September 15, 2017

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

SUBJECT: FINAL SAFETY EVALUATION FOR STUDSVIK SCANDPOWER INC. TOPICAL REPORT SSP-14-P01/028-TR, "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS" (CAC NO. MF7273)

Dear Mr. Haugh:

By letter dated December 18, 2015 (Agencywide Documents Access and Management System Accession No. ML15355A283), Studsvik Scandpower Inc. submitted Topical Report (TR) SSP-14-P01/028-TR, “Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors,” to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval. By letter dated August 3, 2017 (ADAMS Package Accession No. ML17193A407), an NRC draft safety evaluation (SE) regarding our approval of TR SSP-14-P01/028-TR, was provided for your review and comment. By letter dated August 15, 2017, you provided comments on the draft SE (ADAMS Package Accession No. ML17229B276. The NRC staff's disposition of the Studsvik comments on the draft SE are presented in the comment resolution table attached to the SE.

The NRC staff has found that TR SSP-14-P01/028-TR, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in licensing action requests, our review will ensure that the material presented applies to the specific plant involved. Requests for licensing actions that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that Studsvik publish approved proprietary and non-proprietary versions of TR SSP-14-P01/028-TR within 3 months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and if the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Studsvik will be expected to revise the TR appropriately or justify its continued applicability for subsequent referencing. Licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

Sincerely,

/RA/

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 816

Enclosure:
Final SE

SUBJECT: FINAL SAFETY EVALUATION FOR STUDSVIK SCANDPOWER INC. TOPICAL REPORT SSP-14-P01/028-TR, "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS" (CAC NO. MF7273) DATED: SEPTEMBER 15, 2017

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT SSP-14-P01/028-TR, "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS"

STUDSVIK SCANDPOWER INC.

PROJECT NO. 816

1.0 INTRODUCTION AND BACKGROUND

By letter dated December 18, 2015, (Ref. 1) Studsvik Scandpower, Inc. (Studsvik) submitted Topical Report (TR) SSP-14-P01/028-TR, "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors" to U.S. Nuclear Regulatory Commission (NRC) for staff review and approval. The topical report (TR) was supplemented with additional information dated May 22, 2017 (Ref. 2).

The TR (Ref. 3) describes the Core Management System 5 (CMS5) system of codes which are modern production core analysis tools that can be applied generically to modelling 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 (NUF) for physics parameters for which CMS5 predictions can be compared against measurements or higher-order codes. The submitted package consists of an extensive set of benchmarks including validation with 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, "CASMO5 PWR Methods and Validation Report," that accompanies the TR (Ref. 4). This report describes the computational methods and models of CASMO5 to be validated, describes the intended applications of these models in the preparation of cross-sections for use in other CMS codes (i.e., CMSLINK and SIMULATE), and demonstrates 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, Tank Type 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.

In order to obtain additional information and clarifications needed to complete the review and finalize the draft safety evaluation (SE) for this TR, a regulatory audit with Studsvik engineers was conducted during February 27-March 3, 2017 (Refs. 5 and 6). The Nuclear Performance and Code Review Branch staff has completed the review of the TR. The SE for this TR follows.

1.1 Code Range and Validation

In order to assess the code and its range of applicability for various fuel design parameters, cladding type, burnable, and other coated poisons, Studsvik provided the following ranges and materials for applicability of the CMS5 code system in response to the NRC staff's request for additional information (RAI):

- PWR fuel with U-235 enrichment in UO_2 up to 10 w/o of U-235;
- A wide range of cladding materials such as, Zircaloy-2, Zircaloy-4, ZIRLO, Optimized ZIRLO, and M5;
- Structural component materials, Stainless Steel, Inconel-718, Inconel-750;
- Nominal densities of UO_2 pellets as low as 9.5 gram (gm)/cubic centimeter (cc) for fuel pellets containing Gd_2O_3 as well as relatively higher nominal density in the range of 10.3 gm/cc – 10.6 gm/cc (or between 94% and 97% of theoretical density (TD));
- Integral burnable absorber Gd_2O_3 with a range of up to 12% as a mass fraction of enriched Gd poison;
- Code allows coated Integral Fuel Burnable Absorbers (IFBA) by specifying the total Boron-10 loading and thickness with thickness and number densities of B-10 adjusted to preserve the total B-10 atoms for IFBA over the fuel assembly;
- Benchmarking of CMS5 system has been performed for burnups of peak average assembly burnup up to 64.5 gigawatt days (GWd)/ metric ton of uranium (MTU), peak average fuel rod burnup of 72.6 GWd/MTU, and peak pellet burnup of 78.2 GWd/MTU;
- CASMO5 generated cross sections for downstream SIMULATE5 analysis applicable to boron concentration to cover normal power operation and shutdown boron concentrations up to 3000 parts per million (ppm) and reasonably beyond;
- Control rod absorber materials hafnium, aluminum oxide, boron carbide, boron silicate glass, silver-indium-cadmium (Ag-In-Cd), europium oxide, tungsten, and dysprosium;
- PWR reflector materials and radial reflector involve light water, air, stainless steel, zirconium, and Inconel;
- Code capable of modeling incore detectors as part of the cross section generation, incore detectors in the instrumentation tube in the fuel assembly with thimble, detector types include movable U-235 fission chambers and self-powered neutron detectors (SPND) with materials, rhodium, vanadium, cobalt, molybdenum, and platinum; and
- Code system capable of modeling excore detector (source range and power range) response at core simulator level using SIMULATE5.

Various sections of this SE will address the limits of applicability of the CMS5 system that are stated above. All these limits may not be found to be acceptable and will be based on the assessments described in this document. The NRC staff will either restate the limits of applicability above or recommend new limits in Section 4.0, “Limitations/Conditions” of this SE.

2.0 REGULATORY EVALUATION

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 adhere to 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 anticipated operational occurrences.

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 anticipated operational occurrences 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” (SRP) Section 4.3, “Nuclear Design” and Section 4.4, “Thermal and Hydraulic Design.”

3.0 TECHNICAL EVALUATION

Topical Report SSP-14/P01-028 is a methodology which consists of CMS5 system of codes that is intended to be used by the Studsvik customers (NRC licensees) for production of physics analyses for PWR cores. The CMS5 code system consists of CASMO5 (Refs. 4, 7, and 8), CMSLINK5 (Ref. 9), and SIMULATE5 (Ref. 10). The applicant has provided an extensive set of benchmarks including validation to critical experiments and higher order codes. The validation also included a 7 unit/63 cycles comparison of predictions to PWR plant data. The TR presents a methodology to calculate the NUF for physics parameters using the CMS5 predictions that can be compared against measurements or the results from higher-order codes. Based on the 63-cycle benchmark using a wide range of PWR design and operating data a set of NRFs are determined to account for model predictive bias and uncertainty.

This safety evaluation of the CMS5 system consists of the following sections:

- Core Management System (CMS) codes description
    - Interactive code system processing
    - Geometry of system models
    - Cross section processing in codes
- Validation and Benchmarking
- NUF methodology
    - Tolerance limits and statistical analysis

- NRF from engineering/physical arguments
    - CMS benchmarking
    - NUF methodology demonstration
    - Physics parameters
    - Plant transients
- NRF generation

3.1 Core Management System Codes Description

The CMS codes are used to perform the neutronic and thermal hydraulic analysis needed for the design, optimization, and safety analysis of nuclear reactor cores. The system comprises three computer codes: CASMO5, CMSLINK5, and SIMULATE5. These codes are used for the cross section processing, design, depletion, and analysis of light water reactor cores. This section will briefly describe each of the CMS5 code system.

3.1.1 *CASMO5 Code and Methodology*

CASMO5 is a multi-group two dimensional (2-D or 2D) transport theory code for burnup calculations on boiling water reactors (BWRs) and PWR assemblies or simple pin cells. The CASMO5 code is capable of treating cylindrical fuel rods of varying composition in a square pitch array with fuel rods bearing gadolinium, erbium, IFBA, burnable absorber rods, cluster control rods, in-core instrument channels, water gaps, and cruciform control rods in the regions separating fuel assemblies. CASMO5 is capable of performing reflector/baffle calculations in its core analysis (Ref. 7).

Some characteristics of CASMO5 are:

- The two-dimensional transport equation solution is based on the method of characteristics
- Nuclear data for CASMO5 consists of a library of microscopic cross sections in 568 neutron energy groups for neutrons from 0 to 20 million electron volts (MeV) and 18 gamma groups
- CASMO5 can accommodate non-symmetric fuel bundles, capable of performing calculations in half, quadrant, or octant symmetry.
- Absorber rods with different cell positions and water holes covering different pin cell positions
- Thermal expansion of dimensions included
- Effective resonance cross sections for each fuel pin
- Microscopic depletion for each nuclide in fuel pin and absorber pin
- Depletion calculations use a predictor-corrector technique that reduces the number of burnup steps for a particular accuracy
- Reflector calculations are included and discontinuity factors are determined at assembly boundaries and for reflector regions

The flow of calculations in CASMO5 is provided as Figure 1.1 of Reference 8. Macroscopic group cross sections are calculated from densities, and geometries provided in the input for

586 energy groups. Effective cross sections in the resonance region are calculated using an equivalence theorem that relates effective resonance integrals for each resonance absorber in the group to the particular heterogeneous problem. The resonance integrals obtained from the equivalence theorem are used to calculate effective absorption and fission cross sections. The "shadowing" effect between different pins is taken into account through the use of Dancoff factors that are calculated internally by CASMO5. The effective cross sections are used in a series of 1-D collision probability micro-group calculations to obtain detailed neutron flux spectra in 586 energy groups for use in the energy condensation of the cross sections.

The NRC staff requested the applicant in an RAI (Ref. 6, RAI question 4) to provide details of the cross section condensation scheme and whether there is any loss of accuracy due to the cross section condensation. The applicant responded that the energy condensation scheme in CASMO5 consists of 1-D pin cell calculations performed before the 2-D lattice calculation. The flux solution from the 1-D pin cell calculations, which is also in the micro-group structure, is used to condense the macroscopic cross sections to the 2-D group structure. To preserve the accuracy of the energy condensation, spectral effects are modeled in the micro-group calculation with pins modeled in three or four regions such as fuel, air, cladding, and coolant.

Studsvik, in its response to RAI question 4, confirmed the preservation of the accuracy of the cross section condensation. The 1-D micro-group calculation involves a detailed representation of each pin type in terms of energy and local spatial effects. The resulting micro-group flux, which captures local spectral effects, is used to condense cross sections specific to pin types. Thus the local spectra effects are conserved by the micro-group calculations by condensing cross sections to the appropriate 2-D group cross sections. The 2-D data is provided to SIMULATE5 calculations since discontinuity factors (DFs) cannot be obtained from 1-D pin cell calculations.

Burnup/Depletion Calculations

Isotopic depletion as a function of burnup is calculated for each fuel pin and for each region containing a burnable absorber. Burnup calculation is performed using the predictor-corrector method which consists of solving the Bateman equations twice, first using the spectra at the start of the step, and then after a new spectrum calculation, using the spectra at the end of the step. The predictor step, performed after time step t_{n-1} calculations, uses the flux at time step t_{n-1} to predict number densities and cross sections for time step t_n. Using these parameters, a new flux spectrum calculation is performed. The corrector step is then performed based on these results before time step t_n. The number densities obtained are averaged with those from the predictor step. These averaged values are taken to be the actual number densities at time step t_n and are used as input for the calculations at time step t_n.

For depletion calculations, a power normalization factor is determined in order to renormalize the predicted flux distribution to match the user designated power density to ensure that the lattice is simulated at the correct burnup rate. Additionally, a depletion mesh is specified for use in the calculations. For most fuel pins, depletion is tracked using a single spatial zone per pin. For Gadolinia (Gad) bearing fuel, the fuel region is divided into annular rings in order to simulate layered depletion effects. Control rods and other poisons are treated through user inputs. Once the flux has been appropriately renormalized for the burnup calculations and the depletion mesh is determined, the two steps described above follow the same approach. Each step solves the Bateman equations, a system of first order Ordinary Differential Equations (ODEs).

