

SENSITIVITY AND UNCERTAINTY ANALYSIS OF COMMERCIAL REACTOR CRITICALS FOR BURNUP CREDIT

KEYWORDS: *burnup credit, commercial reactor criticals, sensitivity and uncertainty analysis*

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This paper provides insights into the neutronic similarities between a representative high-capacity rail-transport cask containing typical pressurized water reactor (PWR) spent nuclear fuel assemblies and critical reactor state-points, referred to as commercial reactor critical (CRC) state-points. Forty CRC state-points from five PWRs were analyzed, and the characteristics of CRC state-points that may be applicable for validation of burnup-credit criticality safety calculations for spent fuel transport/storage/disposal systems were identified. The study employed cross-section sensitivity and uncertainty analysis methods developed at Oak Ridge National Laboratory and the TSUNAMI set of tools in the SCALE code system as a means to investigate neutronic similarity on an integral and nuclide-reaction-specific level. The results indicate that except for the fresh-fuel-core configuration, all analyzed CRC state-points are either highly similar, similar, or marginally similar to the representative high-capacity cask containing spent nuclear fuel as-

semblies with burnups ranging from 10 to 60 GWd/t U in terms of their shared uncertainty in k_{eff} due to cross-section uncertainties. On a nuclide-reaction-specific level, the CRC state-points provide significant coverage, in terms of neutronic similarity, for most of the actinides and fission products relevant to burnup credit. Hence, in principle, the evaluated CRC state-points could serve as part of a set of benchmark experiments for determining a bias and bias uncertainty to be applied to the calculated k_{eff} of a spent fuel transport/storage/disposal system to correct for approximations in computational methods and errors and uncertainties in nuclear data. Note, however, that an evaluation to quantify the uncertainties associated with various CRC modeling parameters (e.g., fuel isotopic compositions, physical characteristics of reactor core components, and reactor operating history information), which has relevance to the use of these critical configurations for bias determination, was not performed as part of this study.

I. INTRODUCTION

A commercial reactor critical (CRC) state-point is either (a) a hot zero-power (HZP) critical condition attained after sufficient cooling time to allow the fission-product xenon inventory to decay or (b) an at-power equilibrium critical condition at which xenon worth has reached an equilibrium value. The CRC state-points were selected for modeling because the data describing each state-point are near steady state and are relatively well

defined compared with more transient conditions. CRC state-points are the only critical experiments that include a significant amount of actual spent nuclear fuel representative of the waste stream assemblies. CRC state-point description data, including spent-fuel-assembly irradiation parameters, physical characteristics of reactor core components, geometry data, and spent fuel isotopic compositions, have been developed as a component of the criticality analysis methodologies for disposal of commercial spent nuclear fuel and are publicly available.¹⁻³

Owing primarily to the differences in physical characteristics and environmental conditions between CRC

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configurations and spent nuclear fuel in storage and transport casks, the applicability of CRC state-points to validation of burnup-credit criticality calculations has been repeatedly called into question. For example, one concern that has been raised is that the neutron multiplication factor of the CRC state-points is driven primarily by the low-burnup (fresh, or first-cycle) fuel. This emphasis on low-burnup fuel would limit the influence to the system reactivity of the fuel with any significant burnup, thereby limiting the usefulness of the CRC experiments for burnup-credit validation.⁴ A number of studies⁵⁻⁷ have been previously carried out to demonstrate the suitability of CRC state-points as spent-fuel benchmark experiments and their applicability for burnup-credit criticality calculation validation, but a conclusive demonstration of applicability has not yet been published. These previous studies focused on neutron spectra comparisons, determining trends in calculated neutron multiplication factor (k_{eff}) values, and assessing the effects on k_{eff} of specific system characteristics such as material temperatures.

The recent development and release of the Tools for Sensitivity and Uncertainty Analysis Methodology Implementation⁸⁻¹⁰ (TSUNAMI) in the Standardized Computer Analyses for Licensing Evaluation (SCALE) 5 computer code package¹¹ provides the ability to evaluate the relevant issues and concerns related to neutronic similarity between the CRC state-points and the spent fuel in storage and transport casks by calculating and comparing detailed information on the physics of the systems. In particular, sensitivity and uncertainty (S/U) analysis methods can be used to demonstrate that nuclear systems with similar physical characteristics, including material compositions, geometry, and neutron flux spectra, exhibit similar sensitivities of the effective neutron multiplication factor, k_{eff} , to perturbations (i.e., small errors) in the neutron cross-section data on an energy-dependent, nuclide-reaction-specific level. The assessment of system similarities and shared variance, due to cross-section uncertainties, in the computed values of k_{eff} is particularly useful in determining applicability of critical experiments to be used to determine the bias and bias uncertainty of criticality safety calculations. Note that the term “applicability” is used throughout this paper to indicate that sufficient neutronic similarity between the models of an evaluated CRC state-point and of the representative cask has been determined so that the CRC state-point could be used as a component of a benchmarking methodology, provided that the CRC state-point satisfies other benchmark quality requirements.

This paper provides a detailed evaluation of the neutronic similarities between a representative high-capacity, rail-type burnup-credit cask containing typical spent pressurized water reactor (PWR) fuel assemblies [Westinghouse 17 × 17 Optimized Fuel Assembly (OFA) of 3.777 wt% ²³⁵U initial enrichment, 40 GWd/t U burnup, and a 5-year cooling time] and 40 CRC state-

points and identifies the state-points that are most similar to the representative high-capacity cask model. The neutronic similarity was established in relation to sensitivity-based indices and parameters computed with the TSUNAMI tools, including correlations between sensitivity-weighted cross-section uncertainties in the application system and each of the evaluated CRC state-points and comparisons of energy-dependent sensitivities for nuclides relevant to the reactivity of spent nuclear fuel. S/U analyses and results for the representative burnup-credit cask model containing typical spent fuel assemblies with burnup values ranging from 10 to 60 GWd/t U, which are available in Ref. 12, are briefly discussed.

The data for the evaluated CRC state-points were obtained from five PWRs of various designs: Crystal River Unit 3, Sequoyah Unit 2, Surry Unit 1, Three Mile Island (TMI) Unit 1, and North Anna Unit 1. Note that the evaluated CRC data were taken before most of the nuclear industry moved to extended cycles (18- and 24-month cycles) and reactor power uprates. Currently, most of the commercial reactors use higher enrichments (many close to the 5 wt% limit), and some use complex fuel designs such as axial and asymmetric enrichments. These more recent and potentially more complex fuel designs are not represented in the CRC state-points evaluated in this study.

In addition to neutronic similarity, other characteristics, such as material and environmental condition uncertainties, are relevant to the process of selecting critical experiments for validation of codes and nuclear data used for criticality safety evaluations. Criticality validation is based on benchmark-quality critical experiments, which are generally small critical experiments that are performed in a controlled laboratory setting and have small, typically quantified, uncertainties. In contrast, the CRC state-points are large, complex, near-steady-state configurations at a nuclear utility that include considerable uncertainty (e.g., isotopic compositions of the burned fuel, operating history data, and material temperatures). Although the issue of quantification and/or understanding of the uncertainties present in the CRC state-points is important and relevant to their use in criticality validation, the focus of this paper is on neutronic similarity between CRC state-points and spent nuclear fuel storage and transport casks. Hence, this paper addresses neither the uncertainties in the existing CRC state-point data and models nor the influence of these uncertainties on a bias determined based on the CRC state-points.

The remainder of this paper is organized in the following manner: Section II describes the application system, the evaluated CRC state-points, and the KENO V.a modeling for TSUNAMI calculations; Sec. III presents the TSUNAMI cross-section S/U analysis methods and results; and Sec. IV presents the conclusions of the S/U analyses.

II. EVALUATED SYSTEMS AND KENO V.a MODELING

The application system for the S/U analysis is a representative high-capacity spent fuel transport cask containing typical PWR spent fuel assemblies of 3.777 wt% ^{235}U initial enrichment, 40 GWd/t U burnup, and a 5-yr cooling time. The representative high-capacity cask design, referred to as the GBC-32 cask, has been proposed as a reference configuration for burnup credit studies.¹³ The representative cask, which can accommodate 32 PWR assemblies, utilizes Boral panels containing ^{10}B as a fixed neutron poison dispersed uniformly to reduce neutronic interaction between adjacent fuel assemblies.

The KENO V.a cask model consists of identical spent fuel assemblies of Westinghouse 17×17 OFA type. Pin geometry models contain 18 equal-length axial regions that allow specification of different fuel compositions to represent the axial variation in fuel burnup. Figure 1 is a cutaway view of the KENO V.a cask model, illustrating the bottom half of the cask with one-quarter of the model removed. The axially dependent fuel compositions representing the axial variation in assembly burnup were automatically generated using the STARBUCS sequence¹⁴

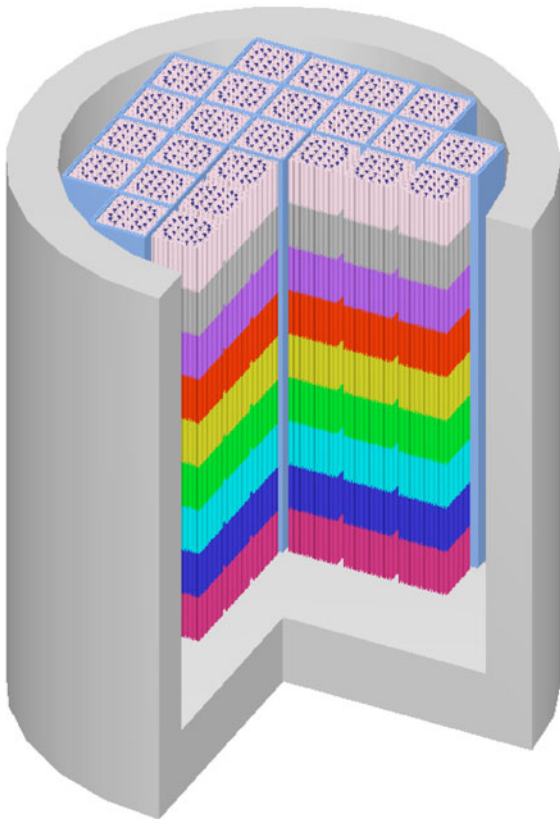


Fig. 1. Cutaway view of the KENO V.a model for the GBC-32 cask, showing the bottom half of the cask with one-quarter of the model removed.

in SCALE and a built-in bounding burnup-dependent axial profile.^{15,16} The nuclides included in the fuel compositions consist of the following actinides and fission products that have been identified in Ref. 4 as being important for burnup credit criticality calculations: ^{234}U , ^{235}U , ^{236}U , ^{238}U , ^{238}Pu through ^{242}Pu , ^{241}Am , ^{243}Am , ^{237}Np , ^{95}Mo , ^{99}Tc , ^{101}Ru , ^{103}Rh , ^{109}Ag , ^{133}Cs , ^{147}Sm , ^{149}Sm through ^{152}Sm , ^{143}Nd , ^{145}Nd , ^{151}Eu , ^{153}Eu , and ^{155}Gd . Note that the irradiated fuel compositions in the GBC-32 model were not adjusted to account for differences between measured and calculated isotope compositions (i.e., isotopic validation). For the enrichment, burnup, and cooling-time conditions described above, the flooded GBC-32 cask has a k_{eff} value of 0.8776 ± 0.0004 and energy of average lethargy of neutrons causing fission (EALF) of 0.284 eV.

