SURVEY AND RECOMMENDATIONS FOR POTENTIAL SPENT NUCLEAR FUEL VERIFICATION AND QUALIFICATION ISSUES FOR TRANSPORTATION, AGING, AND DISPOSAL CANISTER SYSTEM DESIGN

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ABSTRACT

The objective of this study was (i) to survey U.S. Nuclear Regulatory Commission (NRC) regulations and staff guidance documents on potential commercial spent nuclear fuel (SNF)[1] verification issues for the transportation, aging, and disposal canister (TAD)[2] system; (ii) to summarize current NRC practices and lessons learned on SNF verification and qualification; (iii) to survey literature on the probability and potential neutronic reactivity effects of the assembly misloading into a dry cask or a canister; (iv) to conduct scoping calculations on the effect of assembly misloading into a conceptual TAD canister; and (v) to calculate a bounding heat output of the commercial SNF assembly for a wide range of the assembly power history parameters.

Based on the literature survey of the SNF parameters, the burnup and cooling time have a large effect on dose rates. The most important parameters that influence criticality safety and radiation doses were found to be fuel type, enrichment (maximum for criticality safety, minimum for radiation shielding), maximum burnup, minimum cooling time, maximum uranium mass, and maximum cobalt level.

The misloads of fresh and/or underburned assemblies into TAD canisters are factored into the U.S. Department of Energy postclosure performance assessment analyses of the proposed geologic repository. The results of this study show that misloading a single underburned assembly into a prototype TAD canister results in an increase of k_{eff} by several percent.

Recurring requests for additional information (RAI)[3] for 10 CFR Parts 71 and 72 applications in the thermal review area center on the computational modeling—specifically, on the use of computational fluid dynamics methods. Recurring RAIs in the confinement review area include definition of damaged fuel, inappropriate use of the term *leak tight*, leakage-rate testing methodology, and testing of vent/drain port welds on cask lids. Recurring RAIs in the criticality safety area occur in computer modeling and margin to criticality credit taken for the boron neutron absorber content. Recurring RAIs in the shielding review area were found in fuel burnup and use of computer codes.

Because there are indications that utilities continue increasing fuel utilization, this study presents assembly heat output rates for the parameter ranges extending beyond those available in the literature and guidance (NRC, 1998, 1994).

References

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———. NUREG/CR–5625, ORNL–6698, “Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data.” Washington, DC: NRC. July 1994.

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[1]Spent nuclear fuel is referenced throughout this section. The abbreviation SNF will be used.
[2]Transportation, aging, and disposal is referenced throughout this section. The abbreviation TAD will be used.
[3]Requests for additional information is referenced throughout this section. The abbreviation RAI will be used.

CONTENTS

FIGURES

Figure Page

TABLES

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QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

DATA: All CNWRA-generated original data contained in this report meet the quality assurance requirements described in the Geosciences and Engineering Division Quality Assurance Manual. Experimental data have been recorded in CNWRA Scientific Notebook 916E (Povetko, 2008).

ANALYSES AND CODES: CNWRA used SCALE/ORIGEN Version 5.1 (Oak Ridge National Laboratory, 2006) to calculate radionuclide inventory and radiation heat output rate corresponding to this inventory for the light water reactor spent nuclear fuel. Two codes were used for the preliminary scoping calculations on the neutronic reactivity effect of the single misloaded assembly. SCALE/ORIGEN Version 5.1 was used to generate radionuclide inventory, and MCNP Version 5 1.40 (Los Alamos National Laboratory, 2006) was used for criticality calculations. The software was qualified in accordance with CNWRA Technical Operating Procedure–018 Development and Control of Scientific and Engineering Software. Spreadsheet calculations were accomplished using Microsoft® Excel® 2003 SP2 (Microsoft Corporation, 2003). Additional calculations were performed using Mathcad Version 1.1 (Mathsoft Engineering & Education, Inc., 2005).

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1 INTRODUCTION

1.1 Background

This report summarizes a literature survey of the documents and practices pertinent to the spent nuclear fuel (SNF)[1] verification and qualification issues when a cask or a canister is loaded with SNF. The U.S. Department of Energy (DOE) is investigating a potential high-level radioactive waste geologic repository at the Yucca Mountain site in Nevada. To comply with the requirements, DOE analyses are applying burnup credit for 29 radionuclides in commercial spent nuclear fuel (CSNF)[2]. The relationship between the required minimum burnup and fuel assembly initial enrichment forms a loading curve. The CSNF loading in transportation, aging, and disposal (TAD)[3] canisters and transportation casks will be performed according to the loading curves. Therefore, the fuel assembly parameters will have to comply with the loading curve requirements in addition to thermal and radiation dose rate limits. If a fuel assembly is not qualified or its parameters are not adequately verified prior to loading, such loading constitutes a loading curve violation and an unauthorized loading (misloading). This report discusses neutronic reactivity effects related to assembly misloading. For example, if assemblies are loaded in canisters in compliance with the loading curve, the safety margin of subcriticality will be maintained through the period of 10,000 years as required by 10 CFR Part 63. If assemblies with insufficient burnup are loaded in violation of the loading curve, that would likely increase neutronic reactivity of the SNF/canister/cask system and therefore affect criticality safety by potentially reducing the margin of subcriticality to below acceptable levels for the period of 10,000 years. This report also presents the results from CSNF assembly heat output calculations for a wide range of assembly irradiation history parameters. These items are potentially important in the context of high-level radioactive waste geologic repository activities.

The potential repository capacity is limited by the Nuclear Waste Policy Act of 1982, as amended, to 70,000 metric tons of high-level radioactive waste. Nearly all of the projected 11,250 waste packages will contain SNF of some type, including commercial power, U.S. Navy, research, production, and other special reactor fuels. Current DOE plans call for the CSNF to be packaged in TAD canisters placed inside waste packages. According to DOE (2007), the TAD canister may be loaded with CSNF and sealed at the purchaser sites (e.g., reactors) under 10 CFR Part 50 or at the repository. The loaded TAD canister may be used for storage for a period of time at purchaser sites. If used for this purpose, it must be approved as a storage system certified under 10 CFR Part 72. The loaded TAD canister may be delivered to DOE for transportation to the geologic repository operations area. For these operations, the TAD canister must be approved (which includes the transportation overpack) for packaging and be certified under 10 CFR Part 63. At the geologic repository operations area, a loaded TAD canister may also be handled using a shielded transfer cask or aged in an aging overpack. Final disposition of the waste (including the waste package and the TAD canister) will be in

[1]Spent nuclear fuel is referenced frequently throughout this document. The abbreviation SNF will be used.
[2]Commercial spent nuclear fuel is referenced frequently throughout this document. The abbreviation CSNF will be used.
[3]Transportation, aging, and disposal is referenced frequently throughout this document. The abbreviation TAD will be used.

accordance with the requirements of 10 CFR Part 63. According to DOE (2007), the term "approved contents" in the context of the performance specifications means one of the following:

> "Transportation Overpack: The contents of Type B packaging as defined by the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 7.9 Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material and listed in Section 5b 'Contents of Packaging' of Certificates of Compliance issued under 10 CFR Part 71."
>
> "Storage Overpack: The materials to be stored as defined in NRC Regulatory Guide 3.61 Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask and listed in Section 6 'Approved Contents' of Certificates of Compliance issued under 10 CFR Part 72."