Neutron Data Library

The CASMO5 neutron data library is based on ENDF/B-VII.1 data files supplemented with TENDL-2102 data and processed with NJOY-94-105 and NJOY-2012.8. The data library contains cross sections, decay constants, and fission yields for 1095 nuclides and materials. The group structure of cross sections consists of: (1) 128 fast groups to calculate the fast fission and leakage, (2) 41 resolved resonance groups that provide correct flux levels as a function of energy for resonance calculation, (3) 392 narrow constant energy groups for mapping out the resonances in 10 electron volts (eV) to 0.2 eV range, and (4) additional thermal groups below 0.2 eV with energy widths of 0.01 or 0.005 eV.

The CASMO5 neutron data libraries contain absorption, fission, nu*fission, transport, and scattering cross sections. Data are tabulated as functions of temperature when needed. The libraries further contain decay constants for fission products and heavy nuclides, and fission product yield values for 27 fissioning species.

In summary, the NRC staff reviewed the documents on CASMO5, theory and methodology, and determined that the code used standard solution methodology for the neutron/gamma transport equation. The benchmarking and validation of the code using critical experiments and comparisons against higher codes (discussed later in this SE) have confirmed the accuracy of the code for use by plants for core design analysis.

3.1.2 *SIMULATE5 Code and Methodology*

Introduction

SIMULATE5 is the latest Studsvik multi-group analytical nodal code that has been developed for advanced fuel designs and core operational strategies (Ref. 10) for BWRs and PWRs. SIMULATE5 solves the multi-group diffusion or, optionally, the simplified P3 transport equations. Cross sections are described by a hybrid microscopic-macroscopic model that includes 50 isotopes (17 actinides and 30+ fission products and burnable absorbers). A core is divided into heterogeneous unit cells with a side length of approximately 15 cm, the heterogeneity being due to the presence of water gaps, spacers, enrichment zoning, and control rods. Inside the node, material discontinuity is present due to enrichment and burnable absorber zoning, spacer grids, and control rods. The axial homogenization model computes, per node, averaged cross section axial discontinuity factors that reproduce the correct heterogeneous nature of the assembly. In the radial direction an assembly is divided into N x N submeshes where the rectangular submeshes follow pin cell boundaries. For PWRs, the outermost submesh is selected in such a way that it captures the intra-assembly mismatch effects such as plutonium buildup. Submesh cross sections and discontinuity factors are generated by CASMO5; the submesh are less heterogeneous since they are materially uniform.

The 2-D diffusion equation (SP_3) is solved one axial plane at a time using the submesh geometry. The axial leakage, obtained from the 3-D solution, converted into an equivalent absorption, and the resulting flux is used to compute the homogenized cross sections. The buildup of important isotopes are tracked per submesh. Three-D (3-D or 3D) global SIMULATE5 solution involves solving the 3-D diffusion equation with homogenized cross section and discontinuity factors employing analytic nodal method (ANM). In ANM, an approximation is made to determine the shape of the transverse leakage. The 3-D solution is

used to tie the 1-D axial and the 2-D radial solutions together and it provides the radial axial leakages to the 1-D/2-D homogeneous module and thereby accounts for the influence of the neighboring channels/radial planes.

SIMULATE5 employs a “subnode” concept in which the fuel assembly is axially divided into materially homogeneous “subregions.” The subnodalization is used both with the axial homogenization and depletion models. The difference between the axial homogenization model subnodalization and depletion model subnodalization is that the material discontinuity due to control rods is ignored in the latter case. The applicant responded to an NRC staff RAI to provide the impact on sensitivity and accuracy of the depletion results for different model dimensions (Ref. 2). An example calculation performed for a seven cycle PWR model with 24 and 48 axial nodes, compared the axial burnup distribution for an assembly with maximum end of life (EOL) burnup. For this fuel assembly both 24 node model and 48 node model tracked the burnup in a proportional number of depletion subnodes, thereby showing that the 24 node model captures all the details including spacer/grid suppressions, in the assembly burnup.

SIMULATE5 performs depletion calculations and tracks 60 nuclides which are employed in the hybrid macro/microscopic cross section model to account for depletion history and shutdown cooling effects.

SIMULATE5 has a thermal hydraulic (TH) model for PWR systems for regions that extend from lower to the upper tie plate. The core portion of the TH models of both PWRs and BWRs are treated identically with each assembly having an active channel and a number of parallel water channels. For each axial node of a channel, the total mixture mass, steam mass, mixture enthalpy, and mixture momentum balance equations are solved. Three-D fuel temperatures are obtained from solving the radial heat conduction equation.

SIMULATE5 is integrated into the family of CMS codes. The assembly burnup code CASMO5 generates all the necessary cross section data and the CMSLINK code assembles a lattice library from CASMO5 output. The SIMULATE5 reads the data and performs steady-state analyses and the transient codes S3K and S3R read SIMULATE5 data and perform time-dependent analyses.

SIMULATE5 Solution Strategy

Global 3-D solver solves the multi-group diffusion equation using the ANM methodology where each PWR assembly or quarter assembly is divided into several parallelepipeds called nodes, each node having strong material heterogeneity in both axial and radial directions. Homogeneous cross sections are employed through axial and radial homogenization procedures. The 3-dimensional problem is converted into three one-dimensional problems for each of the three dimensions by integrating out the two directions. The leakage term is approximated in the analytic nodal method to have a parabolic shape with a quadratic polynomial form. Each 1-D multi-group equation is converted into NG (number of groups) one-group equations which are coupled through transverse leakage and solved using a spectral analysis method. The 3xG one-group, one dimensional equations are solved locally with the node side fluxes as integration constants. The 3xG individual local solutions are coupled together by inserting them in to the node balance equation to produce an equation with node average flux as an unknown. Then the resulting balance equation is solved in an iterative process to obtain an estimate of the nodewise flux.

Axial and Radial Homogenization

Though the assembly by design is strongly heterogeneous axially and radially due to the presence of spacers, control rods in proximity, enrichment zoning, and varying height assemblies, the diffusion equation solver uses homogeneous nodes as the basic calculation solver. To accurately reproduce the axial behavior three quantities are needed; group fluxes, discontinuity factors, and axial flux shape functions. Once the detailed axial flux is known, homogeneous cross sections are computed by weighting the cross sections with axial flux profile.

In order to obtain the detailed multi-group flux for each channel, a three-step procedure is performed. First, the 1D multi-group diffusion equation is solved for each individual channel. Second, the assembly is axially divided into 'subnodes' such that the cross sections are constant within each subnode with the boundaries of the subnodes coinciding with the boundaries of the original nodes, spacers, control rod zones and the fuel material zones Third, the boundary conditions provided are at the bottom and top albedos and the radial in-leakage from surrounding assemblies. The axial homogenization results in a power/moderator density loop to produce an accurate 3D flux solution, a pin evaluation to calculate the detailed axial power shape that is used to compute power peaking factors and linear heat generation rates (LHGRs), and subsequent depletion calculations for subnodes based on subnode fluxes and power.

The radial submesh model works in conjunction with the global solver and the axial homogenization. The radial submesh model sees the core as KMAX 2D planes where each plane consists of all fuel assemblies or quarter assemblies. A submesh has uniform cross sections after axial homogenization. For this 2D domain, the diffusion equation or simplified SP3 equation is solved and the output of the submesh calculation is radially re-homogenized cross sections, radial discontinuity factors, and pin power form functions.

Once the assembly homogenized cross-sections are corrected for the axial and radial heterogeneities, the 3-D diffusion equation is solved to obtain node average fluxes and surface fluxes/currents. The 3-D solver provides feedback to the 1-D and 2-D solvers via the radial and axial leakage terms. The non-linearity introduced by axial and radial homogenization models are [[small]] compared to the thermal-hydraulic non-linearity. Since all the three steps are repeated during each power/void iteration until the eigenvalue and flux convergence in the 3D global solver the final solution is insensitive to the initialization of the parameters of the models.

SIMULATE5 Cross Section Data

SIMULATE5 cross section data are generated from single assembly CASMO5 calculations. For compatibility with the cross sections, the nodal model must be capable of treating cross sections as functions of node-wise isotopic number density, and other parameters such as instantaneous moderator density, fuel temperature, boron concentration, and control rods. Approximately

60 isotopes are chosen according to their importance to reactivity during normal operation, depletion, transients, and shutdown conditions. These isotopes include actinides, fission products, Gd/B-10, and burnable absorber. Both the macro- and microscopic multi-group cross sections are functionalized into three- or four-dimensional tables as a function of all important instantaneous effects, such as, coolant density, control rods, fuel temperature, etc. Due to its large spatial self-shielding, the Gd isotopes require a special treatment which is described in the following section on "Fuel Depletion Model."

Fuel Depletion Model

In SIMULATE5, the macroscopic depletion model provides the first order correction for the depletion effects and is used if the microscopic model is off. Figure 5-1 of Reference 10 illustrates the fuel assembly with axial heterogeneities due to enrichment and burnable absorber zoning, spacer/grid, and control rod zoning. A subnode concept described above is employed with the depletion model. The material discontinuity due to the presence of control rods is ignored

In SIMULATE5, the macroscopic cross sections for each node are computed as functions of history variables from CASMO5 assembly depletion calculations. History effects are water densities, or control rods inserted. This permits cross sections to be functionalized versus the history variables and SIMULATE5 to model the macroscopic cross sections as functions of nodal history variables.

Microscopic depletion chain is illustrated in Figure 5-2 of Reference 10 for heavy metal, fission products, and burnable absorber. The depletion chain is linearized to decouple the depletion equations for the chains for heavy isotopes (actinides), fission products (other than Samarium chain), fission products (samarium chain), and burnable absorber chains (Gd).

The Quadratic Gd depletion model implemented in CASMO5 and SIMULATE5 solves the burnup equations for Gd chains (Gd-152 through Gd-161 in CASMO5 and the simplified Gd-154 to Gd-155 to Gd-156 to Gd-157 in SIMULATE5). The model assumes that the microscopic absorption reaction rates of gadolinium isotopes are quadratic functions of the number density of Gd-155. This quadratic function models the variations of the spatial shielding effects over the 34 depletion step and therefore improves the overall accuracy of depletion calculations (Ref. 2). The quadratic expansion function differs from the conventional predictor/corrector depletion method in which the absorption rate is assumed to be a constant during each time step.

The quadratic Gd depletion model in SIMULATE5 uses an [[effective Gd model with sub-step integration scheme]] (Refs. 2 and 10). In this model the chain of gadolinium isotopes (Gd-155 to Gd-156 to Gd-157) is replaced by an effective Gd isotope and its number density is defined as: (Ref. 38)

$N_{Gd\text{-}eff}$ = ***[*** ***]*** *and*

$\sigma_{Gd\text{-}eff}$ =***[*** ***]***

Where:

N is the number density of the specified isotope

σ is the microscopic cross section of the specified isotope

Gd_{eff} is the effective Gd isotope

The effective isotope preserves the absorption reaction rates of the isotope chain. The advantage of using the effective Gd isotope over individual Gd isotopes is that the change of absorption cross section as a function of the number density becomes much less pronounced.
[

]

Pin Power Reconstruction

There are two ways of computing pin powers in SIMULATE5. The first method is by using the SMX method where the assembly is divided into N x N submeshes. For each submesh the flux distribution is determined. By superimposing pin power form functions (from CASMO5) on the flux, the power for each of the pin is found. The second method is by using the Fourier flux expansion procedure where the basic geometry unit is not a submesh, but an assembly-wide node or a quarter assembly for PWRs. The Fourier expansion method consists of expansion of the flux into a set of shape functions of which of each solves the local multi-group diffusion equation. The coefficients of the expansion are determined from the boundary conditions.

Both SMX and Fourier expansion method assume that a node is homogeneous in the z direction. Axial heterogeneities within the node are treated by an axial homogenization model. A comparison of SMX and Fourier expansion method has shown that the SMX method is more accurate and the Fourier expansion method is faster.

Detector Calculations

The accuracy of calculated axial and radial power distributions is confirmed by comparisons between measured and calculated incore instrument reaction rates. The neutron and gamma detector responses are evaluated differently in SIMULATE5. For PWRs, the SIMULATE5 evaluates the ex-core detector signals based on a procedure computing the number of epithermal neutrons reaching the ex-core detector location.

The reaction rates in the detectors are reconstructed in SIMULATE5 using intra-nodal fluxes evaluated at instrument locations from pin power reconstruction either with submesh or Fourier expansion method, the CASMO5 instrument form functions, and the microscopic cross section for the instrument. The instrument form functions and microscopic cross sections are functionalized against all the available depletion and statepoint variables. This is important since the microscopic cross sections of the detector depend on the exposure of the fuel assembly and moderator densities. For the SIMULATE5 reaction rate calculations, nodal properties such as exposure and moderator density are used to evaluate the microscopic cross

sections and instrument form functions at the local conditions in order to predict the most accurate values of the reaction rates.

