February 21, 2017

Mr. Brandon Haugh
Studsvik Scandpower, Inc.
300 N. Third St. Suite 400
Wilmington, NC 28401

SUBJECT: REGULATORY AUDIT PLAN FOR FEBRUARY 27-MARCH 3, 2017, AUDIT OF STUDSVIK SCANDPOWER INC. TOPICAL REPORT SSP-14-P01/028-TR, "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS" (TAC NO. MF7273)

Dear Mr. Haugh:

By letter dated December 18, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15355A283), Studsvik Scandpower Inc. submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report SSP-14-P01/028-TR, "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors." The NRC staff has determined that additional information is needed to complete the review. As part of its review, the NRC staff will be performing a regulatory audit at Studsvik Scandpower Inc on February 27 through March 3, 2017. The audit plan includes documents that are to be reviewed during the audit as well as requests for additional information.

If you any questions or require any additional information, please feel free to contact me at 301-415-2375 or Leslie.Perkins@nrc.gov.

Sincerely,

/RA/

Leslie Perkins, Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 816

Enclosure:
Audit Plan

SUBJECT: REGULATORY AUDIT PLAN FOR FEBRUARY 27-MARCH 3, 2017, AUDIT OF STUDSVIK SCANDPOWER INC. TOPICAL REPORT SSP-14-P01/028-TR, "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS" (TAC NO. MF7273) DATED: February 21, 2017

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AUDIT PLAN AND REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF STUDSVIK TOPICAL REPORT STUDSVIK SCANDPOWER-14-P01/028 "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS" TAC NO. MF7273

1.0 INTRODUCTION AND BACKGROUND

By letter dated December 18, 2015, Studsvik Scandpower, Inc. (Studsvik or SSP) (Ref. 1), submitted for U.S. Nuclear Regulatory Commission (NRC) staff review of Topical Report (TR), "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors."

SSP-14-P01-028-TR-P is a TR (Ref. 2) that describes the most modern production core analysis tool, the Core Management System-5 (CMS5) system of codes that are applicable to the modeling and analysis of pressurized water reactor (PWR) cores. The CMS5 code system consists of CASMO5, CMSLINK5, and SIMULATE5 codes. A rigorous methodology is presented to calculate Nuclear Uncertainty Factors (NUFs) for physics parameters for which CMS5 predictions can be compared against measurements or higher-order codes. The submitted package consists of extensive set of benchmarks including validation to critical experiments and higher-order codes and a 7 unit/63 cycle comparison of predictions to PWR plant data. Based on the benchmark results and PWR design and operating data, this TR presents a set of generic Nuclear Reliability Factors (NRFs) that accounts for model predictive bias and uncertainty.

Studsvik has also submitted SSP-14-P01/012-R, Revision 1 (Ref. 3) that accompanies the TR. This report provides description of the computational methods and models of CASMO5 to be validated, to describe the intended applications of these models in the preparation of cross-sections for use in other Core Management System (CMS) codes (i.e., CMSLINK and SIMULATE), and to demonstrate the accuracy of CASMO5 by comparing calculated data to measurements and higher order computer codes. CASMO5 validation and measurements have been presented in the document using the experiments, isotopic measurements and codes such as, B&W 1810, B&W 1484, KRITZ-3, Atomic Energy Authority (AEA) Winfrith DIMPLE, thermal critical assembly (TCA) reflector, Yankee Rowe Measurements, Japan Atomic Energy Research Institute (JAERI) PWR Isotopic Benchmarks, C5G7 mixed oxide (MOX) Lattice benchmark, and MCNP6 Uniform Lattice Comparisons.

The CASMO5 code is intended to use with all PWR fuels with the following attributes:

- Pin lattice geometries ranging from 14x14 to 17x17 including both large and small water hole designs.