The CRC state-points evaluated in this paper are comprised of 40 state-points from various PWRs of Babcock and Wilcox (B&W) and Westinghouse reactor designs, including Crystal River Unit 3 (B&W reactor design of 177 fuel assemblies of 15×15 array, 33 state-points); Sequoyah Unit 2 (Westinghouse reactor design of 193 fuel assemblies of 17×17 array, 3 state-points); Surry Unit 1 (Westinghouse reactor design of 157 fuel assemblies of 15×15 array, 2 state-points); TMI Unit 1 (B&W reactor design of 177 fuel assemblies of 15×15 array, 1 state-point); and North Anna Unit 1 (Westinghouse reactor design of 157 fuel assemblies of 17×17 array, 1 state-point). Detailed descriptions of the state-point models, including calculated spent fuel isotopic compositions and reactor-core-geometry modeling using either MCNP 4B (Ref. 17) or KENO V.a, for the Crystal River Unit 3, Sequoyah Unit 2, Surry Unit 1, TMI Unit 1, and North Anna Unit 1 CRC state-points are available in Refs. 18 through 22, respectively. The CRC state-points consist of 37 HZP reactor start-up critical conditions and 3 hot full-power (HFP) critical configurations and include fresh (i.e., beginning of cycle 1), beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) reactor-core configurations. State-point characteristics, including cycle length to state-point in effective full-power days (EFPDs), core average burnup, initial fuel enrichments, soluble boron concentrations, and moderator densities, are presented in Table I. Moderator temperature is ~ 550 K, and fuel temperatures for the HZP and HFP state-points are ~ 550 and 900 K, respectively. Cycle downtime is < 1 yr for all CRC state-points except for the TMI Unit 1 BOC-5 restart, which occurred after a downtime of 6.63 yr, and the Sequoyah Unit 2 MOC-3 restart, which occurred after a downtime of 2.73 yr. The KENO V.a models contain fine core details that describe axial (Crystal River state-points only) and radial fuel composition variations, reactivity control devices, and homogenized assembly hardware regions. A horizontal and a partial vertical cross section of the KENO V.a geometry for a Crystal River state-point is illustrated in Fig. 2. The fuel pin geometry for Crystal River CRC models includes

TABLE I
Summary of CRC State-Point Characteristics and Calculated k_{eff} and c_k Values

Case Identifier	Reactor Cycle	EFPDs	Core Average Burnup	Initial Enrichment (wt% ^{235}U)	Soluble Boron (ppm)	Moderator Density (g/cm^3)	EALF (eV)	KENO V.a ^a k_{eff}	c_k ^b
CR3SP1	1A	0	0	1.93 to 2.83	1403	0.7682	5.58E-01 ^c	0.9938	0.6726
CR3SP2	1B	268.8	8.09	1.93 to 2.83	1106	0.7686	6.28E-01	0.9925	0.9442
CR3SP3	1B	411	12.34	1.93 to 2.83	493	0.7653	6.09E-01	0.9946	0.9593
CR3SP4	2	0	8.67	2.54 to 2.83	1294	0.7682	6.28E-01	0.9920	0.9374
CR3SP5	3	0	7.50	2.54 to 2.83	1428	0.7682	6.58E-01	0.9924	0.9211
CR3SP6	3	168.5	12.54	2.54 to 2.83	737	0.7653	6.46E-01	0.9919	0.9552
CR3SP7	3	250	14.98	2.54 to 2.83	562	0.7633	6.43E-01	0.9893	0.9594
CR3SP8	4	0	6.92	2.62 to 2.95	1384	0.7682	6.62E-01	0.9915	0.8867
CR3SP9	4	228.1	14.00	2.62 to 2.95	705	0.7682	6.35E-01	0.9890	0.9554
CR3SP10	4	253	14.77	2.62 to 2.95	502	0.7633	6.64E-01	0.9956	0.9554
CR3SP11	5	0	7.08	2.62 to 3.29	1540	0.7682	7.24E-01	0.9952	0.8649
CR3SP12	5	388.5	19.12	2.62 to 3.29	605	0.7633	7.08E-01	0.9960	0.9577
CR3SP13	6	0	12.01	2.62 to 3.49	1574	0.7682	7.88E-01	0.9940	0.8916
CR3SP14	6	96	14.99	2.62 to 3.49	1211	0.7662	7.80E-01	0.9948	0.9335
CR3SP15	6	400	24.41	2.62 to 3.49	390	0.7653	7.36E-01	0.9899	0.9602
CR3SP16	7	0	10.02	2.54 to 3.84	2033	0.7682	8.74E-01	0.9923	0.8545
CR3SP17	7	260.3	18.09	2.54 to 3.84	1223	0.7682	8.40E-01	0.9902	0.9473
CR3SP18	7	291	19.04	2.54 to 3.84	1149	0.7643	8.46E-01	0.9906	0.9501
CR3SP19	7	319	19.91	2.54 to 3.84	1048	0.7682	8.27E-01	0.9895	0.9537
CR3SP20	7	462.3	24.35	2.54 to 3.84	563	0.7691	7.78E-01	0.9912	0.9592
CR3SP21	7	479	24.87	2.54 to 3.84	520	0.7691	7.83E-01	0.9899	0.9604
CR3SP22	8	0	12.26	1.93 to 3.94	2101	0.7682	9.39E-01	0.9905	0.8584
CR3SP23	8	97.6	15.27	1.93 to 3.94	1751	0.7653	9.42E-01	0.9903	0.9120
CR3SP24	8	139.8	16.58	1.93 to 3.94	1612	0.7653	9.34E-01	0.9916	0.9240
CR3SP25	8	404	24.74	1.93 to 3.94	865	0.7643	8.52E-01	0.9910	0.9539
CR3SP26	8	409.6	24.91	1.93 to 3.94	865	0.7643	8.37E-01	0.9910	0.9536
CR3SP27	8	515.5	28.19	1.93 to 3.94	675	0.7643	8.20E-01	0.9873	0.9558
CR3SP28	9	0	14.18	1.93 to 3.94	2212	0.7682	9.60E-01	0.9895	0.8581
CR3SP29	9	158.8	19.10	1.93 to 3.94	1572	0.7683	9.32E-01	0.9919	0.9275
CR3SP30	9	219	20.96	1.93 to 3.94	1481	0.7653	9.23E-01	0.9897	0.9401
CR3SP31	9	363.1	25.42	1.93 to 3.94	963	0.7614	8.81E-01	0.9861	0.9513
CR3SP32	10	0	15.24	3.84 to 4.167	2326	0.7682	1.04E+00	0.9811	0.8493
CR3SP33	10	573.7	33.00	3.84 to 4.167	516	0.7643	8.60E-01	0.9807	0.9620
Sequoyah Unit 2									
SQ2C3BZ	3	0	11.00	2.6 to 3.8	1685	0.7540	8.65E-01	1.0054	0.9010
SQ2C3BF ^d	3	0	11.15	2.6 to 3.8	1150	0.7149	9.58E-01	1.0076	0.8881
SQ2C3M ^d	3	205	19.25	2.6 to 3.8	475	0.7149	9.47E-01	1.0054	0.9391
Surry Unit 1									
SU1C2B	2	0	6.93	1.9 to 3.3	1030	0.7540	6.19E-01	1.0035	0.9016
SU1C2E ^d	2	204	13.85	1.9 to 3.3	123	0.7327	6.72E-01	1.0107	0.9498
TMI Unit 1									
TMI1C5B	5	0	11.44	2.6 to 2.9	1182	0.7540	7.28E-01	1.0017	0.9324
North Anna Unit 1									
NA1C5B	5	0	11.07	3.2 to 3.6	1836	0.7540	9.17E-01	1.0041	0.8493

^aStandard deviation due to statistical uncertainty for the k_{eff} values varies between 0.0002 and 0.0005.

^bValues for c_k obtained from comparison with the GBC-32 cask containing typical PWR spent fuel assemblies of 3.777 wt% ^{235}U initial enrichment, 40 GWd/t U burnup, and a 5-yr cooling time.

^cRead as 5.58×10^{-1} .

^dHFP state-points with fuel temperature of ~ 900 K.

an 18-zone axial profile with each irradiated fuel zone composition containing up to 85 isotopes, including oxygen, whereas the pin geometry for the other CRC models consists of a single mixture, which is based on average

assembly burnup and contains up to 47 isotopes, including oxygen, in the irradiated fuel composition. Absorber materials in the reactivity control devices were B_4C - Al_2O_3 for burnable poison rod assemblies, Ag-In-Cd alloy