PWR Thermal Hydraulics

The thermal hydraulics (TH) model in SIMULATE5 can be executed as a standalone module or as part of a complete core analysis including the neutronics module. Another option is that the TH module may be run in its isothermal mode where only basic TH quantities are computed assuming no heat generation. The PWR TH calculations stretch from lower tie plate (LPT) to upper tie plate (UTP). Bypass flow for both upward and downward directions are treated in the TH model.

Each assembly is analyzed in detail with each assembly split into four quarter sub-bundles in conjunction with the neutronics analysis. An assembly or quarter assembly sub-bundle consists of a number of parallel flow channels with coolant and water rods. The water rods may be lumped into one or a few channels. As an option, the cross flow between the active channels is allowed.

The channels are divided into subnodes in the same manner as in the neutronics model. For each subnode, the requirements will be met regarding mass balance, energy balance, and momentum balance. This means that SIMULATE5 TH is a 4-equation model, keeping track of the total flow rate, vapor flow rate, liquid/vapor mixture enthalpy, and local pressure. The main parameters are calculated and defined at the node exits.

The heat structure model in SIMULATE5 consists of calculation of the heat transferred across boundaries between two TH nodes. Modeling capabilities include fuel pin heat generation, heat transfer across steam generators and heat transfer to the reactor vessel. The SIMULATE5 fuel pin model calculates the temperature distribution within the fuel and the heat transfer from the fuel to the coolant. The average fuel temperature is used as feedback to calculate cross sections and the nodal fuel enthalpy.

SIMULATE5 heat structure model is capable of treating the geometries; such as, cylindrical fuel pin, annular fuel pin with internal and external cooling, slabs (fuel boxes, heat transfer to the vessel), and spherical (heat transfer to the vessel). Temperature and space dependent thermal conductivities and volumetric heat capacities are computed using built in data. Thermal conductivity dependence on fuel exposure is implicitly accounted for using the Halden data as described in this SE. Boundary conditions for solving conduction equations include symmetry or insulated conditions, given heat transfer coefficient, a heat transfer package, or specified surface temperature. The heat transfer package contains correlations for forced convection, nucleate boiling, transition boiling, and film boiling from the wall to the fluid or from the fluid to the wall.

The fuel-cladding gap thermal conductance model consists of the sum of three components; gaseous conduction through areas of non-contact between fuel and clad, the solid contact conductance through contact areas, and the thermal emission from the fuel surface. The gaseous conductance models the physical effects, such as, fuel pellet irradiation swelling and densification, fuel pellet and clad thermal expansion, clad compression caused by irradiation at high pressure, and the gas gap conductivity degradation as a result of fission gas release (Refs. 2 and 10). The empirical relationship for reduction in gas conductivity due to the presence of

fission products assumes that the correction is independent of plenum or gap volume and the correction is assumed proportional to fuel volume and burnup. The fuel pellet radius is corrected by accounting for the thermal expansion and the fuel solid swelling and densification. The applicant responded in detail of the formulation of the gap gaseous conductance model in response to an RAI (Ref. 2). The staff has reviewed the details of the conductance model and determined that the gap conductance model is acceptable.

In summary, the NRC staff reviewed the documents on SIMULATE5, theory and methodology, and determined that the code used standard solution methodology for the multigroup diffusion equation using energy groups that comply with the set used by the lattice code, CASMO5. The staff reviewed the methodology used in the solution of the diffusion equation, the cross section treatment in SIMULATE5, the isotopic depletion model in the code, and the thermal hydraulics model in the code and determined that the code system is qualified to meet the needs of advanced fuel designs and core operational strategies of PWR plants.

3.1.3 *CMSLINK5 Code and Methodology*

CMSLINK5 (Ref. 9) is a linking code that processes CASMO5 Card Image files into a binary formatted nuclear data library for use by SIMULATE5 and XIMAGE. The code collects the following data from CASMO5 Card Image files, multi-group microscopic and macroscopic cross sections, multi-group sub-mesh macroscopic cross sections, detector data, pin power reconstruction data, kinetics data, isotopics data, and spontaneous fission data.

CMSLINK5 is capable of processing data for cold and hot PWR segments with or without burnable poison, standard PWR and BWR reflector segments, and scoping libraries.

The library functionalization used for the macroscopic cross sections is used for the fission product data and discontinuity factors as well. The pin library which includes pin peaking, kinetics, isotopic, and detector data is written for each fuel segment by default.

The output of the code is a summary of card image file(s) content, segments present in the library both before and after the execution of the code, the case matrix functionalization and tables of k_{inf}.

3.2 Validation and Benchmarking Process

This section describes how the validation and benchmarking of the TR is performed. The validation and benchmarking process is implemented to ensure that each code in the CMS5 system accurately performs its intended functions while also ensuring the integral CMS5 system performs in the same manner as intended. Validation and benchmarking of the CMS5 system of codes consists of comparison to physical measurements as well as comparison to higher order computer codes. The higher order code calculations and computational benchmarks will provide the mechanics of the codes as well as to help ensure proper method implementation over the full range of conditions.

3.2.1 Comparisons from Measurements

The CASMO5 validation demonstrates the capability of the code to predict reactivity, burnup (i.e., isotopics), and reaction rates for typical PWR assembly configurations (Ref. 4). The

CASMO5 validation has been performed for PWR fuels with pin lattice geometries ranging from 14x14 to 17x17, integral burnable absorber types (Gadolinia ($Gd_2 O_3$) and IFBA ($ZrB_{2)}$), discrete absorber types (WABA, B_4C-AlO_3, Boron Silicate Glass, and Hafnium), control rod absorber types (B4C, Ag-In-Cd, W, and Hafnium Suppression Rods), low enriched Uranium Oxide (UO2) fuel, soluble boron in the coolant, and in-Core Detector types of movable fission chambers and fixed designs. For the validation process, the results of CASMO5 calculation are compared to either experimental measurements or higher order code calculations. The quantities that are used for comparison are percent absolute error, percent relative error, averages, root mean square error (RMSE), standard deviation, and mean relative pin power error.

B & W 1810 Critical Experiments

The Babcock & Wilcox (B&W) 1810 critical experiments described were designed to represent realistic reactor configurations (Refs. 6 and 13). The experiments consisted of a ~5x5 array of either PWR 15x15 type assemblies (Westinghouse or B&W designs) or PWR 16x16 type assemblies (Combustion Engineering design). Various cores were tested. Some cores contained gadolinium fuel pins, Ag-In-Cd (AIC), or B_4C control rods, or hollow rods. All core configurations from this set were analyzed in Reference 14, with the exception of Core 11, which were specifically designed to measure resonance parameters. All measurements were conducted at a facility temperature of 77^{o} F. Calculated eigenvalues reported in Reference 4 (Tables 3-1 and 3-2) agree well with the experiment with errors less than 1 percent. Table 3-3 of Reference 4 provides the summary results from CASMO5 for fission rates for various B&W 1810 cores. A comparison of average error and absolute error standard deviation in fission rates for the CASMO5 predictions to measured values was performed and the differences are significantly less than 1 percent. This comparison was repeated for all 1810 cores with small and large water holes. Figures 3-3 through Figure 3-8 provides CASMO5 predicted and measured central assembly fission rate for the various B&W 1810 cores and finds that the percent difference is less than 1 percent.

The NRC staff reviewed the results submitted to the staff and verified through the regulatory audit and determined that the B&W 1810 results demonstrate that CASMO5 can adequately model both large and small water hole assembly designs that includes the presence of Gadolinia integral burnable absorber, B4C, and AIC control rods as well as Rhodium detectors.

B&W 1484 Critical Experiments

The B&W 1484 critical experiments consist of twenty-one different geometric configurations (or cores) (Refs. 6 and 15). This experiment is to demonstrate that the CASMO5 transport code can accurately model pseudo core configurations that range from low to high leakage. Figures 3-9 and 3-10 of Reference 4 illustrate Core I and Core II, respectively. These contain no heterogeneities (e.g., water holes, absorber rods, enrichment splits) but since the cores differ in size and shape, they present a wide range of radial leakage. Core I consists of 458 identical fuel pins that contain less than 3% wt% U-235 arranged in a circular shape with a small axial leakage (2% of total reactivity) and significant radial leakage (35% of total reactivity). Core II consists of 1764 identical fuel pins with less than 3 wt% U-235 arranged in square shape with similar axial leakage and radial leakage less than half of the Core I (i.e., relative low leakage compared to Core I). Cores III through IX represent simplified fuel storage configurations where 15x15 pseudo-assembly (individual pins without spacers (i.e., non-unit cell) configurations are arranged in 3x3 spaced lattice configurations. Cores X through XXI represent simplified fuel

storage configurations where 15x15 fuel assembly configurations (arranged in unit cells that are movable) are arranged in 3x3 spaced lattice configurations.

Table 3-4 of Reference 6 lists calculated CASMO5 eigenvalues for Cores 1 through IX for non-unit cell configurations. These results agree very well with the experiment at the B&W 1484 cores. Table 3-5 of Reference 6 lists the calculated CASMO5 eigenvalues for Cores X through XXI with unit cell configurations. A good section of the results that show moderate variation can be attributed to some experimental uncertainty in the concentration of boron in the borated aluminum sheets that are present in Cores XIII through XXI. The model results and uncertainties are very sensitive to the boron content in the boral sheet. To understand the effect of the boral uncertainty different configurations were run by varying the boron content. The further analysis has shown that reducing the boron content does increase the eigenvalue as intended and the spread of the data is also reduced.

The NRC staff has verified the calculations of the validation with B&W 1484 configurations from the regulatory audit (Ref. 6) and determined that CASMO5 effectively models both the low and high leakage configurations and various storage configurations.

KRITZ-3 Critical experiment

KRITZ-3 high temperature critical experiment facility can be operated at a range of temperature 72^{o} F to 473^{o} F and has the ability of CASMO5 to predict reactivity as a function of varying operational conditions (Refs. 4, 6, 16, and 17). The KRITZ-3 facility consists of a cylindrical core loaded with three different fuel pin types: uranium fuel enriched to about 3 wt%, U-235 and MOX fuel, and Plutonium fissile content of low values. The Plutonium fuel pins are loaded at the central location. Eight different core configurations have been analyzed with different critical conditions such as variation in; boron concentration, moderator temperature, water level/neutron leakage (axial buckling) and with or without presence of control rod fingers. Some configurations have small and some others have large water holes depending on the control rod finger cluster studied (Ref. 4, Figures 3-14 and 3-15).

All 89 configurations were modeled in CASMO5 and the results show good agreement with measured data. Of the 89 configurations, 5 of the experiments contained detailed normalized fission rate measurement data which were compared with CASMO5 predictions. These comparisons include the CASMO5 predicted fission rate, measured fission rate, and absolute percent difference. Table 3-8 of Reference 4 summarizes the fission rate comparison results. The CASMO5 average fission rate results agree well with measurement and fall within the 95 percent confidence experiment error. These critical experiments covered a wide range of operating temperatures to better demonstrate code performance at closer to PWR operating conditions. The calculated CASMO5 eigenvalues are plotted as a function of temperature in Figure 3-16 of Reference 4 with good results.

The NRC staff reviewed the experimental procedure, various configurations of the experiment, and the results of comparison during the regulatory audit (Ref. 6) and found the CASMO5 results compare well with measured reactivity and fission rate data. The average absolute error and absolute error standard deviation for the all cores, UO_2 cores, and Pu Cores were within acceptable values. This shows that CASMO5 can perform PWR calculations at varying operating temperatures.

AEA Winfrith DIMPLE

AEA Winfrith DIMPLE (DIMPLE) Criticals are modeled to demonstrate that CASMO5 can model core configurations with and without a baffle (Refs. 4, 6, and 18). Also DIMPLE provides detailed fission rate comparisons to validate both fast and thermal fission distributions. The DIMPLE experimental facility consists of a water-reflected cylindrical system to model power reactor geometries by assembling a cruciform array of 3 percent-enriched uranium dioxide fuel pins arranged to simulate the rectangular corner configuration of a PWR and effectively represented twelve PWR fuel assemblies (Figure 3-22 of Ref. 4). The two configurations are both cruciform shape, with the first being water reflected and the second surrounded azimuthally by a stainless steel region simulating a PWR core baffle. The three experiments were performed in three separate phases known as S06, S06A, and S06B.