- Integral burnable absorbers types: Gadolinia (Gd2O3) and integral fuel burnable absorber (IFBA) (ZrB2).
- Discrete absorber types: wet annular burnable absorber, B_4C-AlO_3, boron silicate glass, and hafnium suppression rods.
- Control rod absorber types: B_4C, Ag-In-Cd, W, and hafnium.
- Low enriched, ≤5.0 wt% 235U, uranium oxide (UO_2) fuel.
- Soluble boron in the coolant, and
- In-core detector types of movable fission chambers and fixed designs.

The NRC staff has determined that additional information is needed to complete the review to finalize the draft safety evaluation for this TR. Clarifications and additional information are expected to obtain through a regulatory audit for which a plan follows. The audit plan includes the requests for additional information (RAIs) questions and documents that are to be reviewed during the audit in Section 6.0 of this document.

2.0 REGULATORY AUDIT BASES

Regulatory guidance for the core management system design for PWRs is to confirm that the fuel system design limits will not be exceeded during normal operation or anticipated operational occurrences (AOOs) and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core and adherence to the applicable general design criteria (GDC):

GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of AOOs.

GDC 11 requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

GDC 20 requires automatic initiation of the reactivity control systems (RCSs) to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to assure automatic operation of systems and components important to safety occurs under accident conditions. There are usually primary and secondary independent RCSs.

GDC 26 requires that two independent RCSs of different design be provided, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes.

Regulatory guidance is also provided in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Sections 4.3, “Nuclear Design” and Section 4.4, “Thermal and Hydraulic Design.”

3.0 REGULATORY AUDIT SCOPE

The regulatory audit of the topical review process is intended to gain a better understanding of the detailed calculations, analyses and/or bases underlying the formal TR, and confirm the staff’s understanding of the contents of the report.

The audit will focus on the following areas and documents that consist of the TR and other supporting documents listed below:

1. SSP-14/P01-028-TR-P, “Generic Application of the Studsvik Scandpower Core Management System to Pressurized Reactors.”

2. SSP-14-P01/012-R Rev 1, “CASMO5 PWR Methods and Validation Report.”

3. SSP-08/405 Rev1, “CASMO5 A Fuel Assembly Burnup Program Methodology Manual.”

4. SSP-10/465 Rev3 “SIMULATE5 - Methodology.”

5. Statistical analysis, tolerance limits calculations, NUF methodology, and NRF methodology.

6. Demonstration of code capability through benchmarking of the CMS5 code system for several cycles of measured versus predicted physics parameters using several plant data.

7. CASMO5 validation details and results from measurements using critical experiments (B&W 1810, B&W 1484, Kritz-3, AEA Winfrith DIMPLE, and Tank Type Critical Assembly (TCA)).

8. Other documents listed in Section 6.0.

4.0 <u>AUDIT TEAM</u>

Mathew Panicker: Lead Reviewer
Daniel Beacon: Reviewer

5.0 <u>LOGISTICS</u>

Dates of Audit: February 27-March 3, 2017
Location: Studsvik offices in Waltham, MA

6.0 <u>TOPICS TO BE COVERED/REVIEWED DURING AUDIT</u>

6.1 <u>RAI Questions</u>

1. Please provide details of the range of applicability of various fuel designs, cladding types, burnable and other coated poisons, etc. that can be analyzed using the CMS5 code system:

    a. Range of Uranium enrichment in UO_2
    b. Types of claddings used in US PWRs and their material compositions
    c. Nominal range of fuel pellet density as a fraction of the theoretical density of UO_2
    d. The range of enrichment of poison isotope in integral burnable absorbers types:

Gadolinia (Gd_2O_3) and IFBA (ZrB2) (for example the maximum Gd enrichment in Gd_2O_3).

e. If coated pellet are used the maximum coating thickness that can be analyzed using the CMS5 code system.
f. Fuel rod average burnups and peak pellet burnup (or range of burnups).
g. Range of linear heat generation rate (LHGR) that can be handled by the CMS5 code system.
h. Maximum boron concentration in coolant that can be handled by CMS5.
i. Control rod absorber types: B4C, Ag-In-Cd, W, and hafnium and the isotopic enrichment for the poison/absorber in the control rods.
j. Reflector materials, their compositions and dimensions.
k. Baffle materials, their compositions and dimensions.
l. Incore and excore detector types and absorber material(s) used in detectors
m. Range of applicability of the CMS5 code system for any OTHER PARAMETERS or fuel designs.