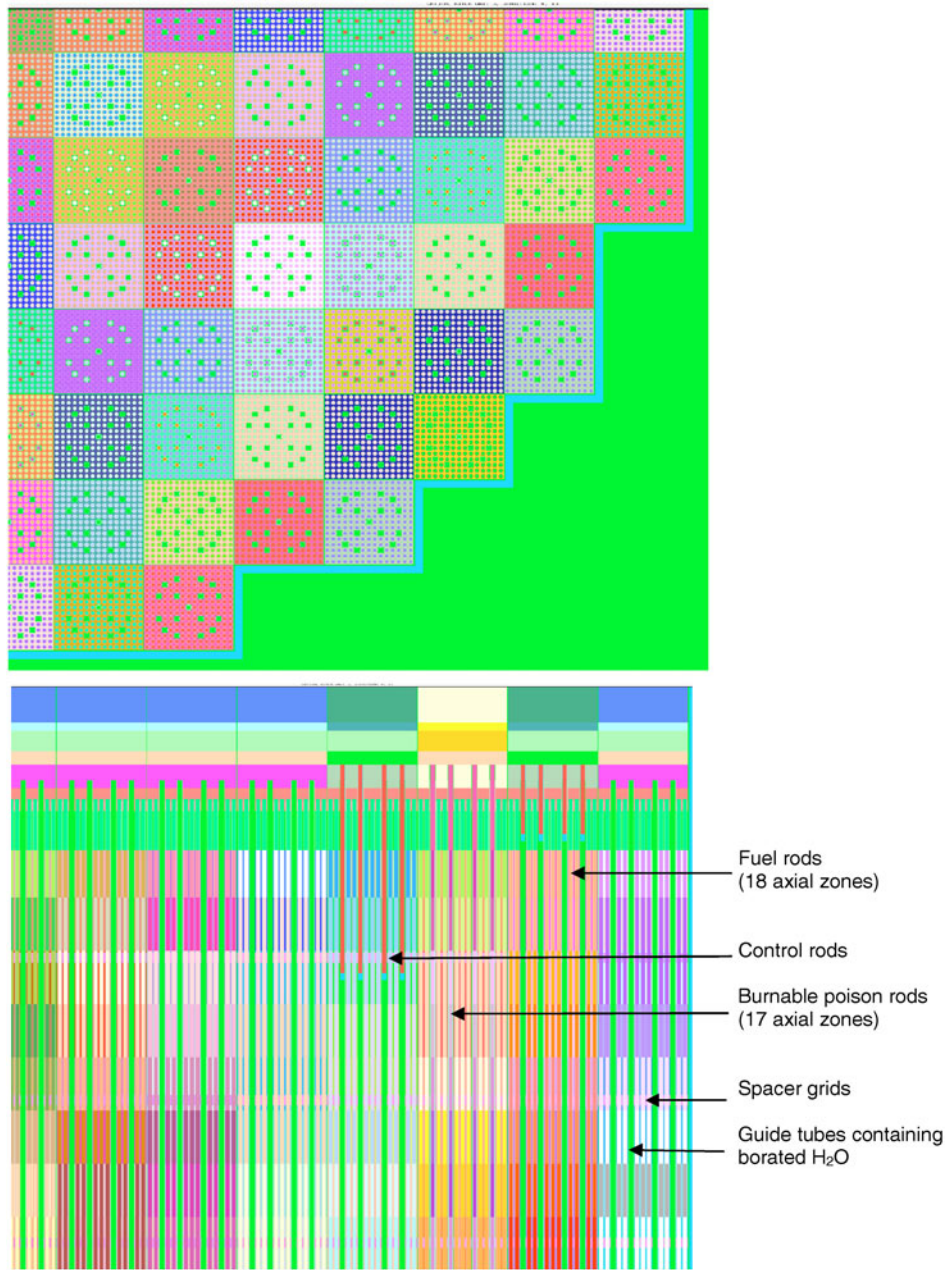


Fig. 2. Horizontal and partial vertical cross sections of KENO V.a geometry for a Crystal River Unit 3 state-point.

for rod cluster control assemblies (RCCAs), and Ag-In-Cd alloy or stainless steel 304 for axial power-shaping-rod assemblies (APSRAs).

The k_{eff} results and EALF values obtained with KENO V.a and the SCALE 238-group ENDF/B-V (238GROUP-NDF5) library are presented in Table I. Generally, the KENO V.a k_{eff} results are in good agreement with the k_{eff} results documented in Refs. 18 through 22 (within $\pm 0.3\%$), indicating that the mixture compositions and geometry models are consistent between the current and previously published calculations. Only the results for the last two

Crystal River state-points are significantly lower than the comparison MCNP calculation results. The largest difference (-1.06%) is observed in the case of Crystal River state-point 32, which has the highest soluble boron concentration (2326 ppm). The impact on sensitivity coefficients is expected to be less important because the sensitivity coefficients are defined in terms of relative change in k_{eff} values to relative change in cross-section values. Further exploration of the k_{eff} deviation exhibited by these two state-point models would be appropriate before their use in bias determination analysis.

Because the two systems contain spent nuclear fuel assemblies placed in lattice configurations with light water as the neutron moderator, the GBC-32 cask and a CRC state-point are physically similar. However, differences in isotopic compositions and neutron spectra may exist as a result of differences in fuel burnup, temperature, and cooling time; moderator density and temperature; and boron concentration and form (boron plates in the cask model versus soluble boron in the CRC state-points). As indicated by the EALF values presented in Table I, the CRC state-points have harder neutron spectra than the GBC-32 cask. Additionally, the CRC state-points contain a number of materials (e.g., Ag-In-Cd in RCCAs and APSRAs) that are not present in the GBC-32 cask. Furthermore, soluble boron, burnable poison rods, and control rods used to balance the excess reactivity of the reactor core decrease the thermal neutron population within the fuel lattice cell, whereas the Boral panels placed between fuel assemblies in the GBC-32 cask decrease the thermal neutron population outside of the fuel lattice. For state-points at or near reactor EOC, however, the amount of neutron absorbers is significantly reduced since control and power-shaping rods are almost entirely withdrawn, the boron in burnable absorber rods is depleted as a result of neutron capture, and the soluble boron concentration is low to maintain a critical state.

To develop a better understanding of the significance of these various issues, the neutronic similarities or dissimilarities between the GBC-32 cask and the CRC state-points were rigorously examined using integral and the nuclide-reaction-specific indices that are available with the SCALE code system for cross-section S/U analyses, as described in Sec. III.

III. SENSITIVITY AND UNCERTAINTY ANALYSES

Previous work has formulated the theoretical bases for use of cross-section S/U analysis methods in criticality safety validation²³ and developed the software tools needed to implement the methodology in the SCALE code system.^{8–10} The technique described in Ref. 23 is similar to that employed in previous studies involving the validation of data for use in the design of fast reactors.²⁴ It should be noted that the S/U methodology and associated computational tools and nuclear data undergo continual improvement based on acquired experience with various applications and on improvements in cross-section covariance data. The analyses presented in this paper are based on results obtained with the cross-section S/U analysis tools in SCALE 5.1, including TSUNAMI-3D and TSUNAMI-IP. The TSUNAMI-3D calculations performed for the analyses invoked functional modules BONAMIST, NITAWLST, KENO V.a, and SAMS to determine the sensitivity of

the calculated value of k_{eff} to applicable ENDF/B-V nuclear data as a function of nuclide, reaction type, and energy in the SCALE 238-group structure. For selected cases, direct perturbation calculations were performed to check the quality of the sensitivity coefficients calculated by TSUNAMI-3D, as described in Ref. 12.

TSUNAMI-IP used the sensitivity values generated by TSUNAMI-3D, which are stored in sensitivity data files, and the SCALE ENDF/B-V-recommended covariance library²⁵ (44GROUPV5REC) to compute a set of integral and nuclide-reaction-specific indices that are indicative of similarity of nuclear systems. The S/U analyses used total reaction sensitivity coefficients and energy-dependent sensitivity profiles,^{9,23} the integral index c_k (Refs. 10 and 23), and the nuclide-reaction-specific integral index g (Refs. 10 and 26) to assess similarities between the representative spent fuel cask and the CRC state-points. Further, the uncertainty in k_{eff} due to dissimilar sensitivity data¹⁰ was evaluated (see Sec. III.D).

A sensitivity profile, $s(E)$, is the energy-dependent ratio of the relative change in k_{eff} , caused by perturbations in the cross section of a nuclide-reaction pair, to the relative change in the cross section {i.e., $s(E) = [\Delta k_{eff}/k_{eff}]/[\Delta\sigma(E)/\sigma(E)]$, where $\sigma(E)$ is the energy-dependent microscopic cross section of a nuclide-reaction pair}. Sensitivity coefficients are the integral of the energy-dependent sensitivity profiles over all energy groups. Total reaction energy-dependent sensitivity profiles and sensitivity coefficients were used in the analyses presented in this paper. The methodologies and mathematical formulations used in the TSUNAMI sequence to determine these parameters are described in Refs. 23 and 27.

The nuclide-reaction-specific integral index g is useful in assessing the ability of an experiment to provide sensitivities that are equal to or greater than the corresponding sensitivities of an application for nuclides and reactions in all neutron energy groups. Cross-section uncertainties from regions of the spectrum where the sensitivity profile of the application is not completely covered by the sensitivity profiles of benchmark experiments will cause an underestimation of the calculational bias and an inadequate determination of the criticality safety margin. The nuclide-reaction-specific integral index g is defined in terms of the normalized differences of the groupwise sensitivity coefficients for a particular nuclide n and reaction x , summed over all energy groups j , as shown in Eq. (1), where superscripts a and e are used to denote the sensitivities of the application and the experiment, respectively:

$$g_x^n = 1 - \frac{\sum_j (S_{x,j}^{a,n} - S_{x,j}^{e,n})}{\sum_j S_{x,j}^{a,n}},$$

where

$$S_{x,j}^{e',n} = \begin{cases} S_{x,j}^{e,n} & , \text{ where } |S_{x,j}^{a,n}| \geq |S_{x,j}^{e,n}| \text{ and } \frac{S_{x,j}^{a,n}}{|S_{x,j}^{a,n}|} = \frac{S_{x,j}^{e,n}}{|S_{x,j}^{e,n}|} \\ S_{x,j}^{a,n} & , \text{ where } |S_{x,j}^{a,n}| < |S_{x,j}^{e,n}| \text{ and } \frac{S_{x,j}^{a,n}}{|S_{x,j}^{a,n}|} = \frac{S_{x,j}^{e,n}}{|S_{x,j}^{e,n}|} \\ 0 & , \text{ otherwise .} \end{cases} \quad (1)$$

Hence, the value of the nuclide-reaction-specific integral index g represents the fraction of the area under the energy-dependent nuclide-reaction sensitivity profile for the application that is covered by the same profile for an experiment. A g value of 1.0 for a nuclide reaction means that the sensitivity of the experiment equals or exceeds the sensitivity of the application in all neutron energy groups. Further discussion of the concept of coverage of an application by an experiment is provided in Sec. III.C.

The integral index c_k provides a measure of similarity of two systems in terms of their shared uncertainty in k_{eff} due to cross-section uncertainties. Generally, the integral index c_k is used to assess neutronic similarity between systems as a whole, because a c_k value is based on uncertainty correlations from all nuclides in the systems. A c_k value represents the correlation coefficient between sensitivity-weighted cross-section uncertainties in the application system and an experiment. Relative uncertainties in cross-section data can be represented as the elements of an $M \times M$ matrix, $C_{\alpha,\alpha} \equiv [\text{cov}(\alpha_n, \alpha_m) / \alpha_n \alpha_m]$, where M is the product of the number of unique nuclide-reaction pairs and the number of energy groups, $\alpha \equiv (\alpha_n)$ is an M -dimensional vector containing nuclear data parameters (i.e., groupwise nuclide-reaction specific cross sections), and $\text{cov}(\alpha_n, \alpha_m)$ is an element of the $M \times M$ absolute covariance matrix. The $C_{\alpha,\alpha}$ matrix contains relative variance data along the diagonal and relative covariance data in the off-diagonal positions that provide a measure of how strong the correlation of uncertainty is between two nuclide-reaction-energy group triplets. A rigorous definition of cross-section covariance data and a description of the process used to convert energy-dependent covariance information in the ENDF libraries to the SCALE 44-group format are given in Ref. 28. In a TSUNAMI-IP calculation, the covariance data are propagated to relative changes in the calculated k_{eff} value of a given system via the sensitivity coefficients. An uncertainty matrix is computed for the system k_{eff} values, $C_{kk} = S_k C_{\alpha,\alpha} S_k^\dagger$, where S_k and S_k^\dagger are the matrix and the transpose matrix, respectively, containing sensitivities of the calculated k_{eff} values of critical systems to the α parameters. The S_k is an $I \times M$ matrix defined as $S_k \equiv [(\alpha_m / k_i) (\partial k_i / \partial \alpha_m)]$, where $m = 1, 2, \dots, M$; $i = 1, 2, \dots, I$; and I is the total number of experiment and application critical systems being considered. The diagonal elements of

the C_{kk} matrix consist of sensitivity-weighted relative variance values, η_i^2 , for each of the systems under consideration, and the off-diagonal elements consist of sensitivity-weighted covariance values, η_{ij}^2 , between two systems. The integral index c_k is defined as shown in Eq. (2) such that the single c_k value represents the correlation coefficient between sensitivity-weighted uncertainties in system i and system j :

$$c_k = \frac{\eta_{ij}^2}{(\eta_i \eta_j)} \quad (2)$$

The interpretation of the integral index c_k is as follows: A value of 1.0 represents full correlation between the systems, a value of 0.0 represents no correlation between the systems, and a value of -1.0 represents a full anticorrelation. Hence, the integral c_k value for the GBC-32 cask compared with itself is 1.0, and the integral c_k value for a CRC state-point model depends on its neutronic similarity to a given application system, which in this case is the GBC-32 cask model. Reference 23 demonstrates that c_k values of 0.80 or higher constitute systems that are sufficiently similar to be useful in the determination of bias and associated bias uncertainty. Further, Ref. 23 recommends that the validation methodology should include about 15 to 20 similar systems ($c_k \geq 0.90$) or 25 to 40 marginally similar systems ($0.90 > c_k \geq 0.80$).