The CASMO5 eigenvalues agree well with the experiment and demonstrate that the CASMO5 can predict reactivity well with and without a baffle (Table 3-9 of Ref. 4). These experiments also include detailed fission rate measurements by isotope. Both thermal and fast fission rate distributions were compared with CASMO5 values and found good agreement between them.

The NRC staff reviewed the calculations and results from the DIMPLE experiments for fast and thermal fissions with and without reflectors and determined that CASMO5 calculated eigenvalues and fission rates show good overall agreement. Also the average errors in fission rates for the bare and baffle arrangements were found to be less than 1 percent.

Tank Critical type Assembly Reflector Experiments

The TCA reflector experiments were performed to demonstrate that CASMO5 can effectively model reflectors of varying thickness, and predict kinetic data as compared to measurement (Refs. 4, 6, and 19). Experimental critical configurations were reported for 2.6 percent enriched typical 15x15 PWR fuel assemblies with various configurations of steel and steel/water reflector slabs. Objective of this experiment is to measure the reactivity effect of the steel reflector plates and reflectors containing about 90 percent steel and 10 percent water (Figure 3-35 of Ref. 4).

The eigenvalues calculated in CASMO5 compare well with measurements. Variation of eigenvalues with thickness also is studied. Kinetics parameter ratio β_{eff}/ l^* also is within the experimental uncertainty and shows very good agreement with the measurement

The NRC staff reviewed the experiment and the results and determined that CASMO5 can appropriately model heavy reflectors.

3.2.2 Comparisons from Isotopic Measurements

Studsvik has performed validation of CASMO5 by comparing results against hundreds of isotopic samples in support of the topical report. The data covers various regions of operating PWR cores including the presence of burnable absorbers. These isotopic measurement comparisons were chosen for their direct applicability to PWR predictions since the samples came from operating reactors (Ref. 4).

Yankee Rowe Measurements

The Yankee Rowe measurements are based on Westinghouse samples from several burned fuel pins of Yankee Rowe to measure nuclide composition for the purpose of verification of burnup codes. CASMO5 analysis has been performed to compare the isotopic predictions with the measured data (Ref. 20). The predictions are based on single assembly calculations depleted and unrodded with a power history corresponding to core average conditions over the cycles containing the measured assemblies, including outages at zero power. These cores were operated at 0 ppm boron and controlled with control rods at power. Figures 38 through 53 of Reference 4 show the comparisons of isotope atom densities as a percent of initial U-238 between measurement and CASMO5. These plots show the CASMO5 comparison to measurement for atom densities of isotopes, U-232, U-234, U-235, U-236, U-238, Np-237, Pu-236, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, AM-241, Am-243, Cm-242, and Cm-244 as a fraction of initial loading of U-238 as a function of burnup.

The scatter in the measured data is, to some extent (depending on isotope) due to the variations in power level and local perturbations to the spectrum. The effects of variations in power level among the samples are not included in the predictions, since local pin powers are not available from published data. The isotopes Am-241 and Cm-244 exhibit a strong sensitivity to power while the sensitivity of U-238, Pu-236, and U-232 is much smaller. The other isotopes do not show the sensitivity. For isotopes that are sensitive to power level, local spectrum, and locations with significant control rod history, these predictions are qualitative and cannot be expected to match the measurements in detail.

The NRC staff has reviewed the isotopic predictions from CASMO5 and their comparisons to measured values and determined that the results show that CASMO5 can reasonably predict the isotopic burnup trends of PWR fuel.

Japan Atomic Energy Research Institute PWR Isotope Benchmarks

The JAERI destructively analyzed samples from several PWR and BWR fuel pins to measure nuclide composition for verification of burnup codes (Refs. 4, 6, and 21). CASMO5 analysis was performed to generate the isotopic data for comparison to the measurements. These predictions are based on single assembly calculations, depleted, with power histories that correspond to the sample histories. Figures 3-54 through 3-66 provide plots for per cent difference between CASMO5 predictions and measurement for fission products, actinides, and Gadolinium Isotopes from various JAERI samples.

The NRC staff reviewed the results of the comparison from submitted documents as well as during the regulatory audit (Ref. 6) and found that the overall results demonstrate that CASMO5 can reasonably predict the isotope burnup of PWR fuel including fuel containing Gadolinia poison.

Halden Fuel Temperature Measurements

SIMULATE5 fuel temperature model uses the predicted nodal-average fuel temperatures to look up the appropriate cross sections as well as to provide data for the TM model. Studsvik has performed benchmarking of these temperatures against experimental measurements from the

Halden boiling water reactor for many fuel rods during the operation and depletion of the core (Ref. 37). The measurements included a wide variety of pin types of which eight of the rods were analyzed for validation. The burnup of the rods ranged from 12 to 60 MWd/KgU. Since the rods had various linear heat generation rates (LHGRs), the center-line temperature of these rods range from 800 to 1550 K. SIMULATE5 fuel temperature predictions have been found to be within [[100 K]] for most of the rods. Given the modeling approximations and adjustments in the measured data, the comparisons show good agreement at zero exposure as well as for the increasing trend in the center-line fuel temperature with burnup.

The NRC staff requested (Ref. 2), a list of Halden rods that were used in the validation process. Studsvik provided a list of Halden rods that were used in the validation process. The Table in Reference 2 indicates that these rods have LHGR ranging from 10 kW/m to 30 kW/m, enrichment that ranges from 3.5 to 13 wt%, and EOL burnup ranging from 11.37 MWd/KgUO2 to 50.57 MWd/KgUO2.

The staff in the same RAI, requested the applicant to provide details of how the thermal conductivity degradation (TCD) with fuel burnup is implemented in the fuel temperature calculations and measurements. Studsvik responded that the correction for temperature dependency on thermal conductivity is implemented through a two-step approach. The first step relates to the reduction in the pellet density due to added **[ ]** for cases where average pin burnup is above **[ ]** MWd/KgU (Ref. 2, RAI 6). For the second step a reduction in the pellet density is subtracted from the as-fabricated density to evaluate the density correction in the Nuclear Fuel Industries (NFI) correlations. The added **[ ]** reduces the fuel thermal conductivity.

The SIMULATE5's fuel pellet average temperature predictions were compared against the solution of a reference fuel performance code for a few typical LWR rods. Reference 2 provides plots where fuel temperature is plotted against rod average burnup. The plots show the variation of fuel temperature with burnup and it can be seen that the SIMULATE5 solution is a direct consequence of the burnup dependence on the NFI correlations.

The NRC staff reviewed the SIMULATE5 fuel temperature solution and determined that it effectively accounts for the dependence on thermal conductivity degradation with burnup of fuel.

3.2.3 Higher Order Code Comparison

Studsvik performed higher-order code comparisons to identify code-to-code trends and biases that may reveal deficiencies in the code system. These comparisons extend the range of benchmarking and validation to materials and configurations that are not available in the experimental data.

C5G7 MOX Lattice Benchmark

OECD benchmark described in Reference 22 was performed in CASMO5 to demonstrate the accurate implementation of the transport solution in the CASMO5 code. The benchmark uses a computation geometry, which is a 17x17 fuel assembly lattice to construct a 16 assembly quarter core symmetric geometry. The results of the benchmark calculations and comparisons to reference values are shown in Tables 1 through 3 of the Studsvik's RAI response (Ref. 2).

Results list the CASMO5 eigenvalues comparison, CASMO5 power distribution comparison for percent error, and power distribution statistics (RMSE and MRE).
The NRC staff has verified the results of CASMO5 validation and benchmarking using the C5G7configuration and determined that there is very good agreement between CASMO5 eigenvalue and MCNP code value. This indicates that the transport solution method implemented in CASMO5 is correct. The staff also found that the error in the CASMO5 results is acceptable.

MCNP6 Lattice Reactivity Comparisons to CASMO5

Studsvik performed comparison (Refs. 4, 6, and 23) between CASMO5 and MCNP6 and examined the reactivity effects of different operating conditions. These operating conditions include Doppler temperature defects, moderator temperature defects, soluble boron worth, and control rod worth. Calculations encompass a wide range of PWR lattices and conditions:

- Lattice Designs: 14x14, 15x15, 17x17, 14x14 CE
- Boron concentrations (ppm): 0 - 2500
- Fuel enrichment (wt% U-235): 2.5 - 5.0
- Moderator temperature (F): 70 - 620
- Fuel temperature (K): 293 - 1200
- Removable/Discrete BP type: PYREX, WABA, Discrete B4C (CE)
- Integral BP Gadolinia (wt% Gd): 2.0 - 8.0
- Integral BP IFBA (ZrB2) (mg/cm B10): 1.5 - 3.0
- Control rod absorber type: AIC, B4C, HAF, W

The two codes were compared using the same nuclear data, ENDF/B-VII-R1 in order to gain insight into any significant modeling differences. The results are listed in Tables 3-17 through 3-19 of Reference 4. The lattice reactivity comparisons with MCNP6 show there are no large systematic biases that need further investigation. Since critical experiments are not available for validation with IFBA, Studsvik has performed additional comparisons for IFBA between CASMO5 and MCNP6 codes.

The NRC staff has reviewed the calculations and comparisons between CASMO5 and MCNP6 and determined that the results presented show reasonable agreement. The results include CASMO5 beginning of life (BOL) reactivity benchmarks vs. MCNP6 for global perturbations, CASMO5 BOL reactivity benchmarks vs. MCNP6 for control rod worth, CASMO5 BOL reactivity benchmark vs. MCNP6 for integral absorbers, CASMO5 BOL reactivity benchmark vs. MCNP6 discrete absorbers, comparison between CASMO5 and MCNP6 moderator temperature coefficient, comparison between CASMO5 and MCNP6 Doppler effect, and hot fission rates comparison for different IFBA for W15x15 and W17x17 cases. The staff reviewed the average absolute error and absolute error standard deviation in all the cases and found that they are within acceptable values.

3.2.4 CASMO5 to SIMULATE5

Studsvik has used CASMO5 as higher order code to generate reference solutions to compare against the nodal code SIMULATE5. This process is used for examining three components of the code system; the standard case matrix, the SIMULATE5 nodal code, and the pin-to-box uncertainties needed to support the NUF development.

S5C Case Matrix Validation (Standard Case Matrix Assessment)

The staff in an RAI requested Studsvik to provide the details on the standard case matrix assessment. Studsvik responded with details of the assessment in Reference 2. The S5C matrix for PWR consists of a single assembly calculations that were performed for the following cases:

- Pin lattice geometries ranging from 14x14 to 17x17 including both large and small water hole designs.
- Integral burnable absorber types: Gadolinia (Gd2O3) and IFBA (ZrB2).
- Discrete absorber types: WABA, B4C-AlO3, Boron Silicate Glass and Hafnium Suppression Rods.
- Control Rod absorber types: B4C, Ag-In-Cd, W, and Hafnium.
- Low enriched Uranium Oxide (UO2) fuel.
- Soluble Boron in the coolant

The comparison between exact solutions in CASMO5 and SIMULATE5 codes were performed using cross sections from the S5C case matrix using the audit function in SIMULATE5. The audit function is used to assess how the case matrix performs from two perspectives: 1) how the case matrix performs for PWR operational and safety analysis conditions, and 2) how well interpolation performs between and beyond the actual S5C calculated statepoints. The operational conditions are divided into the following:

1) Normal PWR operation, cold to hot full power.
2) PWR non-LOCA accidents, both rodded and unrodded.
    a. Rod Ejection
    b. Rod Withdrawal
    c. Locked Rotor
    d. Dropped Rod
    e. Large and Small Steam Line Breaks
3) Other PWR Events
    a. Shutdown Cooling Effects
    b. Cold Xenon Free Conditions for Refueling Boron and Startup Predictions

In order to assess how the case matrix performs for PWR operational and safety analysis, the case matrix is chosen to use base conditions that represent a typical PWR and then to perturb those base conditions over the range of operational and non-LOCA scenarios. From these base conditions for Average Moderator Temperature, Average Fuel Temperature, and Average Soluble Boron Concentration, perturbations are performed. The matrix is split into nine regions as described in Reference 2.