The TAD canister system, therefore, must comply with four federal regulations. The licensing requirements of these regulations are different because their scopes are different and also because 10 CFR Part 63 as compared to Parts 50, 71, and 72, is a risk-informed, performance-based regulation. An overall summary and subject comparison of NRC regulatory criteria and review plans applicable to the TAD canister covering requirements from 10 CFR Parts 63, 71, and 72 are presented in NRC (2006a).

To demonstrate compliance with the postclosure period (i.e., the period of 10,000 years after permanent closure of the geologic repository) performance objectives outlined in 10 CFR Part 63, DOE will conduct a performance assessment to quantitatively estimate radiological exposures to the reasonably maximally exposed individual at any time during this postclosure compliance period. The features, events, and processes considered in the performance assessment will represent a wide range of both beneficial and potentially adverse effects on performance (e.g., beneficial effects of radionuclide sorption, potentially adverse effects of fracture flow or a criticality event). DOE plans to screen out those features, events, and processes having low probability of occurrence, effectively excluding them from the postclosure performance assessment. Current reports indicate that one of the events that DOE may exclude is a criticality event during the postclosure period (Sandia National Laboratories, 2007a). For the period before permanent repository closure (the preclosure period), DOE will conduct a preclosure safety assessment to consider events that have at least a 1 in 10,000 chance of occurring during the preclosure period. In this safety assessment, DOE may elect to demonstrate that event sequences important to criticality have less than a 1 in 10,000 chance of occurring during the preclosure period (Bechtel SAIC Company, LLC, 2007a). The bases for the analyses supporting postclosure and partially supporting preclosure screening arguments may include use of burnup credit (i.e., the reduction in neutronic reactivity due to fuel burnup) for CSNF, performance of corrosion resistant neutron absorbers, and use of prototype TAD canisters with or without neutron flux traps. These DOE criticality analyses provided technical bases for the specifications and postclosure criticality loading curves (DOE, 2007) of the TAD canister postclosure criticality controls.

The period before permanent closure of the repository is called the preclosure period. Its duration is not prescribed by the regulations and is expected to last between 50 and 300 years. The preclosure criticality safety and postclosure criticality screening analyses are conducted based on the assumption that undamaged fuel will remain intact and the design basis fuel/neutron absorber mutual configuration will remain unchanged during pre- and postclosure periods. Damaged fuel is treated separately and differently from intact fuel. Therefore, it is important to verify the SNF condition prior to TAD canister loading because these two waste streams will be treated differently in actual handling.

10 CFR Part 63 requires that DOE performs a preclosure safety assessment for the geologic repository operations area. DOE indicates that in this preclosure safety assessment they will rely heavily on the quality assurance programs of the commercial and DOE SNF loading operators (DOE, 2008). For example, the operations that take place outside the geologic repository operations area site are not considered as initiating events in the safety assessment. DOE expects that all shipments to the repository must be loaded in accordance with the certificate of compliance for a specific transportation cask that is licensed under 10 CFR Part 71.

1.2 Objective and Scope

This report surveys potential SNF verification and qualification issues for the TAD system design focusing on compliance with the requirements of 10 CFR Part 63. Reliance on burnup credit and performance of neutron absorbers during pre- and postclosure periods requires that DOE demonstrate that each fuel assembly qualifies (i.e., meets requirements of the specific loading curve imposed on assembly nuclear parameters) for the loading. Potential assembly misloads and neutron absorber plate misplacements may reduce the criticality safety margin. Because pre- and postclosure criticality safety analyses may rely on the intact fuel/neutron absorber configuration, each assembly may also qualify for loading with respect to the assembly damage. TAD performance specifications limit SNF cladding temperatures and canister surface temperatures (DOE, 2007). Because postclosure hydrogeologic analyses rely on certain thermal outputs of the waste packages placed in the disposal drifts, the heat output of the loaded TAD canister is of great importance; therefore, assembly heat outputs are investigated in the study.

Section 2 of this report describes the regulatory criteria, NRC staff guidance, and current practice on SNF verification and qualification. Section 3 of this report addresses assembly nuclear parameters and the potential effects of assembly misloads and neutron absorber misplacements. Section 4 presents results of CSNF assembly heat output analyses for a wide range of CSNF parameters. Section 5 contains a summary and conclusions of this report.

2 10 CFR PARTS 50, 63, 71, AND 72 REGULATORY CRITERIA, GUIDANCE, AND REGULATORY PRACTICE APPLICABLE TO A SPENT NUCLEAR FUEL CONDITION

2.1 Spent Nuclear Fuel Parameter Verification Requirements Under 10 CFR Parts 50, 63, 71, and 72

This section describes regulatory criteria, U.S. Nuclear Regulatory Commission (NRC) staff guidance, and current practice on spent nuclear fuel (SNF)[1] verification and qualification. The literature survey includes NRC regulations; standard review plans (SRPs)[2], one NUREG, and one NUREG/CR document.

Table 2-1 presents the SNF provisions excerpted from the NRC regulations.

2.2 Spent Nuclear Fuel Parameter Verification and Qualification Requirements in NRC Standard Review Plans

Table 2-2 compiles SNF parameters for disposal, transportation, and interim storage analyses discussed in NRC staff guidance documents and SRPs.

2.3 The Most Important and Typical Spent Nuclear Fuel Parameters Provided in Safety Analysis Reports

To support the NRC licensing procedures for SNF dry cask storage casks, Oak Ridge National Laboratory performed a study (NRC, 2001a) to identify and rank potential SNF specification parameters needed for criticality safety and radiation shielding and rank their importance to a potential compromise of the margin of safety.