III.A. Integral Index c_k

Integral c_k values based on ENDF/B-V covariance data and sensitivity data for the typical spent fuel analyzed in this paper (3.777 wt% ^{235}U initial enrichment, 40 GWd/t U burnup, and a 5-yr cooling time) are presented in Table I. [Refer to Eq. (2) for the definition of the integral index c_k .] A grouping of the CRC state-points based on the degree of similarity, average core burnup, soluble boron concentrations, and EALF is presented in Table II. For this fuel assembly, the integral c_k values vary from 0.6726 to 0.9620, with the lowest value corresponding to Crystal River Unit 3 fresh fuel (CR3SP1) and the highest value corresponding to Crystal River Unit 3 Cycle 10 EOC state-point (CR3SP33). Thirty CRC state-points, or 75% of the total CRC state-points evaluated in this study, have integral c_k values > 0.90 . Among these, 16 Crystal River Unit 3 state-points have c_k values > 0.95 . These state-points, which are highly similar to the representative spent fuel cask, were attained at or near the end of a reactor cycle and have a large range of values for reactor core burnup (12.34 to 33.00 GWd/t U) and soluble boron concentration (390 to 1149 ppm). Except for the BOC core configurations that had fresh fuel assemblies placed at core periphery, such as the Crystal River Unit 3 Cycles 2 (CR3SP4) and 3 (CR3SP5) and the TMI Unit 1 Cycle 5 (TMI1C5B) BOC core configurations, the BOC CRC state-points are only marginally

TABLE II
CRC State-Point Grouping Based on the Degree of Similarity to GBC-32 Cask*

Integral Index c_k	Number of CRC State-Points	CRC State-Points ^a	Core Average Burnup (GWd/t U)	Soluble B Concentration (ppm)	EALF (eV)
0.95 to 0.9620	16	Crystal River Unit 3 state-points 3, 6, 7, 9, 10, 12, 15, 18, 19, 20, 21, 25, 26, 27, 31, and 33	12.34 to 33.00	390 to 1149	0.61 to 0.88
0.90 to 0.95	14	Crystal River Unit 3 state-points 2, 4, 5, 14, 17, 23, 24, 29, and 30; Sequoyah Unit 2 state-points 1 and 3; Surry Unit 1 state-points 1 and 2; TMI Unit 1 state-point 1	6.93 to 20.96	123 to 1751	0.62 to 0.95
0.85 to 0.90	7	Crystal River Unit 3 state-points 8, 11, 13, 16, 22 and 28; Sequoyah Unit 2 state-point 2	6.92 to 12.26	1150 to 2212	0.66 to 0.96
0.80 to 0.85	2	Crystal River Unit 3 state-point 32; North Anna Unit 1 state-point 1	15.24, 11.07	2326, 1836	1.04, 0.92
0.6726	1	Crystal River Unit 3 state-point 1	0	1403	0.56

*The GBC-32 cask contains typical PWR spent fuel assemblies of 3.777 wt% ^{235}U initial enrichment, 40 GWd/t U burnup, and a 5-yr cooling time.

^aRefer to Table I for CRC state-point characteristics.

similar to the GBC-32 cask, due to their fresh fuel contents and lower average burnups. Additional results of the S/U analyses (Ref. 12, Table 4.2) show that except for the Crystal River Unit 3 fresh core, all evaluated CRC state-points are either highly similar ($c_k \geq 0.95$), similar, or marginally similar (i.e., $c_k \geq 0.8$) to the GBC-32 cask containing typical spent fuel of burnup values ranging from 20 to 60 GWd/t U. Approximately 30 of the 40 CRC state-points evaluated have integral c_k values >0.90 for spent fuel with burnup in this range. The comparison of the GBC-32 cask containing spent fuel of 10 GWd/t U burnup and the evaluated CRC state-points resulted in a c_k value of 0.8091 for the Crystal River Unit 3 fresh core and c_k values >0.90 for the other 39 CRC state-points.

Figure 3 shows integral index c_k as a function of burnup, where the c_k values were obtained from the comparison of the typical spent fuel analyzed in this paper (3.777 wt% ^{235}U initial enrichment, 40 GWd/t U burnup, and a 5-yr cooling time) with Crystal River Unit 3 state-points with similar soluble boron concentrations. In Fig. 3, the first (~ 500 -ppm), second (~ 1000 -ppm), and fourth (~ 2000 -ppm) data series from the top show a slight linear variation with burnup for state-points at or near EOC, MOC, and BOC, respectively, whereas the third data se-

ries (~ 1500 ppm) shows a steeper linear variation for BOC and MOC state-points with similar soluble boron concentrations. Similar trends in integral index c_k representations as a function of burnup were observed when the CRC state-points were compared with the GBC-32 cask containing spent fuel of various burnups (Ref. 12, Fig. 4.3). These trends demonstrate that CRC state-points from various reactor cycles characterized by similar fuel reactivity have similar integral index c_k values and that the integral index c_k increases with increasing CRC state-point fuel burnup.

In comparison with the CRC state-point models, publicly available uranium and mixed-uranium/plutonium benchmark critical experiment configurations are, at most, marginally similar to the GBC-32 cask model (i.e., less applicable to validation of burnup-credit criticality calculations). Figure 4 illustrates integral c_k values for various critical experiment sets, including the CRC state-points and mixed-uranium/plutonium and low-enriched-uranium (LEU) benchmark laboratory critical experiments (LCEs) available in the *International Handbook of Evaluated Criticality Safety Benchmark Experiments*²⁹ from the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development. The selected

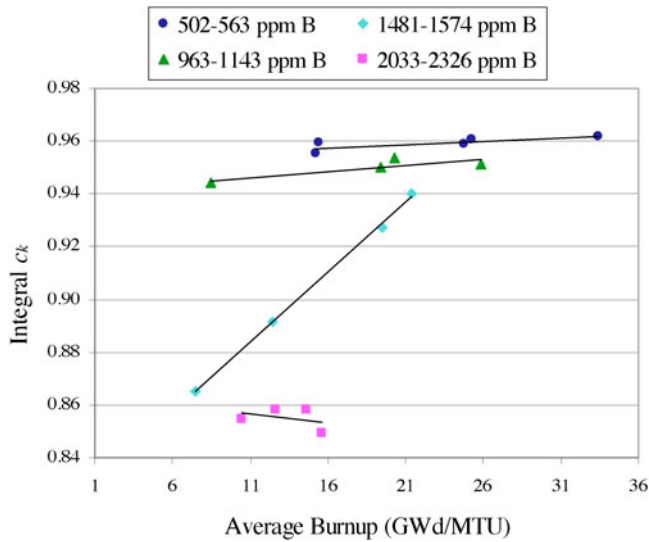


Fig. 3. Integral index c_k as a function of burnup: CRC state-points with similar soluble boron concentrations.

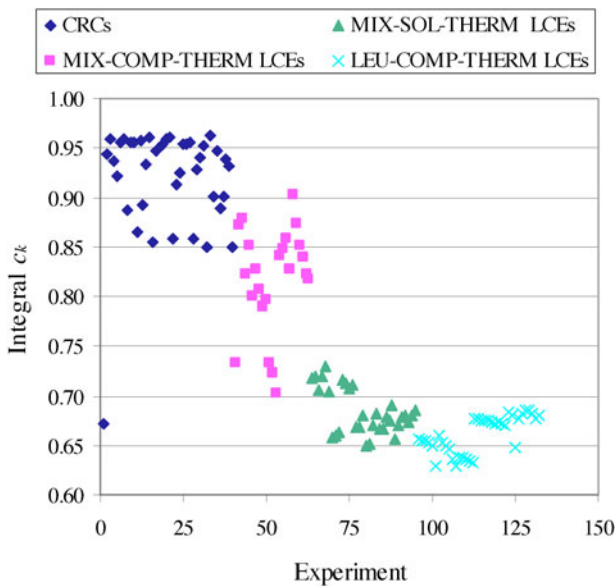


Fig. 4. Integral c_k values for various critical experiment sets.

mixed-uranium/plutonium critical experiments are water-moderated thermal systems containing either mixed-oxide (MOX) fuel rods in a lattice configuration (cases from experiments MIX-COMP-THERM-002 through MIX-COMP-THERM-008) or mixed-plutonium/uranium nitrate solution (cases from experiments MIX-SOL-THERM-001, -002, -004, and -005). The mixed-uranium/plutonium systems contain the isotopes ^{234}U , ^{235}U , ^{236}U , ^{238}U (natural uranium in the case of MOX fuel rods), and ^{238}Pu through ^{242}Pu . The selected LEU critical experi-

ments are water-moderated thermal systems containing LEU in a lattice configuration (cases from experiments LEU-COMP-THERM-010, -017, -026, and -042), in which the LEU compound contains the isotopes ^{234}U , ^{235}U , ^{236}U , and ^{238}U . The graph in Fig. 4 shows that the CRC state-points have better neutronic similarity to the GBC-32 cask than other publicly available LCEs. The CRC state-points have higher neutronic similarity than the mixed-uranium/plutonium LCEs because the uranium/plutonium systems include the uranium and plutonium isotopes in compositions significantly different from those of spent nuclear fuel.