The results are evaluated using an acceptance criteria of a primary reactivity difference limit of **[ ]** percent mil (pcm) and a secondary limit of **[ ]** pcm. Most of the reactivity results fall within these ranges and a vast majority of the results are below the **[ ]** pcm threshold. The results for the PWR operational condition audit matrix show consistent results across all the lattice types with average errors typically well below the **[ ]** pcm criteria. The results for the PWR interpolation audit matrix show consistent results across all the lattice types with average errors typically well below the **[ ]** pcm criteria.

The NRC staff has reviewed the S5C matrix validation process and the results presented in the documents and during the regulatory audit and determined that the results of the analysis have demonstrated the accuracy of the cross section library generated by the S5C case matrix in the CMS5 code system and the S5C case matrix is sufficient for analyzing PWR cores.

SIMULATE5 Pin-to-Box Uncertainty

The objective of this analysis is to establish the uncertainty in the pin-to-box ratios computed by SIMULATE5. A fuel pin's pin-to-box ratio is defined as the ratio of the pin power to the assembly average power. Because of the fact that the fuel pin power in a PWR fuel assembly cannot be measured directly, a high fidelity calculation in which multi-assembly CASMO5 calculations is performed. The accuracy of CASMO5's pin by pin fission rate distributions has been validated against MCNP6 and critical measurements

The staff reviewed the tolerance limits (TLs) or uncertainty calculations during the regulatory audit (Ref. 25). The TL is generated in a two-step process. Step 1 is generation of TLs between CASMO5 (C5) and measured pin-by-pin fission rates. The second step is generation of TLs for the difference between pin-to-box ratios from SIMULATE5 (S5) and C5. Then the two steps are combined to yield the desired limits for the S5-to-experiment relationship. The uncertainty on the SIMULATE5 pin-to-box ratios is established by examining a broad range of potential configurations for a 2 by 2 lattice colorset, considering combinations of lattice types, enrichments, boron concentrations, and integral and removable features. Each 2x2 colorset has two fresh and two once burnt assemblies. The once burnt assemblies are taken from single assembly lattices depleted to 20 MWd/kgU.

The NRC staff found the SIMULATE5 pin-to-box uncertainty calculations acceptable. These results are used in the NUFs calculations.

BEAVRS Benchmark

SIMULATE5 is benchmarked by using the BEAVRS model (Ref. 26) which consists of performing 2D CASMO MxN calculations and consistent SIMULATE5 calculations with depletion.

Massachusetts Institute of Technology provided a complete set of data for a 4-loop PWR for benchmarking purposes and a full CMS5 analysis was run. The 2D SIMULATE5 model was benchmarked against the higher order CASMO5 MxN model. The comparisons made are between CASMO5 MxN and SIMULATE5, so absolute adherence to the modeled core’s specific features/parameters wasn’t necessary, as long as the models are consistent. The results show excellent agreement with only a single exposure point that has an error of more than 50 pcm. Low Power Physics Testing (LPPT) parameters were also compared and showed strong agreement with the largest differences in temperature coefficients being 0.32 pcm/F.
The NRC staff reviewed the calculations for BEAVRS benchmarking and determined that the higher order transport methods of CASMO5 used to assess the accuracy of SIMULATE5 for reactivity, power distributions, rod worth, and isothermal temperature coefficient (ITC) is acceptable.

3.2.5 Conclusion

The NRC staff has reviewed the validation and benchmarking process and the related calculations and results that demonstrate the CMS5 code system's adequacy for application to PWR core analysis and determined that the validation and benchmarking process ensures that the CMS5 code system is appropriate for PWR core analysis. The process consisted of 160 critical configurations for a wide range of PWR lattice geometries, materials, and conditions. The inclusion in the process of isotopic measurements has validated the capability of the CMS5 system for depletion calculations to handle burnup predictions for PWR cores. The use of higher order code comparisons and computational benchmarks ensure examination of the entire range of operational conditions where measurements are not available.

3.3 CMS Nuclear Uncertainty Factor Methodology

Nuclear uncertainty factor (NUF) is a terminology used by Studsvik for uncertainty factors. This section describes the methodology for developing the NUFs which can be used in conjunction with the CMS5 code system. The staff has previously approved plant-specific application methods for determining the CMS NUFs (Refs. 27, 28, 29, 30, and 31). This section reintroduces many of the concepts and methods that have already been approved for the above referenced plants. This section presents a generic methodology that is appropriate for the PWR cores.

The NUF methodology is applicable to PWR plants with square lattice designs with regularly spaced arrays of fuel pins. This methodology is applicable to only low enriched uranium (LEU) (see Section on Limitations) fuel assemblies/cores. This methodology is not applicable to MOX fuel. The NUF methodology is independent of any particular CASMO5, SIMULATE5, CMSLINK5, or base cross sections such ENDF, JEFF or their versions, sub-versions, or modifications.

3.3.1 Methodology Overview

The NUF is defined as the calculational uncertainty for a core physics parameter derived from statistical analysis. If measurements are not available in statistically sufficient quality or quantity to determine NUF directly, comparisons may be made to higher order calculations or uncertainty can be obtained indirectly using observations of dependent quantities.

NRF is the allowance to be applied to a safety related core physics model and is chosen to be equal to or more conservative than the corresponding NUF. This number bounds the NUF. For example, if the integral control rod worth NUF upper and lower values are 1.074 and 0.938 respectively, the corresponding upper and lower NRF could be set to 1.10 and 0.90. A set of physics parameters for which it is possible to use a rigorous and well defined statistical method involving historical plant measurements (and in some cases higher-order codes or critical experiments) and CMS5 predictions to determine appropriate NUFs is listed below:

- Overall core reactivity
- Integral control rod worth
- Peak differential control rod worth
- Isothermal temperature coefficient
- 2D peak pin factor, i.e., "enthalpy rise hot channel factor" ($F\Delta H$ / Fr)

- 3D peak pin factor, i.e., “heat flux hot channel factor” (FQ)

For physics parameters for which it is impossible to generate difference data, or for which the measurement uncertainty component of the difference data dominates, appropriate NUFs are presented based upon engineering arguments. This category includes, differential boron worth, Doppler Temperature/Power Coefficient, and Kinetics Parameters (Delayed Neutron and Prompt Neutron Lifetime). Conservative NRFs for these physics parameters are determined by examining other closely related parameters (e.g., critical boron for boron worth), by modeling plant operational transients, or by relying on NRF values that have been previously reviewed and approved.

3.3.2 One-Sided Tolerance Limit Statistical Analysis

The one-sided tolerance limit statistical analysis is a method for determining conservative NUFs for physics parameters from CMS predictions with measurements and/or higher order codes. The basic steps are: (1) for a given physics parameter, compute a set of differences between prediction and measurement, (2) determine either that the data is normally distributed, or is non-normally (non-parametric) distributed, and (3) construct one-sided statistical TLs using standard deviation multipliers if normally distributed, or sorted data non-parametric method if the difference data is not normally distributed.

Difference Data

The difference between a predicted value and corresponding reference value (measurement or higher-order code) is calculated as either:

$$Absolute\ Difference = (Predicted - Reference) \quad (1)$$

$$Percent\ Difference = \frac{(Predicted - Reference)}{Predicted}\ X\ 100 \quad (2)$$

Summary statistics

$$Average = \overline{X} = \frac{1}{N}\sum_{i=1}^{N} a_i \quad (3)$$

Where a_i is a calculated difference for data point i.

The standard deviation is a measure of the spread of the data from the mean

$$Sample\ Standard\ Deviation = \sqrt{\frac{1}{N-1}\sum_{i=1}^{N}\ (X_i - \ X)^2} \quad (4)$$

Where X is the average of the quantity in the set.

Test for Normality

The Shapiro-Wilks test is used to determine whether the data can be considered to be a normal distribution (Ref. 32). Reference 33 provides the Shapiro-Wilks test used in this methodology

that allows the test to be used for any sample size *n* in the range $3 \leq n \leq 5000$. The significance level chosen for this normality test is $\alpha = 0.01$.

Tolerance Limits Assuming Normality

If the difference data set is determined to be normal, then the lower and upper tolerance limits can be computed from the following formulas:

$$\textit{Lower Tolerance Limit} = TL_L = X - (K \times s) \qquad (5)$$

$$\textit{Upper Tolerance Limit} = TL_U = X + (K \times s) \qquad (6)$$

Where:

X = the average of the difference data

s = the sample standard deviation of the difference data

K = one-sided tolerance multiplier

K is obtained from all samples from the equation:

$$K = \frac{t(\gamma, N-1, \delta)}{\sqrt{N}} \qquad (7)$$

Where:

$t(\gamma\ N-1,\ \delta)$ = inverse cumulative distribution function for non-central t distribution

γ= confidence level (0. 95)

N = number of observations

δ = non-centrally parameter calculated as: $\delta = z(p)\ N^{1/2}$

where:

p = proportion of the population to be bounded (0.95)
z(p) = cumulative normal distribution function for example z (p=0.95) ~ 1.645

N = number of observations

Table 3-1 of Reference 1 lists the K values for the same number of observations (N) values listed in Regulatory Guide (RG) 1.126 (Ref. 34).

Non-Parametric Tolerance Limits

For non-normal difference data the lower and upper tolerance limits are determined from non-parametric ranking of Somerville referenced in RG 1.126 (Ref. 34). The Somerville equation of interest is:

$$\gamma \leq I_{1-p}(m, N-m+1) \qquad (8)$$

Where:

γ = the one-sided confidence level (0.95)

I = regularized version of the incomplete Beta function

P = proportion of the population bounded by the m^{th} value

N = Number of observations

M = m^{th} largest rank in the sorted sample

At 95% bounding population and 95 percent confidence level, the equation 8 becomes

$$0.95 \leq I_{0.05}(m, N-m+1) \qquad (9)$$

Equation (9) is solved by iterating on m for a given number of observations *N*. The largest integer value of *m* that satisfies the above relationship is the value of interest. Table 3-2 of Reference 1 lists examples of m values for a set of N observations.

Determination of NUFs from Tolerance Limits

The sign convention is interpreted such that a positive value indicates over-prediction of the magnitude of a parameter by CMS5, and a negative value indicates under-prediction by CMS5. For absolute difference data, the signs of the TLs are simply switched to form the NUFs. For example, if statistically determined TLs for overall core reactivity are -315 pcm to + 345 pcm, then the uncertainty factor to apply to a predicted reactivity value is at least +315 pcm when maximum reactivity is conservative and -345 pcm when minimum reactivity is conservative.

The percent difference is defined as (CMS5 – Reference)/CMS5 in units of per cent. The uncertainty factor in this case should be a multiplier on the CMS5 value consistent with this definition. If a tolerance interval for a parameter is indicated to be -5 percent to +8 percent, the appropriate multiplier range for CMS5 predictions is 1.05 and 0.92.

3.4 NUFs Derived from One-Sided Tolerance Limit Statistical Analysis

This section will describe the formulation of NUFs for the core physics design parameters that lend themselves to the statistical methods described in Section 3.3, excluding the pin power uncertainties. The pin power uncertainties will be discussed in Section 3.5 of this SE.

3.4.1 Core Reactivity

Core wide reactivity agreement for PWRs is characterized in terms of differences in critical boron concentration (ppm) from predictions and measurements. The reactivity worth of a ppm of soluble boron is dependent on fuel enrichment, fuel exposure, and the presence of competing absorbers. The difference in reactivity is expressed in pcm by converting ppm to pcm.

The following items are considered for this calculation:

- Measured boron concentrations can be used including HZP startup testing through HFP operation, estimated critical conditions, and operational transients.

- The B10 / B atom ratio of any particular boron sample can introduce uncertainty to the comparison if it is not accounted for (Ref. 2). Strategies to reduce this uncertainty include:

    - Use boron samples from times where the B10 / B atom ratio is known through measurement (mass spectroscopy), and adjust the measurement or prediction to ensure they are on the same B10 / B basis.

    - Use boron samples from the end of cycle (~ < 50 ppm) where any boron worth error due to the B10 / B atom ratio being different from that assumed in the prediction is minimized.

    - Correct for B10 depletion effects using a depletion model. The most accurate models assume some knowledge of actual leak rates, RCS inventory changes, and the B10 / B atom ratio of the various storage tanks (BAST, RWST, etc.).

- Soluble boron measurement uncertainty is introduced by the titration techniques, an inferred uncertainty is about 10 ppm.