The study results for the shielding parameters are summarized in Table 2-3. They are not intended to cover the full range of SNF and assembly designs. The results provide only a rough guide to the importance of the fuel specifications as they affect storage cask surface total (i.e., gamma and neutron) dose rates or k_{eff}. The study (NRC, 2001a) also ranks the parameter importance relative to a potential compromise of the margin of safety. The parameters having the largest impact on the dose rate are burnup and cooling time. According to the report, the assemblies with large amounts of stainless steel may have a large effect on dose rates due to the cobalt content. The enrichment level is found to be important, and its importance increases as the neutron-to-gamma (i.e., secondary gamma) dose rate for the cask increases. The report (NRC, 2001a) lists the following most important parameters that influence criticality safety and radiation doses:

- Fuel Type

[1]Spent nuclear fuel is referenced frequently throughout this document. The abbreviation SNF will be used.
[2]Standard Review Plans is referenced frequently throughout this document. The abbreviation SRPs will be used.

Table 2-1. Spent Nuclear Fuel (SNF) Parameter Requirements Under 10 CFR Parts 50, 63, 71, and 72

Regulatory Framework/SNF Parameter	Domestic Licensing, Production, and Utilization Facilities 10 CFR Part 50	Disposal 10 CFR Part 63	Transportation 10 CFR Part 71	Interim Storage 10 CFR Part 72
Definition of Spent Nuclear Fuel (SNF)	None.	Irradiated reactor fuel is one of the possible components of the high-level radioactive waste as high-level radioactive waste is defined in 10 CFR 63.2.	SNF means “fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least 1 year’s decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. SNF includes special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies” [10 CFR 71.4].	“Fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least 1 year’s decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. SNF includes special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies” [10 CFR 72.3].
Cladding Properties	None.	(1) Preclosure safety analysis “must include but not be limited to” consideration of means to limit concentration of radioactive material in air [10 CFR 63.112 (e)(1)]. (2) Postclosure performance assessment: if cladding is a component of engineered barrier system, then performance objectives for the geologic repository after permanent closure are applied [10 CFR 63.113].	None.	None.

Table 2-1. Spent Nuclear Fuel (SNF) Parameter Requirements Under 10 CFR Parts 50, 63, 71, and 72 (continued)				
Regulatory Framework/SNF Parameter	**Domestic Licensing, Production, and Utilization Facilities 10 CFR Part 50**	**Disposal 10 CFR Part 63**	**Transportation 10 CFR Part 71**	**Interim Storage 10 CFR Part 72**
Standards for Packages	None.	None.	Package temperature: "a package must be designed so that in still air at 38 °C [100 °F] and in the shade, no accessible surface of a package would have a temperature exceeding 50 °C [122 °F] in a nonexclusive use shipment or 85 °C [185 °F] in an exclusive use shipment" [10 CFR 71.43]. Test conditions: "ambient air temperature before and after the test must remain constant at the value between −29 and 38 °C [−20 °F and 100 °F] [10 CFR 71.73(b)]." Thermal test: "Exposure of the specimen fully engulfed (…) in a hydrocarbon fuel/air fire (…) to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of 800 °C [1,475 °F] for a period of 30 minutes, or any other equivalent thermal test" [10 CFR 71.73(b)(4)]. External radiation: "–each package(…) must be designed (…) so that	None.

Table 2-1. Spent Nuclear Fuel (SNF) Parameter Requirements Under 10 CFR Parts 50, 63, 71, and 72 (continued)				
Regulatory Framework/SNF Parameter	**Domestic Licensing, Production, and Utilization Facilities 10 CFR Part 50**	**Disposal 10 CFR Part 63**	**Transportation 10 CFR Part 71**	**Interim Storage 10 CFR Part 72**
			under conditions normally incident to transportation the radiation level does not exceed 2 mSv/h [200 mrem/h] at any point on the external surface of the package and the transport index does not exceed 10” [10 CFR 71.47(a)]. A package that exceeds the radiation level limits specified in 10 CFR 71.47(a) “must be transported by exclusive use shipment only; and the radiation levels for such shipment must not exceed during transportation the limits in 10 CFR 71.47(b).” Hypothetical accident conditions: “There would be no escape of krypton-85 exceeding 10 A_2 [10 CFR Part 71, Appendix A)] in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h [1 mrem/h] at 1 m [40 in] from the external surface of the package” [10 CFR 71.73].	

Table 2-2. Spent Nuclear Fuel Parameters				
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
Criticality	Methods of criticality control are specified (i.e., geometry, fixed poisons, borated pool water). Procedures of controlling boron concentration in fixed poisons in the confinement cask or in pool. K_{eff} is less than 0.95 (with 95 percent probability and 95 percent confidence) for all normal events, off-normal events, and postulated events (Section 8.4.1.1). Design criteria require that "the calculation on k_{eff} includes the effects of maximum fresh fuel enrichment, optimum moderation, and computer code computational and experimental benchmark bias" (Section 4.5.3.5). <u>Burnup credit definition</u>: unirradiated reactor fuel, of well-specified nuclide composition, provides a bounding approach to the criticality safety analysis of transport and storage casks.	Single package evaluations: the package "must be designed and constructed and its contents limited so that it would be subcritical if water were to leak into the containment system." "A single package must be subcritical under tests for hypothetical accident conditions" (Section 6.3.4). Packages and contents considered in Section 6.3.4 are in most reactive condition (Section 6.4.4). Array of packages evaluations: determine "the maximum number of packages that may be transported in a single shipment" under normal conditions of transport. Determine the maximum number of packages that may be transported in a single shipment under hypothetical accident conditions of transport (Section 6.3.6).	Material properties: "fissile material properties must be assumed to be those which will result in the highest neutron multiplication." (Section 6.3). "Single packages should be subcritical if water were to leak into the containment system." (Section 6.3). Transport index for criticality control "must be assigned to limit the number of packages in a single shipment" (Section 6.3). Fissile material contents: specifications "include fissile material mass, dimensions, enrichment, physical and chemical	Section 6 applies to mixed oxide radioactive materials and low-enriched uranium radioactive materials.