III.B. Sensitivity Coefficients and Energy-Dependent Sensitivity Profiles

The S/U analyses based on integral index c_k values showed that approximately 30 of the 40 CRC state-points considered in the analysis are similar ($c_k > 0.90$) to the GBC-32 cask loaded with representative spent fuel (3.777 wt% ^{235}U initial enrichment, 40 GWd/t U burnup, and a 5-yr cooling time) and hence meet the neutronic similarity criteria for being useful in bias determination. Only one of the CRC state-points (the Crystal River Unit 3 state-point with all fresh fuel) had a c_k value below the suggested threshold for being marginally similar (e.g., 0.80). The c_k values are dominated by neutron reactions with major actinides, hydrogen, and boron because the sensitivities of these reactions are much higher than those of the minor actinides and fission products. Figure 5 shows the most significant k_{eff} sensitivity coefficients for the GBC-32 cask and for Crystal River Unit 3 state-points obtained at the beginning, middle, and end of Cycle 8 (CR3SP22, CR3SP24, and CR3SP27, which are identified in the figure as BOC, MOC, and EOC CRC, respectively). The high-reactivity-worth fission-product nuclides include ^{149}Sm , ^{143}Nd , ^{103}Rh , ^{151}Sm , ^{133}Cs , and ^{155}Gd . These nuclides are stable except for ^{151}Sm , which has a half-life of 90 yr. Their concentrations in spent fuel compositions increase after fuel discharge because the main precursors ^{149}Pm , ^{143}Pr , ^{103}Ru , ^{151}Pm , ^{133}Xe , and ^{155}Eu decay by negative beta emission with half-lives of 2.212 days, 13.57 days, 39.27 days, 1.183 days, 5.243 days, and 4.75 yr, respectively.³⁰ The decay process affects mainly ^{155}Gd and, to a lesser extent, ^{149}Sm concentrations. These two fission-product nuclides have very large thermal neutron absorption cross sections and exist in smaller concentrations during fuel irradiation. However, ^{155}Gd and ^{149}Sm concentrations in spent fuel with a 5-yr cooling time are approximately 100 times and 2 times, respectively, those of discharged-spent-fuel compositions.

Total and energy-dependent k_{eff} sensitivities for ^{149}Sm , ^{143}Nd , ^{103}Rh , ^{151}Sm , and ^{133}Cs total cross sections are shown in Figs. 6 through 10 for the GBC-32 cask and the three Crystal River Unit 3 state-points obtained at the beginning, middle, and end of Cycle 8 (CR3SP22, CR3SP24, and CR3SP27, respectively). Only

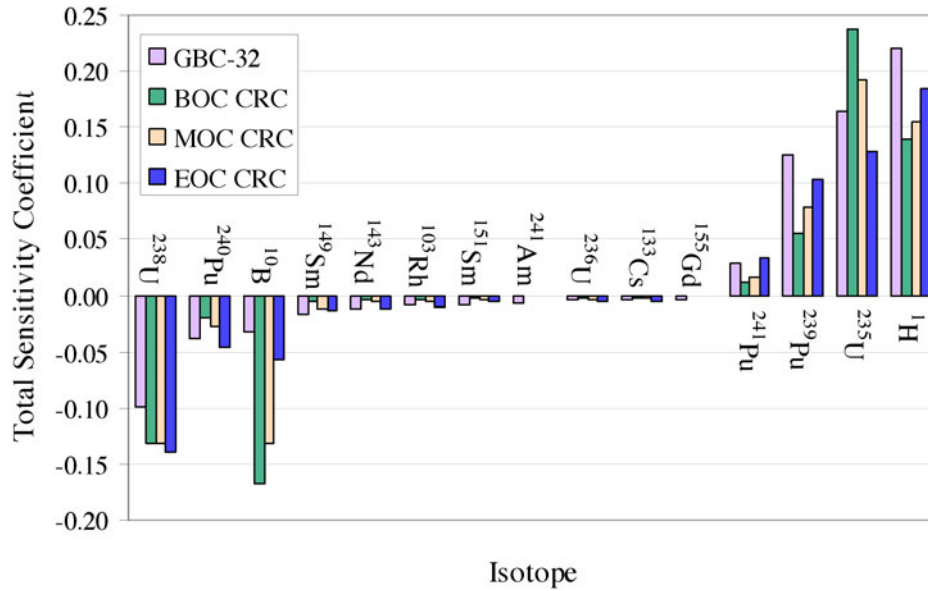


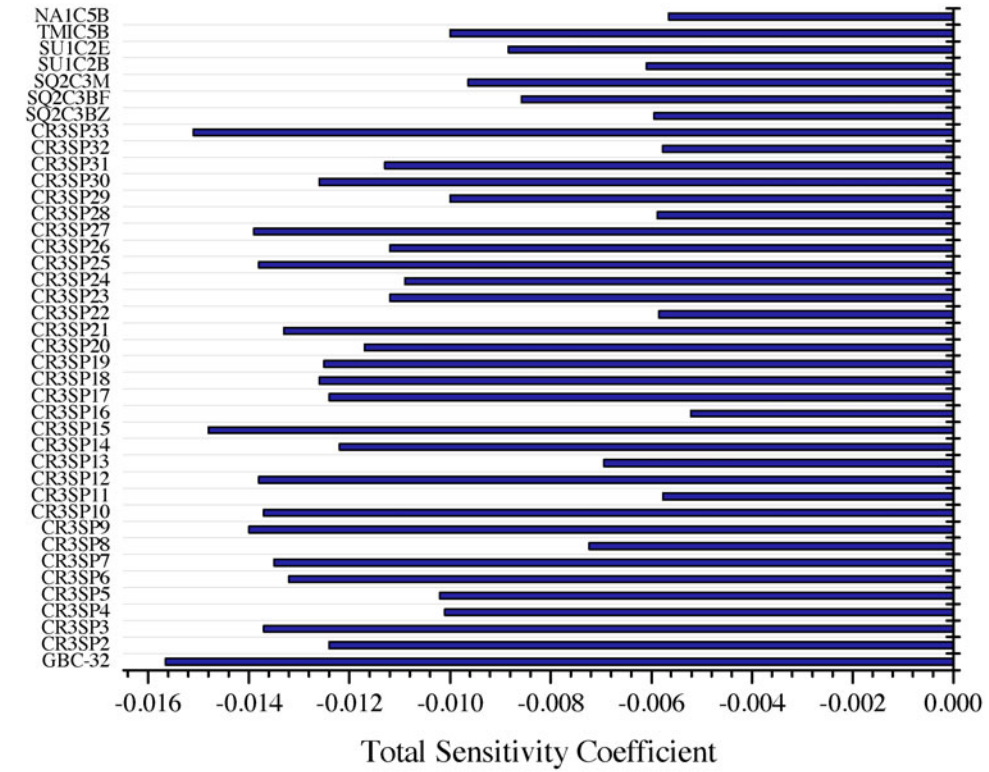
Fig. 5. Comparison of k_{eff} total sensitivity coefficients for GBC-32 (40 GWd/t U), CR3SP27 (EOC), CR3SP24 (MOC), and CR3SP22 (BOC). (Refer to Table I for CRC state-point characteristics.)

the configurations of TMI Unit 1 Cycle 5 BOC (6.63-yr cycle downtime) and Sequoyah Unit 2 Cycle 3 MOC (2.73-yr cycle downtime) are similar to the GBC-32 cask in regard to ¹⁵⁵Gd contents because only these systems contain comparable ¹⁵⁵Gd concentrations (as a result of ¹⁵⁵Eu decay after reactor shutdown). This is illustrated in Fig. 11. Sensitivity profiles for actinides and other fission-product nuclides relevant to burnup credit are included in Appendix B of Ref. 12. Comparison of the sensitivity profiles of the GBC-32 cask and the CRC state-points indicates that for the fission products and actinides relevant to burnup credit, the energy-dependent sensitivities are quite similar between the cask model and many of the CRC state-points. With the exception of a few nuclides that significantly build in after discharge (e.g., ¹⁵⁵Gd and ²⁴¹Am), the total sensitivities are quite comparable between the GBC-32 cask and many of the CRC state-points. This is an important finding that addresses long-standing questions related to the influence of the various nuclides on the k_{eff} values of these systems. CRC state-points with higher average burnup tend to have higher total sensitivity coefficients, and the state-points at or near the end of reactor cycle provide better similarity for the same burnup than do the other state-points. A small displacement or offset between the CRC sensitivity profiles and the GBC-32 sensitivity profile in the thermal region is observed for nuclides with total cross sections inversely proportional to neutron velocity, such as ¹⁴³Nd (see Fig. 7). Similar trends are seen in Fig. 6 and in Figs. 8 through 11. The sensitivity profile displacement can be explained by the differences in the thermal spectra of the two types of systems (i.e., the difference in tem-

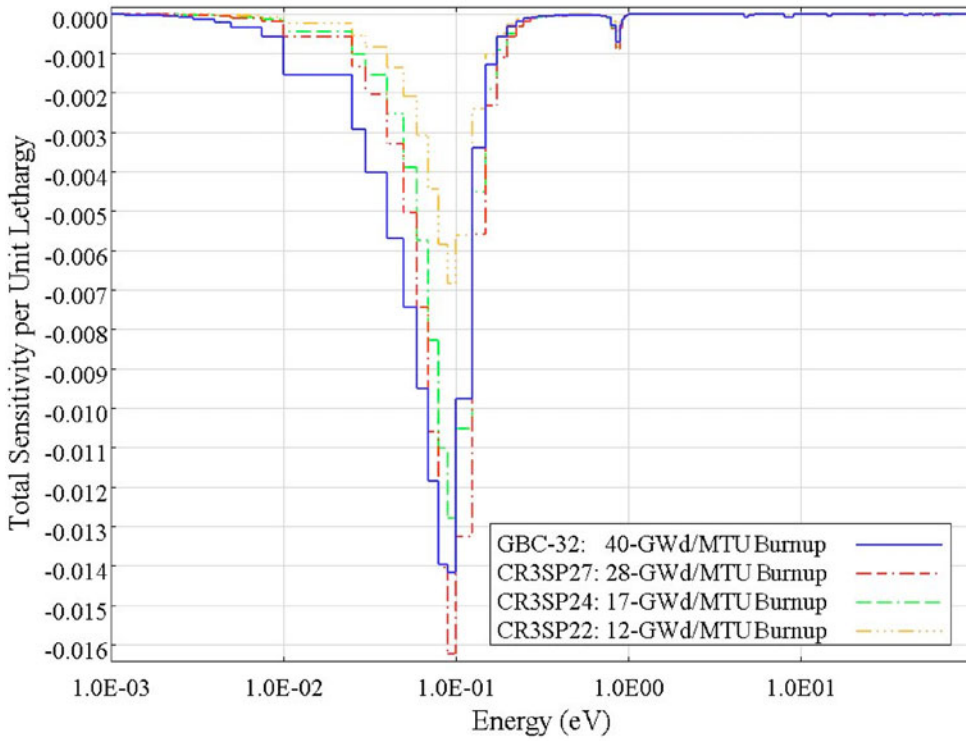
perature between the reactor critical configurations and the cask).