3.4.2 Integral Control Rod Worth

Integral control rod worth (IRW) measurements are performed during HZP startup physics testing. There are several methods used to measure the bank worths including the boron dilution, rod swap, and dynamic rod worth methods (Ref. 1).

There are two items to be considered in the measurement of IRW. First, since rod worth measurements are done using a “reactivity computer” or “reactimeter” that uses point kinetics values from a physics code, a correction for kinetics may be needed when comparing measured-to-predicted values when a different physics code is used. Second, in order to avoid the overestimation of the tolerance limit when the low worth banks are included in the percent differences, a bank worth of 300 pcm or less is excluded from the dataset. This cutoff is acceptable since it retains all but the smallest control rod worth in the difference (between measured and predicted) dataset as per the ANSI test criteria (Ref. 3) when the test criteria switches from percentage basis to a reactivity basis.

3.4.3 Peak Differential Control Rod Worth

When control rod worth is measured using the boron dilution or the dynamic rod worth method, the differential rod worth (DRW) can be measured. Peak DRW differences are calculated as percent differences.

There are two items to be considered in the measurement of peak DRW. First, since rod worth measurements are done using a "reactivity computer" or "reactimeter" that uses point kinetics values from a physics code, a correction for kinetics may be needed when comparing measured-to-predicted values when a different physics code is used. Second, no distinction is made concerning the rod position at which the peak DRW occurs because the most important use of the DRW is typically for the rod withdrawal from a subcritical accident (excessive positive startup rate). The accident is terminated after a few seconds and the maximum DRW is conservatively assumed to occur during the entire withdrawal sequence. Therefore the peak predicted DRW is compared against the peak measured DRW to generate a single percent difference value per measured control rod.

3.4.4 Isothermal / Moderator Temperature Coefficient

Isothermal/Moderator Temperature Coefficient (ITC) is measured at BOC, HZP as part of the startup physics testing (LPPT). ITC differences are expressed in terms of absolute differences, pcm/^{o} F.

There are two items to be considered in the measurement of ITC. First, since ITC measurements are done using a "reactivity computer" or "reactimeter" that uses point kinetics values from a physics code, a correction for kinetics may be needed when comparing measured-to-predicted values when a different physics code is used. Second, although only the ITC is measured, the same comparison statistics are used for the Moderator Temperature Coefficient (MTC) since the MTC is the main component of the ITC measurements. The Doppler coefficient (DTC) portion of the ITC is small and nearly constant at all reactor operating conditions, whereas the MTC component is highly dependent on fuel enrichment, soluble boron concentration, and moderator density.

3.5 Peaking Factor NUFs derived from One-Sided Tolerance Limit Statistical Analysis

This section will summarize how the CMS5 code system is used to determine how peaking uncertainty factors can be obtained from the combination of several physical properties, since in most power reactors there is no method available to directly measure individual pin powers.

3.5.1 Predicted Peak Pin Power

SIMULATE5 can compute pin power by two methods: (1) via SMX method where assembly is divided into N x N submeshes, and for each submesh the flux distribution is determined (Ref. 10). By superimposing pin power form functions (computed by CASMO5) on the flux, the power of each pin is found, (2) by the Fourier Flux Expansion method (Ref. 10) and (3) where the basic geometry unit is not a submesh but an assembly wide node, or quarter assembly node. In both cases the Nuclear Uncertainty Factor for peak pin power is determined by combining the uncertainty from two independent components: a "global" power distribution estimator (normalized thimble reaction rate differences) and a "local" heterogeneity estimator

(pin-to-box ratio). This is followed by the determination of uncertainty for the predicted assembly power and an uncertainty for the pin-to-box ratio and to combine them together to determine an overall uncertainty for 2D / 3D peaking factor quantities such as FΔH/Fr and FQ.

3.5.2 Thimble Reaction Rates

Thimble reaction rate comparisons are used to determine a conservative approximation of the predicted assembly power uncertainty (Ref. 10). It is conservative since it inherently includes not only predicted power uncertainty, but measurement uncertainty and uncertainty associated with reconstructing the predicted thimble reaction rates. Plant flux map data is used for the comparison. For each flux map used in the analysis, measured reaction rates are normalized to the average of all measured reaction rates in instrumented assemblies.

3.5.3 2D Reaction Rate Comparisons with movable and fixed in-core detector systems

Axially integrated reaction rates are used to develop uncertainty factors for 2D quantities FΔH and Fr. For comparisons where both the measured and predicted normalized integral reaction rates are less than 1.0, the differences are discarded because they represent fuel assemblies with less than core average relative power. This is because low power assemblies have higher measurement uncertainty and are not of interest in determining the peaking factor uncertainty.

3.5.4 3D Reaction Rate Comparisons for movable and fixed in-core Detector Systems

Movable in-core detector systems typically take data at a sufficient number of axial locations to suitably represent a 3D quantity such as FQ. The detailed measured axial reaction rates are collapsed to the same nodalization as the CMS5 model (Ref. 1).

Fixed in-core detector systems usually have a small number of axial detectors (4-6) that integrate signals over relatively large portions of the axial core height. For these detectors the SIMULATE5 nodal reaction rates are integrated over the corresponding detector elevations and the normalized measured and predicted reaction rates are computed in the same way that they are for the movable in-core detector systems (Refs. 1 and 10).

The discrepancy in the 3D uncertainty determination with fixed detectors is to relate it to 3-D versus 2D reaction rate statistics by assuming that the sample sizes for both are large enough such that the TL mechanics do not skew the results; 2D uncertainties tend to be smaller than the corresponding 3D uncertainties because the 2D data has been averaged. For the FQ this averaging process appears as a bias. Reference 1, Section 3.5.6, details the calculational procedure for the estimation of this uncertainty.

As a result the following bias is identified:

$$3D\ R.R._{Fixed\ Incore\ Bias} = [\quad\quad] \qquad (10)$$

The total 3D reaction rate tolerance limit for fixed in-core detectors would then become: (Ref. 1)

$$3D\ R.R._{Fixed\ Incore\ Lower\ T.L.} = 3D\ R.R._{Lower\ T.L.} + [\quad\quad] \qquad (11)$$

The $3D\ R.R._{Fixed\ Incore\ Bias}$ component of the fixed in-core 3D tolerance limit has been computed with a large amount of data using specialized tools.

The NRC staff has reviewed the statistical procedure during the regulatory audit (Ref. 6) and determined that the bias value of **[ ]** is acceptable for use by the licensees in their core analysis.

3.5.5 Pin-To-Box Lower Tolerance Limit

This section describes the different methods by which pin-to-box lower TL can be estimated. The NRC staff has previously approved, in Reference 27, a direct SIMULATE-to-MCNP approach to estimate the pin-to-box lower TL. In Reference 27, the pin-to-box uncertainty was computed using two-by-two assembly models using both SIMULATE and MCNP. The uncertainty factor from measured and predicted thimble reaction rate comparisons with the pin-to-box uncertainty factor was derived from comparisons of SIMULATE and MCNP. In this SE, a two-step process (Ref. 36) (CASMO5 to measured, SIMULATE5 to CASMO5) is used to estimate the pin-to-box lower TL component of the peak pin power lower TL.

First step is generation of TLs between CASMO5 and measured pin-by-pin fission rates. The fission rates are a close surrogate to pin powers as discussed in Section 3.2. Figures 3-1 and 3-2 of Reference 1 provide the fission rate tolerances. The large relative errors observed in Figure 3-1 occur in low power pins. Although the TLs could be decreased by discriminating out the low power pins using a cutoff value, the different normalization schemes across the various critical experiments make this challenging and therefore no such discrimination is employed here. The all thermal UO_2 is selected for the first component of the pin-to-box lower TLs

$$CASMO5 - to - Experiment_{Lower\ T.L} = \mathbf{[\ \ ]} \qquad (12)$$

The second step is generation of TLs for the difference between pin-to-box ratios from SIMULATE5 and CASMO5. The comparisons involve a comprehensive range of lattice, enrichment, and burnable poison types. The results of these comparisons are presented in Figure 3-2 and Table 3-4 of Reference 1. The negative tolerance limit for the “all” data from the Table 3-4 of Reference 1 is selected for the second component of the pin-to-box lower TL:

$$\mathbf{[\ \ ]} \qquad (13)$$

The $Pin - to - Box_{Lower\ TL}$ is computed with a large amount of data. The NRC staff has verified and reviewed the procedure through the documents and during the regulatory audit and found that the value of [[-2.3 percent]] is found appropriate for use by the licensees for their core analysis using the CMS5 code system.

3.5.6 *FΔH/Fr* Total Peaking Factor Uncertainty for Movable and Fixed in-core Detector Systems

The total *FΔH/ Fr* peaking factor tolerance limit (%) is the root sum square (RSS) of the 2D reaction rate lower TL and the pin-to-box ratio lower TL (multiplied by -1 to preserve the sense of under prediction):

$$F\Delta H / Fr_{LowerT.L.} = \mathbf{[\ \ ]} \qquad (14)$$

The corresponding NUF is:

$$F\Delta H / Fr_{Upper}\,\text{NUF} = [\quad\quad] \qquad (15)$$

FΔH is the enthalpy rise hot channel factor which is the ratio of maximum coolant enthalpy rise to average coolant enthalpy rise. Fr is the radial nuclear hot channel factor which is the ratio of average heat flux of the hot channel to the average heat flux of the channels in core.

3.5.7 FQ Total Peaking Factor Uncertainty for Movable in-core Detector System

The total *FQ* peaking factor tolerance limit (%) is the root sum square (RSS) of the 3D reaction rate lower TL and the pin-to-box ratio lower TL (multiplied by -1 to preserve the sense of under prediction):

$$FQ_{Lower\ T.L} = [\quad\quad] \qquad (16)$$

FQ is the total nuclear hot channel factor or the nuclear heat flux factor which is the ratio of maximum heat flux in the core to the average heat flux in the core.

3.5.8 FQ Total Peaking Factor Uncertainty for Fixed in-core Detector System

The total *FQ* peaking factor TL (%) is the RSS of the 3D reaction rate lower TL (including the *FQ* bias described in Section 3.5.4 of the SE) and the pin-to-box ratio lower TL (multiplied by -1 to preserve the sense of under prediction):

$$FQ_{Lower\ T.L.} = [\quad\quad] \qquad (17)$$

The corresponding NUF is:

$$FQ_{Upper\ NUF} = [\quad\quad] \qquad (18)$$

3.5.9 NRFs Derived from Engineering / Physical Arguments

NRFs have been discussed in Section 3.3.1 of this SE. This section defines NRFs for differential boron worth, Doppler temperature/power coefficient, and delayed neutron and prompt neutron lifetime.

Differential boron worth is a quantity that is often measured at BOC, HZP as part of startup physics testing. This would seem to make it a good candidate for using the one-sided statistical TL to determine its uncertainty factor. The NRC staff approved the amendment request (Ref. 27) for Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations. Based on Reference 27, the upper and lower NRFs for differential boron worth is set to: (See Section 3.3.1 of the SE).

$$\textit{Differential Boron Worth}_{\ Upper\ NRF} = [\quad\quad] \qquad (19)$$

Differential Boron Worth $_{Lower\ NRF}$ = **[ ]** (20)

Direct determination of a NRF for Doppler feedback remains very difficult. Benchmarking of CASMO5 Doppler Temperature Defects to Monte Carlo methods (Section 3.2.3 of this SE) suggests that there is very little bias contributed from the CASMO5 cross sections. The good agreement of the measured and predicted axial offset oscillation magnitude in the modeled operational transient demonstrates that the xenon-Doppler balance is well predicted. Also based on Reference 27, the upper and lower NRFs for Doppler coefficient are written as below:

Doppler Coefficient $_{Upper\ NRF}$ = **[ ]** (21)

Doppler Coefficient $_{Lower\ NRF}$ = **[ ]** (22)

Similarly, the kinetics parameters, delayed neutron and prompt neutron lifetime upper and lower NRFs are set to the same values as in Reference 27:

Kinetics Parameter $_{Upper\ NRF}$ = **[ ]** (23)

Kinetic Parameter $_{Lower\ NRF}$ = **[ ]** (24)

3.5.10 Conclusion

Studsvik has presented one-sided TL statistical analysis methodology to generate TLs, NUFs, and NRFs for most physics parameters important to core design and safety analysis. For those parameters that do not use the statistical analysis method, NRFs have been presented. The NRC staff has reviewed the documents and calculation notebooks during its review and during the regulatory audit and determined that NUFs and NRFs for most physics parameters for the licensees' core design analysis are acceptable.