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
	Ignoring the presence of burnable poison, as the fuel is irradiated in the reactor the nuclide composition changes causing the reactivity of the fuel to decrease. Allowance in the criticality safety analysis for the decrease in fuel reactivity resulting from irradiation is typically termed “burnup credit” (Section 8.4.5). Fresh fuel provides the worst case criticality analysis, so no burnup credit is taken (Section 8.4.1.1). Spent nuclear fuel, however, by definition [10 CFR 72.3], has been withdrawn from a nuclear reactor following irradiation, and has undergone at least 1 year’s decay. Alternative guidelines when burnup credit may be taken are provided in the following paragraphs. Burn-up credit granting: the amount of burnup credit should be limited “to that available from actinide compositions associated with pressurized water reactor irradiation of UO_2 fuel to an assembly-average burnup value of 40 GWd/MTU or less.” “The initial enrichment of the fuel assumed for the licensing–basis	Burnup credit: “no regulatory requirements are specific to burnup credit; general criticality requirements apply” (Section 6.3.8). Licensing-basis analysis performed to demonstrate criticality safety limits the amount of burnup credit to that available from actinide compositions associated with pressurized water reactor irradiation of UO_2 fuel to assembly-average burnup of 40 GWd/MTU. This analysis is restricted to intact spent nuclear fuel assemblies with 5-year out-of-reactor cooling time; the initial enrichment of the fuel should be no more than 4.0 wt% U-235, unless an loading offset is	composition, moisture, and other characteristics dependent on specific contents.” (Section 6.5.2). Under normal conditions of transport and hypothetical accident conditions, “the relative location and physical properties of the contents within the packaging should be justified as those resulting in the maximum multiplication factor” (Section 6.5.3.1). Appropriate mass and atom densities are provided for materials used in the models of the packaging and contents (Section 6.5.3.2). Poison materials: Criticality evaluations for packaging “should generally not consider	

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
	analysis should be no more than 4.0 wt% U-235 unless a loading offset is applied." "The loading offset is defined as the minimum amount by which the assigned burnup loading value (…) must exceed the burnup value used in the licensing safety basis analysis." The loading offset should be at least 1 GWd/MTU for every 0.1 wt% increase in initial enrichment above 4 percent. In any case, the initial enrichment should not exceed 5 wt%. "For example, if the safety analysis demonstrates a appropriate subcritical margin for 4.5 wt% fuel burned to the limit of 40 GWd/MTU, the loading curve (…) should be developed to ensure that the assigned burnup loading value is at least 45 GWd/MTU" (Section 8.4.5.1). Loading curve: are "curves that plot as a function of initial enrichment the assigned burnup loading value above which fuel assemblies may be loaded in the cask. Loading curves should be based on a 5-year cooling time and only fuel cooled at least 5 years should be loaded in a cask approved for burnup credit" (Section 8.4.5.4).	applied (Section 6.4.8.1). Neutron poison: criticality evaluations for packaging "should generally not consider more than 75 percent of the specified neutron poison concentrations." (Section 6.4.3) "The sum of k_{eff}, two standard deviations (95 percent confidence), and the bias adjustment should not exceed 0.95 to demonstrate subcriticality by calculation." Biases reducing the calculated value of k_{eff} "should not be applied" (Section 6.4.3). Loading curves based on a 5-year cooling time and only spent nuclear fuel cooled at least 5 years should be loaded in a cask approved for burnup credit (Section 6.4.8.4).	more than 75 percent of the minimum specified neutron poison. Verify that materials will not degrade during the service life of the packaging" (Section 6.5.3.2). Computer codes and cross-section libraries: appropriate cross-section libraries are used, and the neutron spectrum of the package has appropriately been considered. Key input data for criticality calculations are used (i.e., number of neutrons per generation, number of iterations, convergence criteria, mesh selection) (Section 6.5.3.3). Maximum reactivity: analyses demonstrate the maximum reactive configuration of single package, array of undamaged packages,	

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
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			and array of damaged packages (Section 6.5.3.4).	
Thermal Output	Design criteria include maximum heat output of the radioactive materials, material temperature limits (Section 6.5.1.1). In Dry Storage Systems, fuel cladding temperatures are close (within 5 percent) to their limiting values during an accident. "Maximum temperatures (under normal conditions) of the pool water and/or other water used in the cask cavity during loading and unloading operations are below the temperature assumed in the cask criticality safety analysis if a time restriction exists in the corresponding technical specifications" (Section 6.5.1.2). Pool Systems: the bulk temperature of the pool will be kept as indicated in Section 6.5.1.3. Dry Transfer Systems: ensure that fuel cladding temperature will not exceed, "under normal, off-normal, and accident conditions," 570 °C [1,058 °F] (Section 6.5.1.4).	Accessible surface temperatures of a package in still air at 38 °C [100 °F] in the shade should not exceed 85 °C [185 °F] in an exclusive-use shipment (Section 6.3.3.4). Normal conditions of transport: preceding and following the tests, ambient temperature(…) must remain near constant at the value between −29 °C [−20 °F] and +39 °C [102 °F] "which are the most unfavorable for the feature under consideration" (Section 3.3.5). Hypothetical accident conditions: "–Except for water	Contents decay heat: Ensure that decay heat is properly determined from the maximum allowed radioactive contents (Section 3.5.1.2). Temperatures: Maximum and minimum temperatures affecting shielding and criticality "are presented for normal conditions of transport and hypothetical accident conditions" (Section 3.5.1.3). Margins of safety for temperatures are appropriately evaluated (Section 3.5.3.3). Hypothetical accident conditions: "Assume contents of the package at its maximum decay heat, unless a lower heat (consistent with	Contents decay heat generation for mixed oxide radioactive materials is "four or five orders of magnitude higher than for low-enriched uranium radioactive materials" (Section 3.5.1.3). Under normal conditions of transport: "pressure from hydrogen and/or other gases" may be produced by thermal- or radiation-induced decomposition of moisture associated with impure plutonium-containing oxide powders; during the process of converting powder to pellets/rods, processing temperatures should have removed all the impurities from the plutonium oxide. From this point, "mixed-oxide pellets and low-enriched uranium pellets should be virtually identical" (Section 3.5.4.2).