III.C. Nuclide-Reaction-Specific Integral Index g

As suggested by the sensitivity profiles illustrated in Figs. 6 through 11, the use of total sensitivity coefficients alone in assessing neutronic similarity is not sufficient because systems with similar total sensitivity coefficients may have very different energy-dependent sensitivity profiles for some nuclides. Therefore, the concept of coverage of an application by an experiment was developed and can be used to assess system similarities on an energy-dependent, nuclide-reaction-specific level. Figure 12 illustrates this concept by comparing energy-dependent k_{eff} sensitivity profiles to the ²³⁹Pu nubar data for an application and an experiment. In Fig. 12, the sensitivity profile of the application comprises areas 1 and 2, whereas the sensitivity profile of the experiment comprises areas 1 and 3. Area 1 of the sensitivity profile for the application is completely covered by the sensitivity profile of the experiment and is used in determining the value of integral index g . Nuclide-reaction integral index g , representing the fraction of the application sensitivity profile area that also pertains to the area delimited by the sensitivity profile of the experiment, provides a measure of the coverage.^{10,26} [Refer to Eq. (1) for the definition of the nuclide-reaction-specific integral index g .] A g value of 1.0 indicates complete coverage of the application by the experiment for the particular nuclide-reaction pair, while a g value of 0.0 indicates no coverage of the application by the experiment for the particular

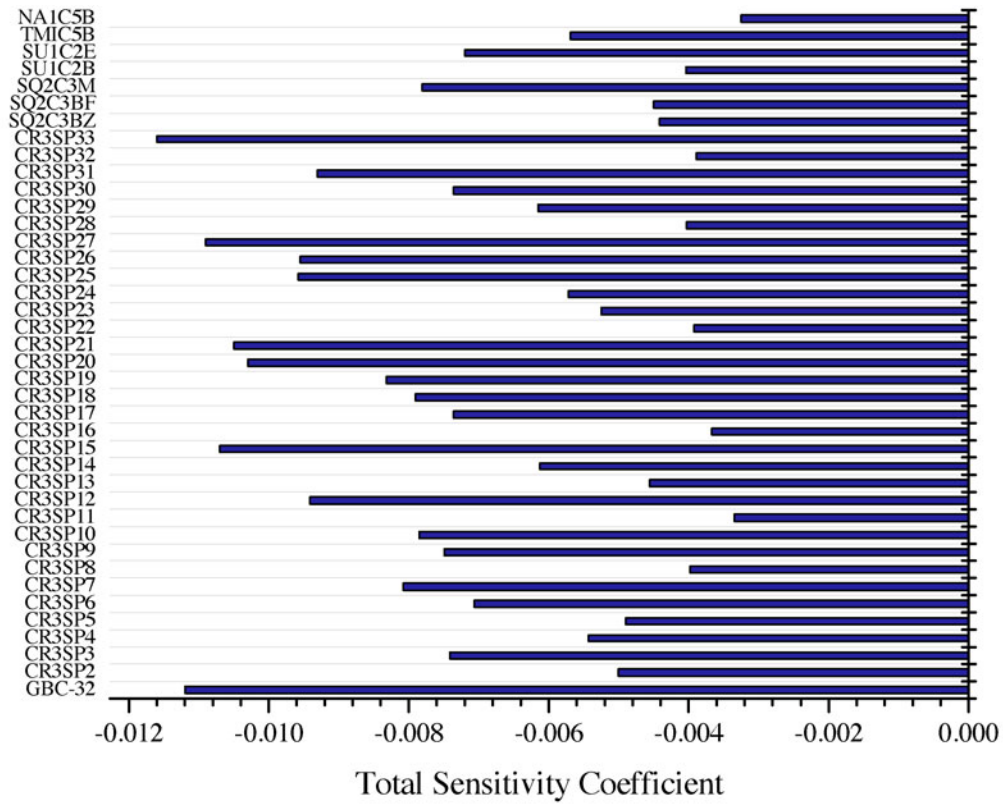


(a)

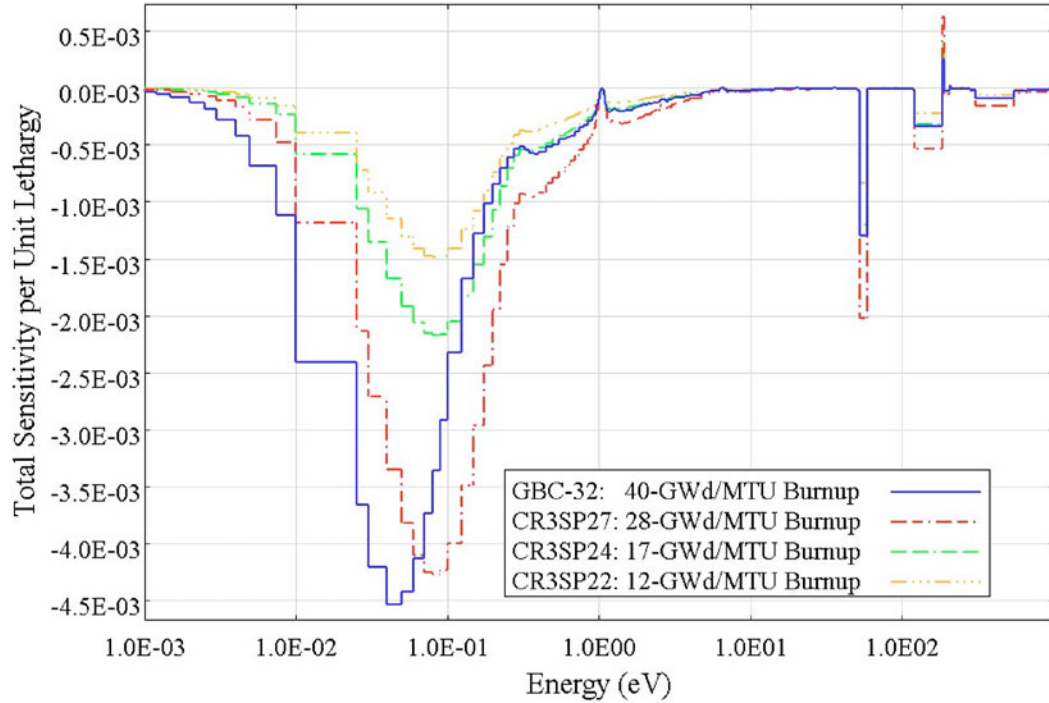


(b)

Fig. 6. (a) Energy- and region-integrated k_{eff} sensitivities to ^{149}Sm total cross section. (b) Comparison of ^{149}Sm sensitivity profiles for GBC-32 (40 GWd/t U), CR3SP27 (EOC), CR3SP24 (MOC), and CR3SP22 (BOC). (Refer to Table I for CRC state-point characteristics.)

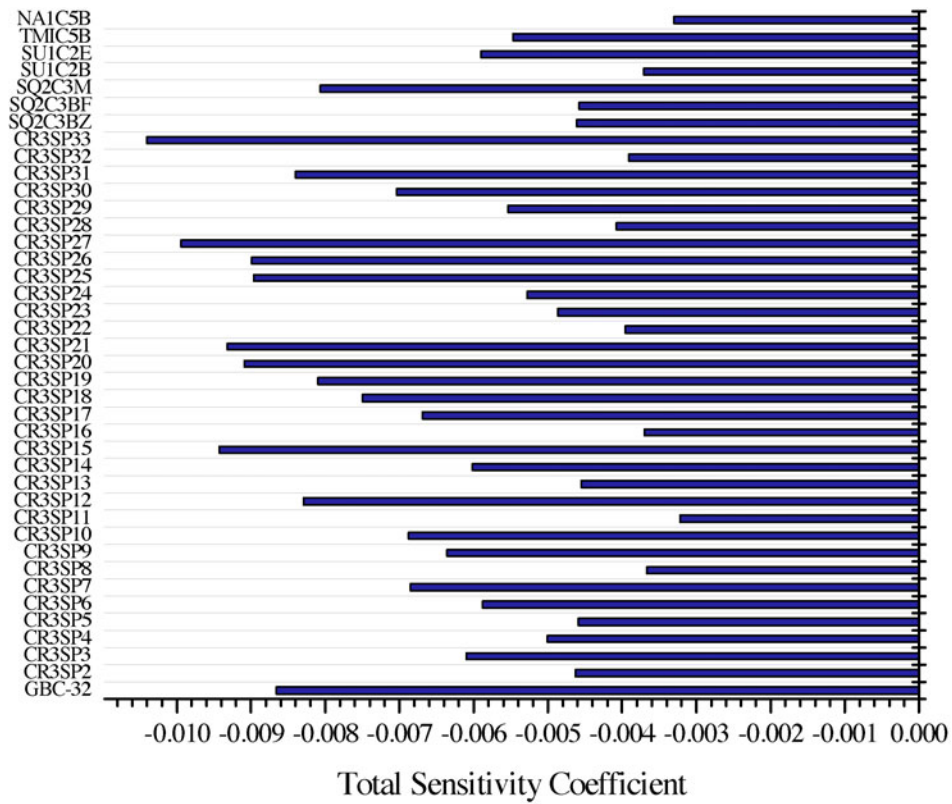


(a)

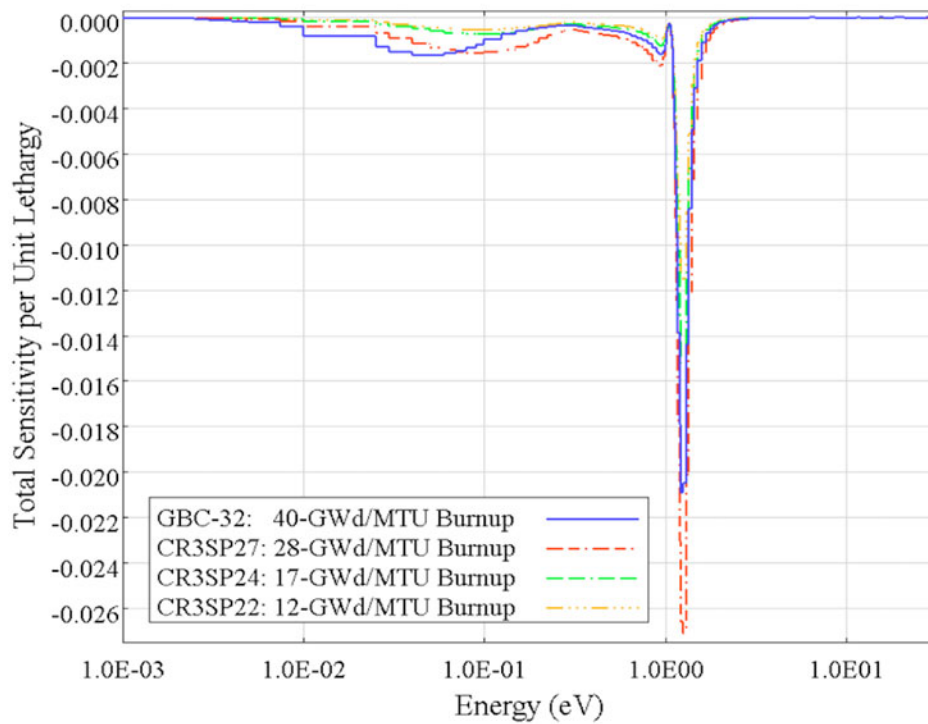


(b)

Fig. 7. (a) Energy- and region-integrated k_{eff} sensitivities to ^{143}Nd total cross section. (b) Comparison of ^{143}Nd sensitivity profiles for GBC-32 (40 GWd/tU), CR3SP27 (EOC), CR3SP24 (MOC), and CR3SP22 (BOC). (Refer to Table I for CRC state-point characteristics.)