3.6 CMS Benchmark, NUF and NRF Methodology Demonstration

This section describes how Studsvik used 63 cycles of measured plant data for CMS5 benchmarking, NUF methodology demonstration, and the development of NRFs.

For benchmarking, of the CMS5 system, CASMO5 cross sections are generated and passed to the SIMULATE5 core models. This integrated benchmark tests many of the calculations that can be performed by the CMS5 system such as:

1. Core reactivity (soluble boron concentration / boron worth)
2. Flux and power distributions (radial, axial, local peaking)
3. Fuel burnup
4. Fuel and burnable poison nuclide concentrations as a function of fuel burnup
5. Integral and differential control rod bank worths
6. Moderator and Doppler temperature coefficients and defects
7. Operational transient simulation
8. Generation of delayed neutron parameters and prompt neutron lifetime

9. Detector reaction rates, coupling coefficients, and peaking factors for flux map analysis

10. Fixed source neutron multiplication

11. Ex-core detector response

The second use of the 63 cycle measured plant data is to provide a worked example of the uncertainty factor Methodology. NUFs are created from the difference data generated for the benchmark for the physics parameters.

The third use is to formulate sufficiently conservative PWR NRFs based upon the NUFs generated so that licensees may use them for their core design analyses.

Five PWR plants were selected for this benchmarking process, they vary from 14x14 to 17x17 with movable, fixed in-core detector types, and 2, 3, 4 loops (Ref. 1, Table 4-1). Tables 4-2 through 4-8 of Reference 1 provide details of each plant cycle that has been benchmarked including thermal power (range 1650 MW(th) to 3650 MW(th), burnable absorber used (Gadolinia, IFBA, WABA, and B_4C), cycle fuel exposures (ranging from 13.321 GWd/MTU to 20.970 GWd/MTU), and enrichments (ranging from 2.15 wt% of U-235 to 4.95 wt%).

3.6.1 CMS5 Benchmark – Physics Parameters

The benchmarking process for CMS5 using data from 63 cycles of the 5 PWR plants comprise of physics parameters such as, critical boron comparisons (core reactivity), control rod worths (integral and peak differential), isothermal temperature coefficients, and reaction rate comparisons from flux maps taken at various times during cycle life. Key approximations that were used for benchmarking include using nominal enrichment and average fuel assembly loadings. The SIMULATE5 models consists of options such as; full core radial model, 32 axial nodes, radial sub-mesh option, and four energy groups.

Core Reactivity (Critical Boron Concentration)

CMS5 code capability is assessed by comparing predicted critical boron concentration (CBC) against plant measurements. Table 4-9 of Reference 1 lists the results of the core reactivity for HZP (hot zero power) and HFP (hot full power) at various stages of the cycle. Differences are expressed in terms of reactivity by converting the ppm differences to pcm differences using the predicted boron worth.

The results for the boron worth (core reactivity) as reported in Table 4-9 of Reference 1 has been found within the criteria set forth in the ANSI/ANS standard (Ref. 35) for reload startup physics tests for PWRs.

Integral Control Rod Bank Worth

The 63 cycle benchmark data includes control rod worths measured by the boron dilution, rod swap, and dynamic rod worth methods with reactivity computer corrections made to the measurements. Table 4-10 of Reference 1 presents the results of the integral rod worth benchmark including mean difference, standard deviation, number of observations, minimum, and maximum (%). The NRC staff has verified the results of the benchmark during the

regulatory audit. The staff finds that the results are in agreement with the ANS/ANSI standard for physics measurements.

Peak Differential Control Rod Bank Worth

The control rod worth that was measured using the rod swap or dynamic rod worth method, the peak predicted differential control rod bank worth, was compared against the peak measured differential control rod bank worth regardless of where the peak actually occurred in each case. Reference 3, Table 4-11, lists the results of the benchmarking. The staff verified the results and calculations during the regulatory audit (Ref. 6) and determined the results are reasonable.

Isothermal Temperature Coefficient

All of the 63 cycle benchmark data includes BOC HZP ITC measurements as part of the startup physics testing. For this benchmark, a heatup and cooldown of about 5 degree F was averaged and compared against the measured ITC. Table 4-12 of Reference 1 lists the results of ITC benchmarking. The NRC staff verified the results and reviewed the calculations during the regulatory audit and determined that the benchmarking results are in reasonable agreement with the ANSI/ANS standard criteria.

2D/3D Reaction Rate Comparisons

Predicted 2D/3D reaction rates are compared against corresponding measured reaction rates that are derived from flux map surveillances. Tables 4-13 and 4-14 of Reference 1 present the results of the 2D reaction rate comparisons for movable and fixed incore detector systems respectively. Tables 4-15 and 4-16 of Reference 1 present the results of the 3D reaction rate comparisons for movable and fixed incore detector systems respectively. The NRC staff has verified the process and the results of the benchmark and determined that the process and the results are consistent with the Section 3.5 of this SE and found that the results are consistent.

3.7 CMS5 Benchmark – Plant Transients

Three normal operational plant transients models are selected to demonstrate CMS5 subcritical fixed source solver and ex-core detector models in the code system. Normal operational transients involve power changes, control rod movements, temperature variations, xenon concentration and distribution changes, and boron concentration changes. These variations are useful for demonstrating the ability of the CMS models to accurately predict core behavior associated with the above changes. The following subsections provide results of transient from three plants for specific cycles.

Plant "D" Unit 2 Cycle 14 Transient

For this transient, the plant power was reduced from HFP equilibrium conditions to approximately 27 percent power. After holding at low power for about 10 hours, a ramp to 100 percent power over about 11 hours was initiated.

Figures 4-1 through 4-3 of Reference 1 illustrates the results of the Plant "D" transient. Figure 4-1 shows the control rod position and power level versus time for the duration of the transient. Figure 4-2 compares the measured (as recorded by ex-core nuclear instrumentation)

delta-I (axial offset multiplied by core percent power) and CMS5 predicted delta-I versus time. Figure 4-3 compares the measured and CMS5 predicted reactivity in the form of critical boron concentration versus time. The measured delta-I indicates stable or slightly damped natural axial core behavior. SIMULATE5 demonstrates excellent agreement with the timing and magnitude of the axial oscillation.

Plant "B" Unit 2 Cycle 23 Transient

This Plant "B" transient consists of a shutdown using only boron with rods-out position for the duration of the shutdown.

Figures 4-4, 4-5, and 4-6 of Reference 1 plot the results of the transient. Figure 4-4 shows the control rod position and power level variation with time for the duration of the transient. Figure 4-5 compares the variation of measured Axial Shape Index (ASI) and CMS5 predicted ASI with time. Figure 4-6 compares the measured and CMS5 predicted reactivity in the form of critical boron concentration variation time.

Figure 4-5 shows that predicted power agrees very well with measured power. Figure 4-6 shows reasonable agreement between measured and SIMULATE5 ASI. Figure 4-6 (CBC vs. time) shows the predicted CBCs are within 10 ppm of the available measured points.

Plant "A" Cycle 27 Transient

For Plant "A" transient, power was reduced from HFP equilibrium conditions to approximately 40 percent power. After holding at low power for approximately 65 hours, a ramp to 100 percent power over about 12 hours was initiated.

Figures 4-7, 4-8, and 4-9 of Reference 1 plots the results for this transient. Figure 4-7 shows the control rod position and power level versus time for the duration of the transient. Figure 4-8 compares the measured delta-I and CMS5 predicted delta-I versus time. Figure 4-9 compares the measured and CMS5 predicted reactivity in the form of critical boron concentration versus time. The NRC staff reviewed the results and determined that there is reasonable agreement between predicted and measured values; the CMS5 delta-I agrees within 2 percent of measurements during the transient and the predicted reactivity agrees within 32 ppm of the measured reactivity.

Fixed Source and Ex-core Detector Model Demonstration

CMS5 is capable of calculating fixed source solution in a subcritical core with both the internal neutron sources from burnt fuel as well as from external sources such as secondary source rods. The CMS5 has capability to model ex-core detector response. For this transient the two models are combined to model a hypothetical transient in which k_{eff} increases from subcritical value to just critical (k_{eff} = 1). The inverse count rate ratio (ICCR) for each k_{eff} is calculated as the ex-core detector response in neutrons/cm^2 per second for that statepoint divided into the beginning detector response similar to constructing a 1/M plot for the reactor startup process.

For the Plant "E" transient, the model is analyzed at HZP, BOC conditions to demonstrate the ex-core detector response for the three situations: (1) no secondary source, (2) a secondary

source placed two fuel assembly rows away from the ex-core detector, and (3) a secondary source placed three fuel assembly rows away from the ex-core detector.

In each of the above cases the CMS5 results are compared against the results from a corresponding MCNP6 model. Figure 4-10 of Reference 1 plots the normalized ICCR versus $k_{eff.}$ The NRC staff has reviewed and found that the CMS5 results reasonably agree with the MCNP6 results.

3.7.1 Nuclear Uncertainty Factor Generation

The NUF methodology described in Sections 3.3 and 3.4 of this SE and in Reference 1 is demonstrated using the 63 cycle benchmark data.

The individual plant and combined reactivity discussed in Section 3.6.1 of the SE and in Table 4-17 of Reference 1 were tested for normality and none of the data passed the test. Based on the number of observations, the appropriate m^{th} value was selected to set the TLs. The upper and lower NUF for core reactivity was identified to be **[ ]** and **[ ]**, respectively.

The individual plant and combined integral rod bank worth data sets summarized in Table 4-10 of Reference 1 and discussed in Section 3.6.1 of the SE were tested for normality. All of the data sets passed the normality test so an appropriate K multiplier value was calculated and then combined with the mean and standard deviation to determine the TLs. Equations (5) and (6) are used to calculate the lower and upper TLs. Table 4-18 of Reference 1 lists the upper and lower TLs and the corresponding NUFs. The lower NUF is calculated to be **[ ]** and the upper NUF is found to be **[ ]**.

The individual plant and combined peak differential control rod bank worth (Section 3.61 of SE) data sets summarized in Table 4-11 of Reference 1were tested for normality. Most except Plant “E” and the combined data set did not pass the test. The appropriate m^{th} value or K multiplier is used to determine the TLs. Table 4-19 of Reference 1 lists the peak differential control rod bank worth TLs and NUFs. The upper NUF value is found to be **[ ]** and the lower NUF value is found to be **[ ]**.

For the isothermal and moderator temperature coefficient NUF, the data sets discussed in Section 3.6.1 of the SE and listed in Table 4-12 of Reference 1 were tested and passed for normality. An appropriate K value was determined using equation (7). Table 4-20 of Reference 1 lists the isothermal and moderator temperature coefficient tolerance limits and NUFs. The upper NUF is **[ ]** and the lower NUF is **[ ]**.

The first step in determining the *FΔH/Fr* NUF is to determine the 2D integral reaction rate lower TL for the movable in-core detector plants. The individual plant datasets are non-normal and the combined dataset is greater than 5000 observations so non-normality is assumed. The individual plant dataset is non-normal and the combined dataset has over 5000 observations and non-normality is assumed. Using equations (14) and (15), the lower and upper TLs for FΔH are calculated. Table 4-21 lists the movable in-core detector FΔH/Fr TL and NUF. The upper NUF value is found to be **[ ]**.

Table 4-22 of Reference 1 lists the FΔH/Fr fixed in-core detector TLs and NUFs. The upper NUF value for FΔH/Fr is found to be **[ ]**.

For the determination of FQ movable in-core detector NUF, the first step is to determine the 3D nodal reaction rate lower TL for the movable in-core detector plants. Since all of the datasets involved had a number of observations outside the range of the Shapiro-Wilks normality test, the data is treated as non-normal. The combined data set nonparametric m^{th} value based on 187882 observations is 9239, and the 9239th most negative value on the sorted combined list is **[ ]**. Equations (17) and (18) are used to determine the lower and upper TLs for FQ. Table 4-23 lists the reaction rate lower TL and the upper NUF for FQ. The upper NUF for FQ is listed as **[ ]**.

For the fixed in-core detector FQ, the NUF is determined much like the movable in-core detector NUF with the addition of the fixed in-core FQ bias value (Section 3.5 of SE). The dataset is not normal and the 201^{st} (based on 4498 observations) most negative 3D reaction rate difference is **[ ]**. FQ lower and upper TLs are calculated using equation (17) and (18) respectively. The NUF for fixed in-cored detector FQ TL is found to be **[ ]**.