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
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		immersion tests (…) preceding and following the tests, ambient temperature(…) must remain near constant at the value between 29 °C [84 °F] and +39 [102 °F] °C which are the most unfavorable for the feature under consideration." The 30-minute, 800 °C [1,472 °F] fire test of 10 CFR 71.73(c)(4) on a damaged package is the primary thermal test for hypothetical accident conditions [10 CFR 71.73].	temperature and pressure conditions) is less favorable" (Section 3.5.5.1).	Thermal stresses in fuel/clad shall be determined; usually they are small because the temperature gradients in metal are small (Section 3.5.4.3). Under hypothetical accident conditions, "for mixed oxide-fresh-fuel rods and assemblies the internal heat load of the mixed oxide-fresh-fuel contents shall be at its maximum allowable power unless a lower power consistent with temperature and pressure is more unfavorable." "For mixed oxide powders and fuel pellets, the internal heat load of the mixed oxide-fresh-fuel contents shall be at its maximum allowable power unless a lower power consistent with temperature and pressure is more unfavorable" (Section 3.5.5.1). Maximum temperatures and pressures: "(i) for

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
				mixed oxide-fresh-fuel rods and assemblies, possible increases in gas inventory due to fuel pellet failure should be considered in the pressure determination; (ii) for mixed oxide powders and fuel pellets," processing temperatures should have removed all the impurities, "so the only additional increase in pressure should be result of any helium released from the contents, as a result of the increased temperature" (Section 3.5.5.3).
Shielding	Shielding design features of the independent spent fuel storage installation and monitored retrievable storage should "meet the NRC criteria for protection against direct radiation from the material to be stored" (Section 7.1). Contained radiation "sources of each type should be described as a basis for shield design calculations" (Section 7.4.1).	"Contents must be described in sufficient detail to provide an adequate basis for their evaluation" (Section 5.3.2). Radiation level limits for package or freight container and roadway or railway vehicle for exclusive-use shipments are: –2 mSv/hr	Package design meets "the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions" (Section 5.1). Examples for summary table of external radiation levels (non-exclusive use) are given in Table 5.2. Normal	Supplement 1 shows only the significant deviations in the review procedures due to mixed oxide radioactive materials. Gamma source: although the decay photon emission rate for mixed oxide radioactive materials can be larger than the one for low-enriched uranium radioactive materials by one or more orders of

Table 2-2. Spent Nuclear Fuel Parameters (continued)					
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
	Gamma sources: radiological characteristics for each gamma –ray source type must be provided, including isotopic composition and photon yields by x- and gamma-ray energy group. Both gamma source terms for spent nuclear fuel and activated materials must be specified. Energy group structure from the source term calculation must correspond to that of the cross-section set of the shielding calculations. Source terms for induced radioactivity by interactions involving neutrons originating in the stored materials must be described (Section 7.4.1.1). Neutron sources: the bases used to determine the neutron source terms must be described; neutron yield by energy group must be tabulated (Section 7.4.1.2).	[200 mrem/hr] at the external surfaces and undersides of the package on open (flat-bed) vehicles, and 10 mSv/hr [1,000mrem/hr] at the external surfaces of the package on open vehicles; –2 mSv/hr [200 mrem/hr] at points of the outer surface of the vehicle, depending on position (Table 5.1). –0.02 mSv/hr [2 mrem/hr] at the occupied positions (Table 5.1). Under the tests specified in 10 CFR 71.73 (hypothetical accident condition), the external radiation levels at 1 m [40 in] from the package surface must not exceed 10 mSv/h [1 mrem/h] (5.3.4). Package and radiation level limits for exclusive-use	conditions of transport and hypothetical accident conditions are compared to 10 CFR 71.47(a) and 10 CFR 71.51(a) limits, respectively. These limits are (mSv/hr): 2 (at package surface), and 0.1 {1 m [40 in] from package surface} for non-exclusive use packages, and 10 {1 m [40 in] from package surface} for hypothetical accident conditions (5.5.1.1). Package and vehicle radiation level limits are given in Table 5.2. Radiation source: the contents producing the highest external dose rate at each location are clearly identified and evaluated for packages designed for multiple types of contents (Section 5.5.2).	magnitude, "no significant differences in the general methods to be used to review mixed oxide radioactive materials and low-enriched uranium radioactive materials exist." Gamma source should be appropriately described for mixed oxide radioactive materials containing various grades of plutonium. Appendix C includes information on gamma emission rates from mixed oxide radioactive materials containing different grades of plutonium (Section 5.5.2.1). Neutron source: mixed oxide radioactive materials neutron dose rate "can be much larger than the gamma dose rate;" therefore, appropriate neutron source should be determined. "Contribution from (alpha, n) reactions can be large relative to spontaneous fission for mixed oxide radioactive

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
		shipments (Table 5.1). General source specification: "the ranges of fuel type, burnup, enrichment, and cooling time should be stated." Source terms will be typically determined by using computer codes such as ORIGEN-S (or a SAS2 sequence of SCALE), ORIGEN2, or the DOE Characteristics Data Base. "Verify that the cross-section library is appropriate for the fuel being considered (for example, many libraries are not appropriate for burnup in excess of 33,000 MWd/MTU)" (Section 5.5.2)§. Gamma source: should be specified as a function of energy for both spent nuclear fuel and activated hardware. "In general, only gammas from	Gamma source: maximum gamma source strength and spectra are evaluated by appropriate methods (standard computer codes, hand calculations); source contribution from radioactive daughter products is included if it produces higher dose than the contents w/o decay; source term determination should be presented as a listing of gammas per second, or MeV per second, as a function of energy; secondary gammas from (n, gamma) reactions should be accounted for (Section 5.5.2.1). Neutron source: neutrons from spontaneous fission in the transuranics and (alpha, n) reactions in the fuel should be considered. Appropriate justification should be provided if any of this	materials." Neutron multiplication effects "can be also important." Appendix C includes information on neutron emission rates from mixed oxide radioactive materials containing different grades of plutonium (Section 5.5.2.2).

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
		approximately 0.8 to 2.5 MeV significantly contribute to external radiation levels‖.” Regrouping photons by energy may be necessary. Source term for fuel assembly hardware depends on the concentration of impurities, neutron flux during irradiation, etc.; effort for reviewing the calculation should be appropriate to the radiation levels (Section 5.5.2.1). Neutron source: should be expressed as a function of energy. The neutron energy group spectrum may be independently determined by selecting the nuclide with the predominant contribution to the spontaneous fission, such as Cm-244. Neutron source will generally result from (i) spontaneous fission	contribution to neutron source is negligible. Neutron multiplication in the subcritical fissile material should be included in the analysis (Section 5.5.2.2).	

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
		in the transuranics (predominant mechanism) and (ii) (alpha, n) reactions in the fuel. Neutron multiplication in the fissile material should be included in the analysis. Fissile content assumed for the multiplication effect should be justified and conservative (Section 5.5.2.2).		
Material Temperature Limits	Fuel cladding: generally, sufficiently low to prevent cladding failure during storage (Section 6.5.2.1). Normal conditions of storage: cladding temperatures for each fuel assembly type will be below their expected damage threshold. For Zircalloy, the temperature limit at the beginning of dry storage is 380 °C [716 °F] (5-year cooled fuel assemblies) and 340 °C [644 °F] (10-year cooled fuel assemblies), 20-year minimum cask storage. For off-normal and accident conditions, 570 °C [1,058 °F] is acceptable (acceptable for fuel transfer operations also).	None.	None.	None.