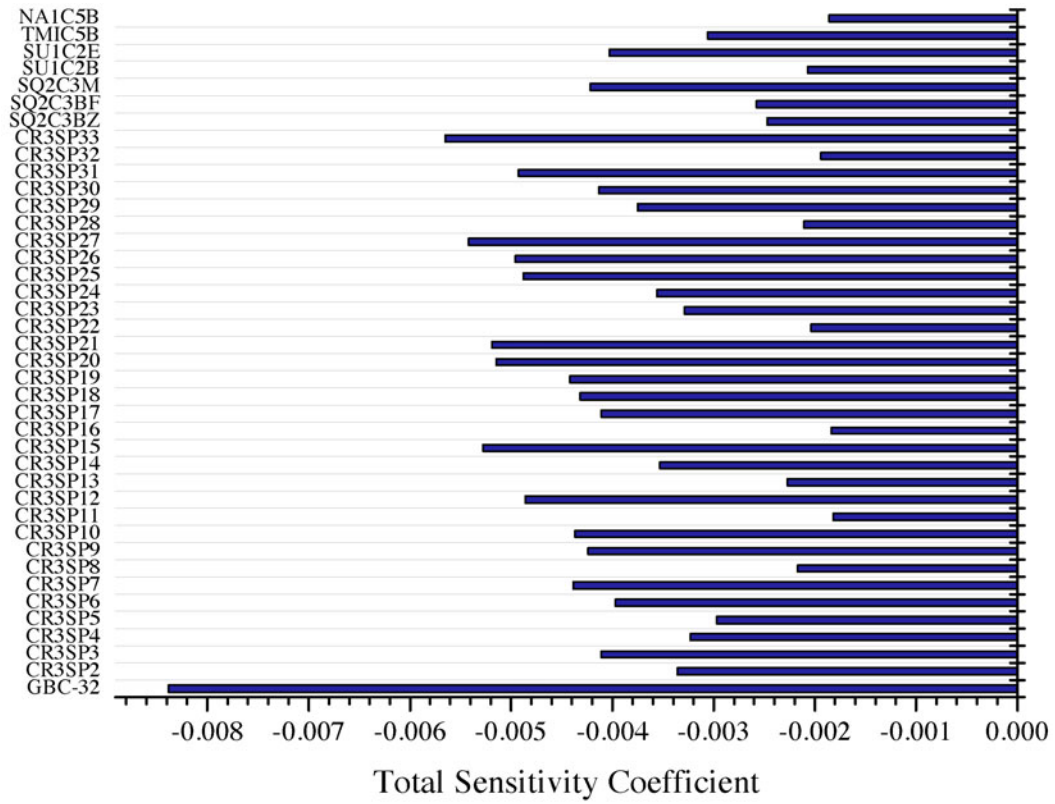


(a)

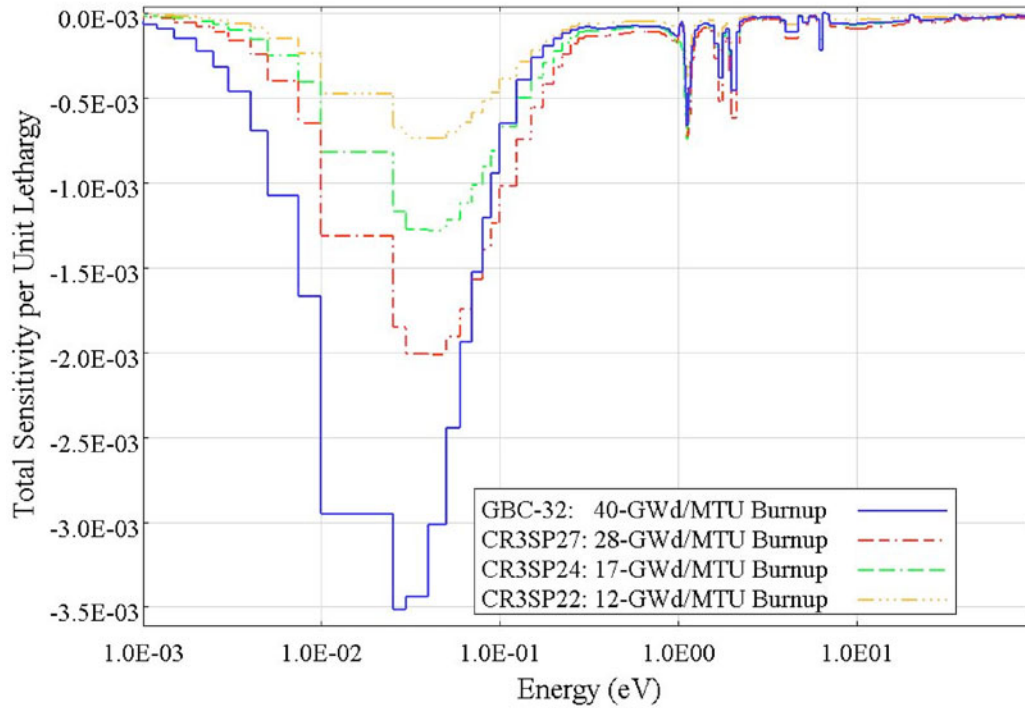


(b)

Fig. 8. (a) Energy- and region-integrated k_{eff} sensitivities to ^{103}Rh total cross section. (b) Comparison of ^{103}Rh sensitivity profiles for GBC-32 (40 GWd/tU), CR3SP27 (EOC), CR3SP24 (MOC), and CR3SP22 (BOC). (Refer to Table 1 for CRC state-point characteristics.)

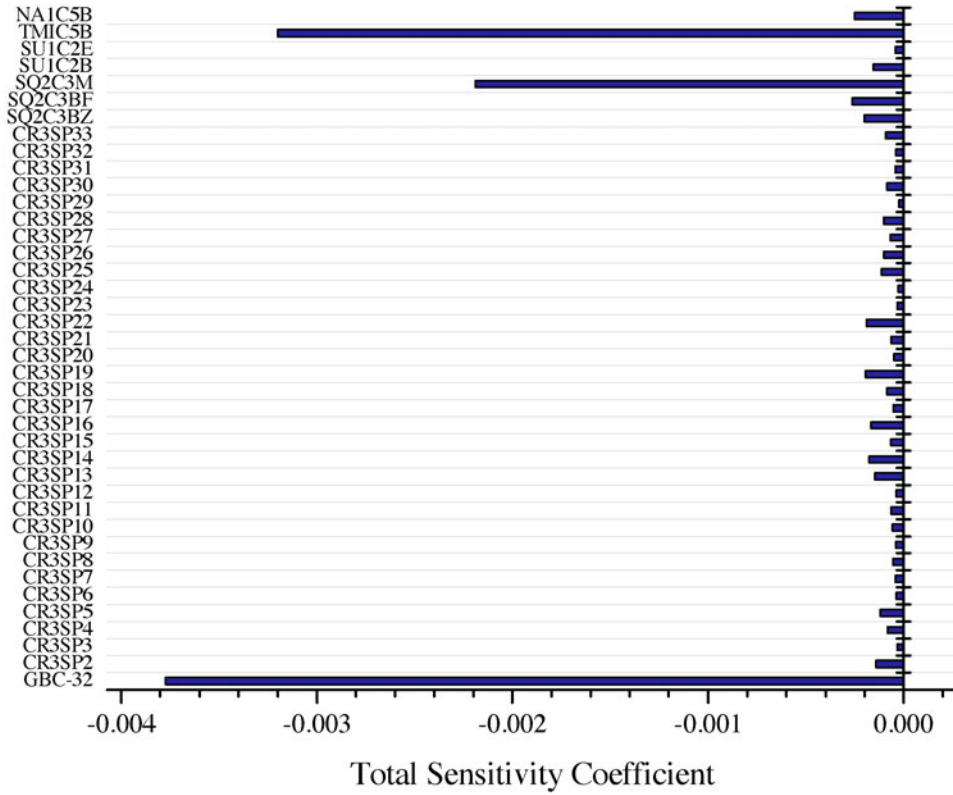


(a)

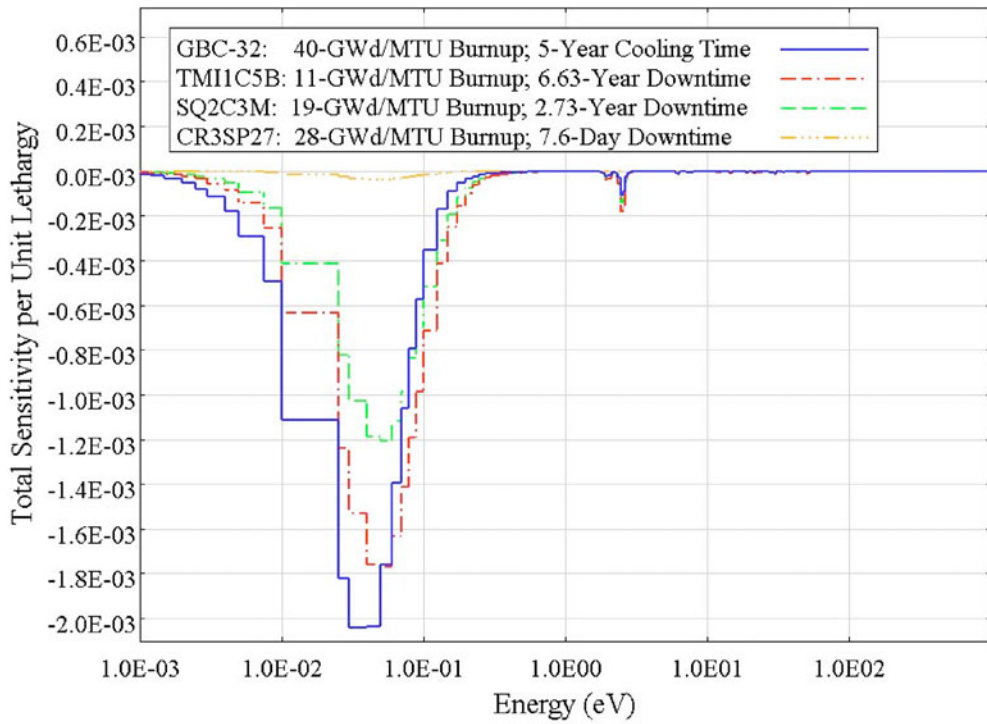


(b)

Fig. 9. (a) Energy- and region-integrated k_{eff} sensitivities to ^{151}Sm total cross section. (b) Comparison of ^{151}Sm sensitivity profiles for GBC-32 (40 GWd/t U), CR3SP27 (EOC), CR3SP24 (MOC), and CR3SP22 (BOC). (Refer to Table I for CRC state-point characteristics.)