3.8 Generic Nuclear Reliability Factors

The 63 cycle benchmark used design and operational data which is representative of nearly all operating PWRs. Table 1 lists the generic NRFs and can be used for in-scope applications by licensees for their core design analysis. The Table-1 also lists the corresponding NUFs on which the NRFs are based. The NRFs are considered more conservative than their corresponding NUFs. This conservatism in NRFs reduces the risk when there are small variations in the NUFs due to change in software or model updates resulting in change in NRFs as long as the NRFs remain bounding.

Table 1 Nuclear Uncertainty Factors and Generic Nuclear Reliability Factors

Physics Parameter	Upper NUF	Lower NUF	Upper NRF	Lower NRF
Core Reactivity (Critical Boron Concentration)(pcm)	[ ]	[ ]	[ ]	[ ]
Integral Control Rod Bank Worth (Individual Bank and Total of All Banks)	[ ]	[ ]	[ ]	[ ]
Peak Differential Control Rod Bank Worth	[ ]	[ ]	[ ]	[ ]
Isothermal and Moderator Temperature Coefficient	[ ]	[ ]	[ ]	[ ]
FΔH / Fr (Movable and Fixed In-core detectors)	[ ]	N/A	[ ]	N/A
FQ (Movable and Fixed In-core detectors)	[ ]	N/A	[ ]	N/A
Differential Boron Worth			[ ]	[ ]
Doppler Temperature / Power Coefficient			[ ]	[ ]
Kinetics Parameters (Delayed Neutron and Prompt Neutron Lifetime)			[ ]	[ ]

3.9 Conclusion

The NRC staff has reviewed the benchmark performed using 63 cycle data and has found agreement between CMS5 predictions and measurements in core reactivity, integral and peak differential control rod bank worth, isothermal temperature coefficient, and 2D/3D reaction rates.

The nuclear reliability factors are conservatively proposed for PWR core designs using CMS5 methodology for the physics parameters.

4.0 LIMITATIONS/CONDITIONS

The NRC staff limits the applicability of the Studsvik Scandpower Core Management System (CMS5) for PWRs TR, SSP-14-P01/028-TR-P to the following material or conditions of fuel, cladding, poison, and other core materials specified below:

1. Fuel and Poison

- Uranium Oxide fuel enriched up to 5 wt% U-235
- Pin lattice geometries ranging from 14x14 to 17x17 including both large and small water hole designs
- The typical range for nominal density between 10.3 and 10.9 g/cc (or between 0.94 and 0.985 as a fraction of theoretical density).
- Integral burnable absorber Gd_2O_3 with range up to 12 percent as a mass fraction of the total fuel weight.
- Integral burnable absorber (IFBA) ZrB_2
- Discrete absorber types WABA, B_4C-AlO_3, Boron Silicate Glass, and Hafnium suppressor rods

1. Cladding

- Zircaloy-2, Zircaloy-4
- ZIRLO and Optimized ZIRLO
- M5
- TVEL cladding materials (E110 and E635)
- Japanese cladding material (NDA and MDA)
- CMS5 code system can handle Zr based cladding materials with trace elements of other materials (such as Nb, Sn, Fe, Cr, Ni, and O)

2. Structural Materials (but not limited to)

- Stainless Steel, Inconel-718, Inconel-750

3. Fuel Burnup

CMS5 is capable of handling burnups listed below:

- Peak average assembly burnup **[ ]** GWd/MTU
- Peak average fuel rod burnup **[ ]** GWd/MTU
- Peak pellet burnup **[ ]** GWd/MTU

However, the NRC staff has restricted the rod average fuel burnup up to 62 GWd/MTU for all currently approved types of cladding. For fuel designs with new cladding material, change in the fuel rod burnup shall be subjected to the burnup approved by the NRC staff for the new fuel/cladding design.

4. Change Management

Any change management with respect to addition of new features, new functionality, correction of software errors, and/or usage of additional data from operating reactors/test reactors/Halden must ensure that the nuclear reliability factors generated by exercising the methodology reported in this CMS5 TR remain conservative.

5. Resonance upscatter model in CASMO5

The resonance upscatter model (RUP) described in the CASMO5 methodology manual and further described in response to an RAI has not been implemented in the CMS5 application. The NRC staff has reviewed the RUP model and determined that Studsvik may use the RUP factors and apply to the resonance integral (RI) in CASMO5.

5.0 SUMMARY AND CONCLUSION

Studsvik CMS5 topical report that consists of CASMO5/CMSLINK5/SIMULATE5 codes has been reviewed by the NRC staff. The accuracy of the CMS5 code system has been demonstrated through a set of benchmarks and validation to critical experiments, higher order codes, and a 7 unit (nuclear plant)/63 cycle comparison of predictions to PWR plant data.

Studsvik has generated the NUFs that are created from the difference data generated for the benchmark for the physics parameters for which the CMS5 predictions can be compared against the measurements or higher order codes.

Based on the 63 cycle benchmark data that includes a wide array of PWR design and operating data, a set of conservative generic NRFs were determined to account for predictive bias and uncertainty.

The NRC staff has reviewed the CASMO5/SIMULATE5 models, validation and benchmarking of the models using critical experiments, higher order code comparisons, TL statistical analysis, NUFs derived from one-sided TL analysis and the development of conservative generic NRFs. The staff concludes that the CASMO5/SIMULATE5 models together with the generic NRFs are suitable for core design, analysis, and depletion calculations.

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Attachment: Resolution of Comments

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Date: September 15, 2017

Comment Summary for Draft Safety Evaluation for SSP-14-P01/028-TR, "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors"

Page No.	Line(s) No.	Comment	NRC Resolution
1	1	SS-14-P01.028-TR is missing 'P' and should read 'SSP-14-P01/028-TR'	Comment accepted. Change incorporated in final SE.
2	8	"[…] arrange up to" should read "a range up to"	Comment accepted. Change incorporated in final SE.
2	14	"gaseous waste disposal" should be: "gigawatt days"	Comment accepted. Change incorporated in final SE.
2	17	"CASMO55" should be "CASMO5"	Comment accepted. Change incorporated in final SE.
4	40	Replace 'performed' with 'determined'	Comment accepted. Change incorporated in final SE.
6	3	"ENDF/B-VII" should be "ENDF/B-VII.1"	Comment accepted. Change incorporated in final SE.
8	38	"SIMULAT5" should be "SIMULATE5"	Comment accepted. Change incorporated in final SE.
9	34-36	Suggest text change from: "This quadratic function models the variations of the spatial shielding effects over the depletion step and therefore improves the overall accuracy of depletion calculations (Ref. 2). This quadratic function models the variations of the spatial shielding effects over the depletionstep and improves the overall efficiency of depletion calculations" To: "This quadratic function models the variations of the spatial shielding effects over the 34 depletion step and therefore improves the overall accuracy of depletion calculations (Ref. 2)."	Comment accepted. Change incorporated in final SE.
11	32	"heat model" should be "heat structure mode"	Comment accepted. Change incorporated in final SE.

Page No.	Line(s) No.	Comment	NRC Resolution
15	17-21	Change the sentence: " The DIMPLE experimental facility consists of water-reflected cylindrical systems to power reactor geometries by assembling a cruciform array of 3 percent-enriched uranium dioxide fuel pins arranged in an array that simulated the rectangular corner configuration of a PWR and 21 effectively represented twelve PWR fuel assemblies (Figure 3-22 of Ref. 4)." To: "The DIMPLE experimental facility consists of a water-reflected cylindrical system to model power reactor geometries by assembling a cruciform array of 3 percent-enriched uranium dioxide fuel pins arranged to simulate the rectangular corner configuration of a PWR and effectively represented twelve PWR fuel assemblies (Figure 3-22 of Ref. 4)."	Comment accepted. Change incorporated in final SE.
15	26	"CASO5" should be "CASMO5".	Comment accepted. Change incorporated in final SE.
16	37	"Japan Atomic Energy Research Institute PER Isotope Benchmarks" Should read "[…] PWR Isotope Benchmarks".	Comment accepted. Change incorporated in final SE.
17	28	**[ ]** should be bracketed as proprietary as in the previous portion of the paragraph.	Comment accepted. Change incorporated in final SE.
17	37	The sentence: "… SIMULATE5 treatment temperature solution and determine …" Should be updated to: "… SIMULATE5 fuel temperature solution and determined …"	Comment accepted. Change incorporated in final SE.
18	33	Change "to" to "using the same"	Comment accepted. Change incorporated in final SE.
18	44	"MCNCP6" should be changed to "MCNP6"	Comment accepted. Change incorporated in final SE.
24	20	"equation 6" should be "equation 8"	Comment accepted. Change incorporated in final SE.
24	24	"Equation (7)" should be "Equation (9)"	Comment accepted. Change incorporated in final SE.

Page No.	Line(s) No.	Comment	NRC Resolution
26	12	Insert a space between 'therod' to become "the rod"	Comment accepted. Change incorporated in final SE.
26	24	"… peak DRW." should be changed to "… ITC."	Comment accepted. Change incorporated in final SE.
28	23	"p-to-box" should be "pin-to-box"	Comment accepted. Change incorporated in final SE.
29	39	"9" should be replaced by ":"	Comment accepted. Change incorporated in final SE.
31	10	Insert a period after the sentence that ends "… analyses."	Comment accepted. Change incorporated in final SE.
32	9	Reference 3 should be inserted before Table 4-11.	Comment accepted. Change incorporated in final SE.
32	43	"HZP" should be changed to "HFP"	Comment accepted. Change incorporated in final SE.
33	33	"32 percent" should be changed to "32 ppm"	Comment accepted. Change incorporated in final SE.
34	12	"Table 4-9" should be "Table 4-17"	Comment accepted. Change incorporated in final SE.
34	43	The value **[ ]** should be bracketed.	Comment accepted. Change incorporated in final SE.
34	46	The value **[ ]** should be bracketed.	Comment accepted. Change incorporated in final SE.
36	Table 1	The values of the NUF's and NRF's should all be bracketed.	Comment accepted. Change incorporated in final SE.
36	16	TR document number should be "SSP-14-P01/028-TR-P"	Comment accepted. Change incorporated in final SE.
37	20-21	Suggest changing the text: "CMS5 code system can handle cladding materials with composition Zr, Nb, Sn, Fe, Cr, Ni, and O" To "CMS5 code system can handle Zr based cladding materials with trace elements of other materials (such as Nb, Sn, Fe, Cr, Ni, and O)"	Comment accepted. Change incorporated in final SE.

Page No.	Line(s) No.	Comment	NRC Resolution
37	35-36	Suggest changing the text: "However, the NRC staff has restricted the rod average fuel burnup up to 52 GWd/MTU for all approved types of cladding." To "However, the NRC staff has restricted the rod average fuel burnup for all approved types of cladding."	The final SE was revised to state the following: “However, the NRC staff has restricted the rod average fuel burnup up to 62 GWd/MTU for all currently approved types of cladding. For fuel designs with new cladding material, change in the fuel rod burnup shall be subjected to the burnup approved by the NRC staff for the new fuel/cladding design.”
37	38-42	Suggest changing the text: "Any change management with respect to addition of new features, new functionality, correction of software errors, and/or usage of additional data from operating reactors/test reactors/Halden must ensure that the nuclear reliability factors reported in this CMS5 TR remain conservative." To "Any change management with respect to addition of new features, new functionality, correction of software errors, and/or usage of additional data from operating reactors/test reactors/Halden must ensure that the nuclear reliability factors generated by exercising the methodology reported in this CMS5 TR remain conservative."	Comment accepted. Change incorporated in final SE.
38	39	"SSP-14/P01-028-TR-P", should be: “SSP-14-P01/028-TR-P"	Comment accepted. Change incorporated in final SE.
38	42	"SSP-14/P01-028-TR-P", should be: “SSP-14-P01/028-TR-P"	Comment accepted. Change incorporated in final SE.

Page No.	Line(s) No.	Comment	NRC Resolution
38	46	SSP-14/P01/012-R" should be: "SSP-14-P01/012-R"	Comment accepted. Change incorporated in final SE.
39	13	"(Proprietary)" should be inserted at the end of the reference.	Comment accepted. Change incorporated in final SE.
39	15	"(Proprietary)" should be inserted at the end of the reference.	Comment accepted. Change incorporated in final SE.