Table 2-2. Spent Nuclear Fuel Parameters (continued)				
Parameter	**Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities, NUREG–1567***	**SRP for Transportation Packages for Spent Fuel, NUREG–1617†**	**SRP for Transportation Packages for Radioactive Material, NUREG–1609‡**	**SRP for Transportation Packages for Mixed Oxide—Radioactive Material, NUREG–1609‡, Supplement 1**
	Higher burnup may lower these limits. Information to show that high burnup fuels will remain intact for the licensing period will be provided by the applicant (Section 6.5.2.2).			

*NRCNUREG–1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities." Washington, DC: NRC. March 2000.
†NRCNUREG–1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." Washington, DC: NRC. January 2000.
‡NRCNUREG–1609, "Standard Review Plan for Transportation Packages for Radioactive Material." Washington, DC: NRC.1999.
NUREG–1609 Supplement 1 report is not is not a stand-alone document; it is intended as a supplement to NUREG–1609, providing details on package review guidance resulting from the differences between contents of low-enriched uranium oxide radioactive materials and contents of mixed oxide radioactive materials.
§The validation of ORIGEN-ARP is based primarily on the validation of TRITON/NEWT, which is used to generate the cross-section libraries (Bowman, S. "Latest Validation Reports." E-mail communication (July 1, 2008) to R. Nes, Center for Nuclear Waste Regulatory Analyses. Oak Ridge, Tennessee: ORNL. 2008). The highest burnup for which these cross section libraries are validated is 47 GWd/MTU (NRCNUREG/CR–6798, "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples From the Takahama-3 Reactor." ORNL/TM–2001/239. Washington, DC: NRC. 2003).
||In common NRC practice, however, the lower limit for gamma energy contributing to external dose is extended to approximately 0.4 Mev. (Holtec International. "Final Safety Analysis Report for the HI-STORM 100 Cask System." Marlton, New Jersey: Holtec Center. 2006).

Table 2-3. Summary of NUREG/CR–6716 Fuel Specification Study Findings*			
Technical Specifications (Candidate)	**Parameter Range Studied**	**Observed Variation in Total Cask Surface Dose Rate Corresponding to Parameter Range Studied†**	**Bounding Value With Respect to Total Cask Surface Dose Rate**
Initial Enrichment (weight percent U-235)	2.5–5.0	100 percent	Minimum Enrichment
Cooling Time, Years	5–100	>100 percent	Minimum Cooling Time
Assembly Burnup, MWd/MTU	20–60	>100 percent	Maximum Burnup
Assembly/Cask Uranium Mass, kg U	Mean + or – 20 percent	20 percent	Maximum Mass
Fuel Assembly Type (No Burnable Poisons)	ABB-CE 14 × 14, W 15 × 15, W 17 × 17	5 percent	17 × 17 Design
Integral Burnable Poison Rods	5 weight percent Gd_2O_3, 2 weight percent Er_2O_3	5 percent	Maximum Poison Loading
Burnable Poison Rods	12 Borosilicate Glass and Stainless Steel Rods	10 percent	Maximum Poison Loading and Cobalt Level (If Applicable)
Assembly Structural Materials	Stainless Steel Cladding	>100 percent	Maximum Cobalt Level
Moderator Density, g/cm^3 [kg/m^3]—Boiling Water Reactor	0.3–0.7	10 percent	Minimum Moderator Density
Specific Power, MW/MTU	20–40	10 percent after 5 years <5 percent after 10 years	Maximum Specific Power

*Adapted from NRCNUREG/CR–6716, "Recommendation on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks." ORNL/TM–2000/385. Washington, DC: NRC. 2001.

†Approximate maximum variation in total dose rate expressed as percentage difference over full range = (max/min–1) × 100, concrete storage cask design.

    — Array size; number of fuel rods, including number of partial length rods (where applicable); and cladding type

    — Number and material of guide and instrument tubes

- Enrichment (maximum for criticality safety, minimum for radiation shielding)
- Maximum burnup
- Minimum cooling time
- Maximum uranium mass
- Maximum cobalt level

Other criticality safety parameters (e.g., pitch, pellet outer diameter, clad thickness, clad outer diameter, guide tube and water rod thickness, and fuel stack density) and radiation shielding parameters (e.g., maximum poison loading, minimum boiling water reactor moderator density, and maximum specific power) were found to be less important by their influence on k_{eff} and dose rates, respectively.

NRC (2001b) also lists the following parameters as the most important for 10 CFR Part 72 compliance:

- Fissile isotopes (UO_2 versus mixed oxide)
- Maximum initial (planar average) enrichment
- Fuel class (e.g., 14 × 14, 15 × 15)

    — Number of fuel rods
    — Number of water holes (i.e., assembly cells with no fuel rods)

- Maximum assembly average burnup
- Minimum cooling time after reactor shutdown
- Minimum active fuel average enrichment
- Cladding material
- Nonfuel hardware [e.g., burnable poison rod assembly/thimble plug devices (cooling time and burnup)]

Maximum weight per storage location (including fuel channels and nonfuel hardware)

- Maximum decay heat per storage location
- Fuel condition (intact, damaged, or debris)

The typical list of parameters used by cask/canister vendors in their safety analysis reports is presented in the upper half of Table 2-4. For comparison, the lower half of the table contains TAD canister system performance specifications for SNF parameters that DOE specifies for potential vendors (DOE, 2007). In the loading curve determination, in absence of the TAD system canister designs, DOE used values (Sandia National Laboratories, 2008) for these parameters specified either in DOE (2007) or extracted from the typical safety analysis report (Holtec International, 2002) and previous DOE studies.