2. Explain summary details of how the CMS5 code system works interactively, in terms of processes such as; migration and maintenance (propagation) of uncertainty and accuracy in neutronics parameters from one code to the next during the coupling process.

3. Section 5.3.1 of SIMULATE5 methodology document briefly describe the depletion geometry used in the depletion model. Figure 5-1 provides the axial nodalization of an assembly in the depletion model without providing any nodal dimensions. What is the impact on sensitivity and accuracy of the depletion results for different dimensions of the nodes used?

4. SSP-14-P01/012-R Revision 1 states that the effective cross sections are used in a series of 1D, collision probability micro-group calculations to obtain detailed neutron energy spectra for use in the condensation of the cross sections. The following questions are related to the cross section processing in CASMO5.

   (a) Section 2.3.3 states that in order to decrease execution time, the cross sections are condensed with an approximate flux spectrum. Explain how the energy condensation of the cross section is done. Also explain whether or not the accuracy in the results is compromised due to condensation of cross sections.

   (b) Explain how the cross sections developed in the above process (1-D) is compatible with CASMO5 2-Dimensional transport theory code and eventually compatible with SIMULATE 5 multi-group diffusion or the simplified P3 transport equation solutions.

   (c) Section 2.3.3 of CASMO5 method and Validation Report states that

   (d) accurate collision probabilities for the cylindrical pin cell with flat source in each region are calculated using the FLURIG-2 method developed by Carlvik. Please provide details of this FLURIG-2 method.

(e) Please explain (in more detail than provided in CASMO5 Methodology Manual) the resonance upscatter model included in CASMO5. What is the impact of the resonance upscatter model on the Doppler Temperature Coefficient in CMS5 code system?

5. The following questions are related to the SIMULATE5 methodology for cross section processing and depletion models:

   a) It has been stated that SIMULATE5 uses "quadratic Gd depletion model" from CASMO5. Please explain the details of how the quadratic depletion model is implemented in CASMO5 and SIMULATE5.

   b) Explain what is a "hybrid macroscopic/microscopic depletion" model that is used in SIMULATE5?

   c) It is stated that radial submesh Cross-Sections and Discontinuity Factors are less dependent on the CASMO Boundary Conditions (BCs) than the full assembly counter parts. Why is this? Does this hold true for edge meshes that are a single pin cell thick? What about corner meshes that are only comprised of a single pin cell and are bordered by two BCs?

6. The following questions are related to the SIMULATE5 fuel temperature measurement validation using HALDEN rods and used in the SIMULATE5 thermal-hydraulic model.

   a) Provide a list of the HALDEN rods with details such as gap dimension, gap fill gas and pressure, and exposure that have been used in the fuel temperature validation listing the end of life burnup and Gad content, if any.

   b) An outstanding issue related to the thermal-hydraulic-mechanical and material design of UO_2 fuel is the thermal conductivity of irradiated UO_2 fuel considering the effects of burnup (exposure). The thermal conductivity of irradiated UO_2 fuel is affected by changes that take place in the fuel during irradiation: solid fission product buildup (both in solution and as precipitates), porosity, and fission gas-bubble formation.

      Fuel thermal conductivity section of SIMULATE5 (Section 10.3.1) describes the UO_2 and MOX thermal conductivity with a correction factor (r_p) for fuel density that is accounted for the effect of fuel burnup on thermal conductivity. Provide details of the basis and formulation this factor that appears in Equations 10.3.4 and 10.3.5 of the SIMULATE5 Methodology manual. Is this factor related to the HALDEN rod data mentioned in (a) above?

