of an accident previously evaluated. The CASMO–4/SIMULATE–3 codes are used to perform reactivity and power distribution calculations to develop power distribution limits and provide confirmation of reactivity and power distribution input assumptions used in the evaluation of UFSAR Chapter 15 accidents. The SIMULATE-3 code is also used to confirm the acceptability of thermal limits at post accident conditions.

   The benchmark calculations performed verified the acceptability of the CASMO–4/SIMULATE–3 code for performing reload design calculations for reactor cores containing both lumped burnable and/or gadolinia integral absorbers. These calculations confirmed the accuracy of the codes and developed a methodology for calculating power distribution uncertainties for use in reload design calculations. The use of appropriate power distribution uncertainties applicable to core designs in conjunction with predicted peaking factors ensures that thermal accident acceptance criteria are satisfied.

   Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The benchmark calculations performed verified the acceptability of the CASMO–4/SIMULATE–3 code for performing reload design calculations for reactor cores containing both lumped burnable and/or gadolinia integral absorbers. These calculations confirmed the accuracy of the codes and developed a methodology for calculating power distribution uncertainties for use in reload design calculations.

   The application of the CASMO-4/SIMULATE-3 reload design software to perform reload design calculations for reactor cores containing lumped burnable and/or gadolinia integral absorbers will not create the possibility of a new or different kind of accident from any accident previously evaluated. The CASMO–4/SIMULATE–3 software is not installed in plant equipment and therefore the software is incapable of initiating an equipment malfunction that would result in a new or different type of accident from any previously evaluated. The evaluation of accidents and the associated acceptance criteria for these accidents remains unchanged.

3) Involve a significant reduction in a margin of safety.

   The application of the CASMO-4/SIMULATE-3 reload design software to perform reload design calculations for reactor cores containing lumped burnable and/or gadolinia integral absorbers will not involve a significant reduction in a margin of safety.

   Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design function during and following an accident. These barriers include the fuel cladding, the reactor coolant system and the containment system. The reload design process assures the acceptability of thermal limits under normal, transient, and accident conditions. The CASMO–4/SIMULATE–3 reload design software was qualified for the analysis of reactor cores containing lumped burnable and/or gadolinia integral absorbers and a methodology for developing appropriate power distribution uncertainties for application in reload design analyses was developed. The use of these uncertainties for analysis of reload cores ensures that design and safety limits are satisfied such that the fission product barriers perform their design function.

   Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the preceding discussions, Duke concludes that the proposed amendment does not involve a significant hazards consideration under the standard set forth in 10

CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.0 ENVIRONMENTAL CONSIDERATION

Duke has evaluated this LAR against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Duke has determined that this LAR meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

See above evaluation.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

This LAR requests approval of a methodology and this will not affect effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

This LAR requests approval of a methodology and this will not affect occupational radiation exposure.

6.0 REFERENCES

1. DPC-NE-1004-A "Nuclear Design Methodology Using CASMO-3 / SIMULATE-3", Revision 1a, (Revision 1, SER date: April 26, 1996).
2. DPC-NE-1005-PA "Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 MOX", Revision 0, SER date: August 20, 2004.
3. DPC-NE-1005-PA "Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 MOX", Revision 1, SER date: November 12, 2008.