Table 2-4. Transportation, Aging, and Disposal (TAD) Canister Spent Nuclear Fuel (SNF) Evaluation Parameters		
Representative Cask Vendor Safety Analysis Report SNF Parameters		**SNF Assembly Parameters and Parameter Specifications**
	General	Clad Material
		Design initial uranium (kg/assembly)
		Initial enrichment (wt% U-235)
		Density of UO_2
		Number of fuel rods
		Clad outside diameter
		Clad inside diameter
		Pellet diameter
		Active fuel length
		Number of guide tubes
		Guide tube thickness
		Fuel rod pitch
		Channel thickness
		Number of water rods (boiling water reactor)
		Water rod thickness (boiling water reactor)
		Principal isotopes for commercial SNF burnup credit
	Physical	Maximum assembly width
		Maximum assembly length
		Maximum assembly weight
		Maximum active fuel length
	Radiological and Thermal	Maximum heat generation (W)
		Maximum average burnup (MWd/MTU)
		Minimum cooling time (years)
		Spent fuel isotopic compositions
		Pressurized water reactor assembly axial profile data
		Bounding uncertainty in assembly burnup values
TAD Canister System Performance Specification SNF Parameters*	General and Radiological	TAD capacity of 21 pressurized water reactor or 44 boiling water reactor commercial SNF assemblies
		SNF limits of 5 wt% initial enrichment U-235

Table 2-4. Transportation, Aging, and Disposal (TAD) Canister Spent Nuclear Fuel (SNF) Evaluation Parameters (continued)		
Representative Cask Vendor Safety Analysis Report SNF Parameters		**SNF Assembly Parameters and Parameter Specifications**
		Less than 80 GWd/MTU burnup for pressurized water reactor fuel assemblies
		No less than 5 years out-of-reactor cooling time
	Thermal	Maximum commercial SNF cladding temperature not to exceed 400 °C [752 °F] during normal operations
		Maximum commercial SNF cladding temperature not to exceed 570 °C [1,058 °F] during draining, drying, and backfill operations
		Maximum commercial SNF cladding temperature not to exceed 350 °C [662 °F] when emplaced in the waste package
		TAD canister SNF cooling features and mechanisms shall be passive
*DOEDOE/RW–0585, "Transportation, Aging and Disposal Canister System Performance Specification." Rev. 0. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2007.		

2.4 NRC Lessons Learned on Spent Nuclear Fuel Verification and Qualification

NRC regulations in 10 CFR Parts 71 and 72 govern SNF transportation and storage, respectively. It is NRC practice to standardize the review and approval of applications that involve the development of SRPs. Each SRP summarizes the regulatory requirements necessary for application for approval and describes the staff procedures used to determine that the requirements have been satisfied. SRPs are guidance documents that are intended to be updated and revised as regulations or practices change. To help keep this guidance up to date, interim staff guidances (ISGs)[3] are used to identify emergent issues and develop staff positions in a timely manner.

In addition to the SRP and ISG documents, NRC staff may also issue regulatory issue summary (RIS)[4] documents to promulgate the experience gained and lessons learned through the review and approval of a number of license applications. The lessons learned are typically identified through evaluating and refining of staff processes and procedures during the review of multiple

[3]Interim staff guidance is referenced throughout this document. The abbreviation ISG will be used.
[4]Regulatory issue summary is referenced throughout this document. The abbreviation RIS will be used.

licensee applications and interactions. Using the RIS documents in this area, as well as selected requests for additional information (RAIs)[5] and safety evaluation reports, the following sections identify the important and recurring issues associated with current NRC staff practice to verify and qualify SNF.

The issues discussed are meant to shed light on issues to be considered in the TAD canister design review based on current NRC staff practice. Each of the following sections will begin with a summary of recurring RAI examples from RIS 2007-09 (NRC, 2007a). A number of RAI response documents with input into safety analysis reports such as those for the Humboldt Bay, Idaho, and Trojan independent spent fuel storage installations (ISFSI)[6] were also reviewed to identify any important issues the RIS had not captured as recurring issues in NRC (2007a).

2.4.1 Thermal Review Area

Recurring RAIs for 10 CFR Parts 71 and 72 applications in the thermal review area center around a lack of adequate information to assess compliance with the 10 CFR 72.236(b) requirements: "Design bases and design criteria must be provided for structures, systems, and components important to safety." NRC has repeatedly asked applicants for additional details on how the computational modeling was performed to meet this regulation. Sufficient details on how the analyses are performed should be provided in the application.

NRC staff has also addressed this recurring issue through the issuance of ISG–21 (NRC, 2006b). NRC practice in this area has resulted in recurring RAIs on the use of computational fluid dynamics (CFD)[7] methods. These RAIs have included asking applicants to provide additional information on the methodology for determining porous media flow-resistance parameters, based on the assumption of using CFD codes to model buoyancy-driven flows and on the use of convection correlations for modeling internal natural circulation (NRC, 2007a).

Also in the area of computational modeling, recurring RAIs have been generated for applicants to justify the uncertainties used in the calculations. These uncertainties generally relate to the computational method, input model, and assumptions the applicant used in the models.

2.4.2 Confinement Review Area

Recurring RAIs for 10 CFR Parts 71 and 72 applications in the confinement (or containment) review area have centered around four areas. These include (i) definition of damaged fuel, (ii) inappropriate use of the term *leak tight*, (iii) leakage-rate testing methodology, and (iv) testing of vent/drain port welds on cask lids. Most questions that recur for damaged fuel could be avoided through the appropriate use of ISG–1, Revision 2 (NRC, 2007b). ISG–1 provides guidance to applicants and staff on classifying SNF as damaged, undamaged, or intact and summarizes the fuel-specific regulations that provide the regulatory basis. Applicants can supersede or modify the guidance provided in ISG–1, but this must be supported with

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[5]Requests for additional information is referenced throughout this document. The abbreviation RAIs will be used.
[6]Independent spent fuel storage installations are referenced throughout this document. The abbreviation ISFSI will be used.
[7]Computational fluid dynamics is referenced throughout this document. The abbreviation CFD will be used.

appropriate technical arguments; the lack of these supporting arguments has resulted in recurring RAIs in this area.

Recurring RAIs have been generated due to a number of applicants' use of the term *leak tight* and leak-testing methodologies in the confinement section of the safety analysis report. ANSI Standard N14.5–1997 (American National Standards Institute, 1998) states that *leak tight* means that leak testing has been performed on the confinement boundary as a whole. The methodology described in ISG–18 (NRC, 2003a) provides justification that leakage from the confinement boundary is not credible, not that the boundary is *leak tight*. The statement that leakage is not credible in this guidance applies only to the final closure welds of austenitic stainless steel canisters. This would not include items such as the vent and drain port cover welds, which comprise a portion of a typical cask confinement system, and therefore, ISG–5, Revision 1 (NRC, 1999a), should be consulted for NRC staff practice on the confinement evaluation as a whole.

2.4.3 Criticality Safety Review Area

Recurring RAIs for 10 CFR Parts 71 and 72 applications in the criticality safety area occur in primarily two areas—computer modeling and margin to criticality credit taken for the boron neutron absorber (B-10) content. Issues in computer modeling typically involve staff requests for representative sample input files, description and justification of differences between modeling approaches, and code options between the design computations and those used to benchmark the code. Sample input files should be representative of different fuel geometry configurations, bounding (i.e., most reactive), and other configurations (e.g., as intact fuel, damaged fuel and fuel debris, and loose or consolidated rods) (NRC, 2007a). NRC staff practice has requested the calculation files so staff can appropriately review and directly compare the applicant's descriptions, justification, and assumptions of these calculations against the actual code calculation.