7. For the gaseous conductance model in SIMULATE5 (Section 10.3.3 Equation 10.3.10) the gas conductivity is expressed as a function of fuel burnup and an effective fuel pellet radius and thermal conductivities of xenon, krypton, and helium gases. Explain the basis and formulation of the proportionality constant (0.3) and the effective fuel pellet radius (r_{fp}).

8. The following questions are related to the methodology and calculations performed for Nuclear Uncertainty Factor (NUF) determination.
   Provide a detailed summary of NUF factor methodology:

- Details of statistical analysis for tolerance limit determination
- Details of 95/95 one-sided tolerance limits determination
- Details of non-parametric tolerance limits calculations

9. No details have been included in the submitted documents related to the TR regarding the determination of NRFs associated with neutronics/physics parameters. Please provide the basis, and the relationship of NRFs to the NUFs associated with the physics/neutronic parameters.

10. Section 3.4.2 of SSP-14/P01-028 TR-P indicates that one of the ways to minimize the uncertainty in critical boron concentration is to restrict/assume a default value for B^{10}/B atom ratio of 19.71 atom present. Explain the basis of this default value. Is this value realistic in actual reactor situations under normal, Hot Zero Power, Hot Full Power, and other operational transients?

11. In the axial portion of the SIMULATE-5 solution technique (the first step), "the 1-D multi-group diffusion equation is solved in the subnode geometry for each assembly." Does this mean it is solved once to represent each subnode of a specific type (with the same axial materials), or is this step performed for literally every subnode in the calculation?

12. In SIMULATE5 the 2D diffusion equation is solved one axial plane at a time. Does this imply that the axial spatial discretization has to be consistent across the core?

13. In SIMULATE5, what assumptions are involved with representing the submesh equations in the finite difference-like format that is used to increase calculational efficiency of the submesh solution?

14. For SIMULATE5, when correcting the initial nuclide number densities (N^{actual}), to correct for as-built enrichment and fuel weight, the term N_i^{SA} is used. What phenomena does this term represent, and how is it obtained?

15. In SIMULATE5 three options can be used for tracking Iodine and Xenon concentrations. Are the same options used for tracking Promethium and Samarium?

16. In SIMULATE5 the transverse leakage is approximated with a parabolic shape, but it is also stated that the actual shape is known from the submesh calculations. Is the parabolic assumption used initially, and eventually replaced by the submesh shape? Are the two combined in some way?

17. Why does SIMULATE-5 perform the radial submesh calculation to get a flux shape when this shape should be already known from the CASMO-5 submesh? Is the CASMO-5 shape used as a starting point for the iteration process?

18. What histories and branching variables are typically varied in CASMO-5 and subsequently used by SIMULATE-5? Does radial submesh-wise data get processed by CMS link and passed between the codes based on these histories and branches?

19. The three-step solution process (3D solver, axial, and radial homogenizations) is iterative by nature, feeding a number of parameters into each other until convergence. During the first iteration, how are parameters that have not yet been determined initialized? Does this have any effect on the final solution?

20. For the TCA reflector experiment CASMO5 Eigenvalue results listed in Table 3-11 (SSP-14-P01/012) the eigenvalue and the Delta increases with increase in steel reflector thickness as well as for increase in steel/water thickness. Explain the implications of this increase when the CMS code system is used for the modeling and analysis of PWR cores.

21. In Tables 4-2 through 4-8, the end of cycle burnup is labelled gaseous waste disposal metric ton of uranium (MTU), but the values are listed in megawatt day MTU. Please correct either the values or the label for burnup.

6.2 <u>Documents to be Reviewed During Audit</u>

1. SSP-14-P01/022-C, J. Hykes, R.M. Ferrer, "Benchmarking SIMULATE5 with Halden Fuel Temperature Measurements," Waltham, MA, 2015 (Reference 11 of Section 2.3.3 of the TR).