(a)



(b)

Fig. 11. (a) Energy- and region-integrated k_{eff} sensitivities to ^{155}Gd total cross section. (b) Comparison of ^{155}Gd sensitivity profiles for GBC-32 (40 GWd/t U), TMI1C5B, SQ2C3M, and CR3SP27. (Refer to Table I for CRC state-point characteristics.)

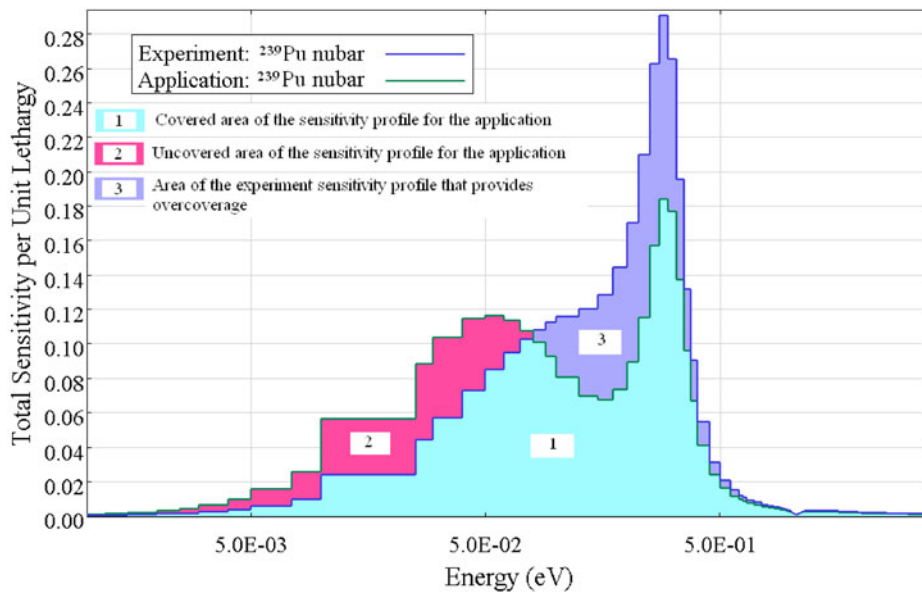


Fig. 12. Comparison of the energy-dependent k_{eff} sensitivity profiles to the ^{239}Pu nubar data for an application and an experiment.

nuclide-reaction pair. The uncovered portion of the application sensitivity profile (i.e., area 2 on the graph) is used to compute an uncertainty in k_{eff} through the cross-section-covariance data, which can be used as an additional margin to subcriticality in criticality safety applications (see Sec. III.D). Area 3 of the sensitivity profile of the experiment provides overcoverage, that is, in this example, the experiment is more sensitive to the

^{239}Pu nubar data than the application system for energy groups exceeding ~ 0.08 eV.

The CRC state-points that provide significant coverage for ^{149}Sm , ^{143}Nd , ^{103}Rh , ^{151}Sm , ^{133}Cs , and ^{155}Gd are summarized in Table III. Values of nuclide-reaction integral index g for the actinides and major fission products relevant for burnup credit in the GBC-32 cask in relation to the CRC state-points are available in

TABLE III

Summary of CRC State-Points Providing Significant Coverage for Cask k_{eff} Sensitivities to ^{149}Sm , ^{143}Nd , ^{103}Rh , ^{151}Sm , ^{133}Cs , and ^{155}Gd

Fission Product	Integral g Index Interval ^a	CRC State-Points ^b
^{149}Sm	0.7 to 0.81	Crystal River Unit 3 state-points 2, 3, 6, 7, 9, 10, 12, 14, 15, 17, 18, 19, 20, 21, 25, 27, 30, and 33
^{143}Nd	0.6 to 0.78	Crystal River Unit 3 state-points 3, 7, 9, 10, 12, 15, 18, 19, 20, 21, 25, 26, 27, 31, and 33; Sequoyah Unit 2, state-point 3
^{103}Rh	0.7 to 0.89	Crystal River Unit 3 state-points 7, 9, 10, 12, 15, 18, 19, 20, 21, 25, 26, 27, 30, 31, and 33; Sequoyah Unit 2, state-point 3
^{151}Sm	0.5 to 0.6	Crystal River Unit 3 state-points 7, 10, 12, 15, 20, 21, 25, 26, 27, 31, and 33
^{133}Cs	0.8 to 0.92	Crystal River Unit 3 state-points 7, 9, 10, 12, 15, 17, 18, 19, 20, 21, 25, 26, 27, 30, 31, and 33; Sequoyah Unit 2, state-point 3
^{155}Gd	0.7493	TMI Unit 1 state-point 1

^aResults are shown for the GBC-32 cask containing typical PWR spent fuel assemblies of 3.777 wt% ^{235}U initial enrichment, 40 GWd/t U burnup, and a 5-yr cooling time.

^bRefer to Table I for CRC state-point characteristics.

Appendix C of Ref. 12. Review of these data confirms that the EOC state-points provide better coverage of the k_{eff} sensitivities of the cask to ^{149}Sm , ^{143}Nd , ^{103}Rh , ^{151}Sm , and ^{133}Cs than the other state-points and that the integral g index values do not vary significantly with the burnup of an EOC state-point. As expected, only TMI Unit 1 BOC and Sequoyah Unit 2 MOC provide coverage of the cask k_{eff} sensitivities to nuclides such as ^{155}Gd and ^{241}Am that significantly build in after fuel discharge.

III.D. Penalty on k_{eff} due to Uncovered Sensitivity Data

A method is available in TSUNAMI-IP to assess the impact of, or penalty for, uncovered sensitivity data, when sufficient critical experiments are not available to provide complete coverage for an application. (Refer to Fig. 12 for a graphical representation of the concept of sensitivity data coverage.) This quantity is intended to represent the uncertainty in the calculated value of k_{eff} due to inadequate validation coverage and can be used as an additional margin to subcriticality. The penalty on k_{eff} due to uncovered sensitivity data by all CRC state-points with $c_k > 0.8$ was evaluated using the TSUNAMI-IP methodology.¹⁰ The values based on an assumed relative standard deviation for unknown covariance data of 25% vary from 0.79 to 0.95% $\Delta k_{eff}/k_{eff}$ for cask burnups ranging from 10 to 60 GWd/t U. For spent fuel of 40 GWd/t U burnup, the total relative standard deviation of k_{eff} due to uncovered sensitivity data is 0.93% $\Delta k_{eff}/k_{eff}$ with a contribution of 0.89% $\Delta k_{eff}/k_{eff}$ from major actinides— ^{234}U , ^{235}U , ^{238}U , ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , and ^{241}Am —and a contribution of 0.01% $\Delta k_{eff}/k_{eff}$ from all fission-product nuclides. For comparison purposes, the penalty on k_{eff} due to uncovered sensitivity data is 0.79% $\Delta k_{eff}/k_{eff}$ when assuming a relative standard deviation for unknown covariance data of 5% (instead of 25%) for the typical spent fuel of 40 GWd/t U burnup. Those results indicate that the calculated application k_{eff} could reflect $\sim 1\%$ uncertainty due to incomplete neutronic testing (validation) of the software by the CRC state-points. Contributions to uncertainty in k_{eff} (% $\Delta k_{eff}/k_{eff}$) by individual energy covariance matrices are listed in Appendix C of Ref. 12. As a cautionary note, the penalty described above addresses only sensitivity undercoverage. Overcoverage exists if experiments are significantly more sensitive to a nuclide reaction than is the safety analysis model. In this case, the contribution to the overall bias due to the nuclide reaction may be overestimated. Overcoverage is not necessarily conservative, and potentially nonconservative bias contributions due to overcoverage may be obscured by compensating bias contributions from other nuclide-reaction pairs.

IV. CONCLUSIONS

Neutronic similarities between a representative high-capacity, rail-type burnup-credit cask containing typi-

cal spent fuel (3.777 wt% ^{235}U initial enrichment, 40 GWd/t U burnup, and a 5-yr cooling time) and available CRC state-points were evaluated. Forty CRC state-points from five PWRs were selected for the study, and the characteristics of CRC state-points that may be applicable for validation of burnup-credit criticality safety calculations for spent fuel transport/storage/disposal systems were identified. The study employed cross-section S/U analysis methods developed at Oak Ridge National Laboratory and the TSUNAMI set of tools in the SCALE code system as a means to investigate system similarity on an integral and nuclide-reaction-specific level. The results indicate that except for the fresh-fuel-core configuration, all analyzed CRC state-point models are either highly similar, similar, or marginally similar to the representative cask model containing typical spent fuel with burnups ranging from 10 to 60 GWd/t U. Based on the integral system parameter c_k , approximately 30 of the 40 CRC state-points are applicable to validation of burnup credit in the representative cask containing typical spent fuel assemblies with burnups ranging from 10 to 60 GWd/t U. The state-points providing the highest similarity ($c_k > 0.95$) were those that were at or near the end of a reactor cycle. The c_k values are dominated by neutron reactions with major actinides and hydrogen, as the sensitivities of these reactions are much higher than the sensitivities of reactions with the minor actinides and fission products. On a nuclide-reaction-specific level, the CRC state-points provide significant similarity for most of the actinides and fission products relevant to burnup credit. A comparison of energy-dependent sensitivity profiles shows a slight offset between the CRC sensitivity profiles toward higher energies in the thermal region as compared with the sensitivity profile for the representative cask. Parameters representing coverage of the application by the CRC state-points on an energy-dependent, nuclide-reaction-specific level (i.e., effectiveness of the CRC state-points for validating the cross sections as used in the application) were also examined. Based on the CRC state-points with $c_k > 0.8$ and an assumed relative standard deviation for missing (i.e., unavailable) covariance data of 25%, the relative standard deviation of k_{eff} due to uncovered sensitivity data varies from 0.79 to 0.95% $\Delta k_{eff}/k_{eff}$ for casks loaded with fuel having burnup values ranging from 10 to 60 GWd/t U. As expected, this uncertainty in k_{eff} is largely dominated by noncoverage of sensitivities from major actinides and hydrogen. The contributions from fission products and minor actinides are very small and comparable to statistical uncertainties in the k_{eff} results. These results (again, assuming a 25% uncertainty for missing covariance data) indicate that there could be $\sim 1\%$ uncertainty in the calculated application k_{eff} due to incomplete neutronic testing (validation) of the software by the CRC state-points. However, this conclusion also assumes that all other uncertainties in the complex CRC configurations (e.g., isotopic compositions of burned fuel, operation history, and dimensional

and environmental data) are well known, which is not actually the case. Hence, while this work addressed the issue of neutronic similarity between CRC state-points and a representative storage and transport cask, future work is suggested to evaluate and better understand the uncertainties in the CRC configurations that should be considered in their use as part of code validation.

ACKNOWLEDGMENTS

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