2. SSP-14-P01/026C, Revision 0, "Tolerance Limits for SIMULATE5's Pin-to-Box Predictions," J. Hykes, Studsvik Scandpower, 2015.

   a) Calculation notebooks that show detailed calculations for all the CMS benchmark, NUF methodology and development of generically applicable NRF's as reported in Chapter 4 of the TR, Tables 4-2 through 4-8, Tables 4-9 through 4-16.

   b) Calculations that support the Figures 4-1 through 4-10 of the TR.

   c) Calculation notebooks that support the NUFs and NRFs generated in Tables 4-17 through 4-25 of the TR.

3. L.W. Lewman, "Urania-Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," B&W 1810, DOE/ET/34212-41, Babcock & Wilcox, 1984.

4. E. Johansson, et al., "KRITZ-3 Experiments", Internal Report, AB Atomenergi, Studsvik, Sweden, 1973.

5. B. Haugh, D. Dean, "S5C Case Matrix Assessment", SSP-14-P01/020-C, Waltham, MA, 2015.

6. E. Wehlage, M. Kruners, "Benchmarking the SIMULATE5 BEAVRS Model with CASMO5 MxN," SSP-14-P01/023-C, Waltham, MA, 2015.

7. NEA/NSC/DOC (2003)16, "Benchmark on Deterministic Transport Calculations Without Spatial Homogenisation, A 2-D/3-D MOX Fuel Assembly Benchmark."

8. E. Wehlage, “Modeling of the B&W 1810 Series of Critical Experiments in CASMO5,” SSP-14-P01/003-C, 2014.

9. B. Haugh, “Modeling of the B&W 1484 Criticals in CASMO5,” SSP-14-P01/011-C, 2014.

10. M. Kruners, “Modeling of the KRITZ-3 Critical Experiment in CASMO5,” SSP-14- P01/002-C, 2014.

11. B. Haugh, “Modeling of the AEA Winfrith Dimple Critical Experiments in CASMO5,” SSP-14-P01/001-C, 2014.

12. D. Dean, “Yankee Rowe PWR Isotopic Benchmark,” SSP-14-P01/007-C, 2014.

13. D. Dean, “JAERI PWR Isotopic Benchmark,” SSP-14-P01/008-C, 2014.

14. B. Haugh, “C5G7 2D Benchmark Calculation with CASMO5,” SSP-14-P01/006-C, 2014.

15. E. Wehlage, “MCNP6 Lattice Reactivity Comparison to CASMO5,” SSP-14-P01/009-C, 2014.

16. S. Vanevenhoven, “TCA Reflector Critical Experiment Analysis in CASMO5,” SSP-14-P01/005-C, 2014.

7.0 DELIVERABLES

Within 45 days of the audit, the NRC staff will prepare a detailed audit report documenting the information reviewed during the audit and any open items identified during the audit. The audit report will be provided to the applicant in draft form for any proprietary markup.

8.0 REFERENCES

1. Letter SSP-14-P01/031L from Studsvik Scandpower, Inc. to USNRC, “Studsvik Scandpower Inc. Request for Approval of Topical Report SSP-14-P01/028 (NRC Project No. 0816) Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors,” December 18, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15335A283).

2. SSP-14/P01-028-TR-P, Revision 0, “Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors,” Studsvik Scandpower, December 2015 (ADAMS Accession No. ML15355A285).

3. SSP-14/P01/012-R, Revision 1, “CASMO5 PWR Methods and Validation Report,” Studsvik Scandpower, December 2015 (ADAMS Accession No. ML15355A290).

4. Letter from US NRC to Studsvik Scandpower, Inc., “Acceptance for Review of Studsvik Topical Report SSP-14-P01/028-TR,” “Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors,” May 4, 2016 (ADAMS Accession No. ML16112A299)