To address the neutron absorber boron content issue, current NRC staff practice requires the license application technical specifications to indicate the boron percentage credit, manufacturer's trade name, minimum areal density (B-10 content per unit area) required to be measured, and the volume percentage of boron carbide (B_4C) used in the metal matrix composites for each neutron absorber proposed in the application (NRC, 2007a). Applicants should provide neutron-attenuation testing and acceptance criterion for neutron-absorbing materials that take greater than 75 percent credit in the criticality safety analysis. For test methods other than neutron attenuation, applicants are required to benchmark the results of the proposed acceptance test method against the results of neutron attenuation testing.

An important item in the evaluation of one applicant's criticality safety analysis was the NRC staff request for the administrative procedures to be used in cask loading to prevent a misload. In this case, the applicant determined a misload was not a credible event due to the administrative procedures in place (Pacific Gas and Electric Company, 2004). This required an RAI to allow review of the procedures themselves.

2.4.4 Shielding Review Area

Recurring RAIs for 10 CFR Parts 71 and 72 applications in the shielding area are found in two main areas—fuel burnup determination and use of computer codes. Because fuel burnup affects the radiological source term used in shielding analyses, recurring RAIs typically have requested that information be provided on the peak average burnup per fuel rod and on how the source term for fuel with high burnup was determined. The primary guidance documents for current NRC staff practice to address these issues are ISG–8, Revision 2 (NRC, 2002), and ISG–11, Revision 3 (NRC, 2003b).

Recurring RAIs on the use of computer codes in the shielding area have typically been concentrated in the following specific areas: (i) additional information from the applicant on the verification and validation of any nonstandard codes used; (ii) additional information on modifications to standard codes, how the code was modified, and the process to validate the modified code; (iii) additional information to provide sample inputs of the design basis fuel used for the source term; (iv) additional information to provide the neutron energy spectrum used in the analyses; and (v) additional information to explain and justify that the analyzed case is bounding. NRC staff practice and guidance in this area is found in a number of SRPs and in detail in ISG–21 (NRC, 2006c). For example, in one case, an applicant was asked to provide a number of engineering documents that had been used and referenced to conduct the shielding calculations, but had not been provided in the safety analysis report for staff review.

3 DISCUSSION OF NUCLEAR, THERMAL, AND RADIOLOGICAL PARAMETERS OF SPENT NUCLEAR FUEL INTENDED FOR TRANSPORTATION, AGING, AND DISPOSAL CANISTER LOADING

3.1 Loading Curve Requirements for Nuclear Parameters: Misloads of Underburned Assemblies and Their Potential Neutronic Reactivity Effects

The spent nuclear fuel (SNF)[1] loading is expected to be performed in accordance with the specific loading curves. A loading curve depicts the relationship between the initial enrichment of a fuel assembly and the required minimum burnup needed to sufficiently suppress the reactivity of that fuel assembly so the assembly can be safely loaded into the canister, cask, or waste package. U.S. Department of Energy (DOE) will develop loading curves for each type of fuel and each type of transportation, aging, and disposal (TAD)[2] canister. The loading curve for a specific canister or a cask, therefore, requires that the fuel assembly for each initial uranium enrichment has a sufficient burnup to qualify for loading in the canister or the cask.

10 CFR Part 71.55(b) requires that the package used to ship fissile material must remain subcritical if hypothetical optimum moderation and reflection is provided for the most reactive credible configuration of the fissile material, consistent with the chemical and physical form of the material. Current U.S. Nuclear Regulatory Commission (NRC) staff practice considers a fissile material configuration "subcritical" if the calculated maximum k_{eff} (i.e., the upper limit of a two-sided 95 percent confidence interval for a calculated k_{eff}) does not exceed 0.95 (NRC, 2000). According to NRC (1997), the maximum allowable value of k_{eff} is called the upper subcritical limit. 10 CFR Part 63 does not contain an explicit requirement for such an upper subcritical limit.

DOE postclosure criticality analysis will determine a criticality limit, and the loading curve development involves a preclosure upper subcriticality limit. The DOE postclosure criticality analysis may not include an administrative margin for the postclosure criticality limit determination. Preclosure criticality analysis is expected to include an administrative margin for the preclosure upper subcriticality limit determination. An administrative margin in k_{eff} is intended to account for the unknown uncertainties in the criticality analyses, such as in neutron cross section values (NRC, 2006b). The postclosure criticality limit is higher than the preclosure subcriticality limit by approximately the value of the preclosure administrative margin, which may be between 0.02 and 0.05. According to DOE (2007), the calculated maximum k_{eff} for different fuel types and configurations must remain below the criticality and subcriticality limits for post- and preclosure applications, respectively. For postclosure criticality control, DOE (2007) proposed design bases or criticality analysis options for compliance by a potential canister manufacturer. For preclosure applications that involve a TAD canister loading and for each type of TAD canister, each point of the TAD canister loading curve determines minimum burnup for a certain initial enrichment of a fuel assembly. The calculated maximum k_{eff} of such configuration

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[1]Spent nuclear fuel is referenced frequently throughout this document. The abbreviation SNF will be used.
[2]Transportation, aging, and disposal is referenced frequently throughout this document. The abbreviation TAD will be used.

must be below the preclosure subcriticality limit for this TAD canister, which must (i) be loaded with the maximum allowable number of the identical fuel assemblies having the initial enrichment and burnup determined by the loading curve, (ii) have a cooling time of 5 years, (iii) be filled with water having optimum neutron moderation properties, and (iv) be reflected by water, according to DOE (2007). Based on these analyses, DOE generates several design basis loading curves for each type of TAD canister and for different SNF types. Loading curve examples are presented in DOE (2007), Sandia National Laboratories (2007b), and Bechtel SAIC Company LLC (2004a). DOE uses 29 principal isotopes in the loading curve development analyses: 14 actinides and 15 fission products (CRWMS M&O, 2003). The increase in neutronic reactivity of the fuel during the postclosure period results from the decay of some neutron-absorbing fission products and will be accounted for in the loading curve determination. The misload of one or more underburned fuel assemblies into a canister during loading operations would increase the system neutronic reactivity, and in the presence of a moderator, the calculated k_{eff} may exceed the upper subcriticality limit for preclosure and/or the criticality limit for postclosure. The underburned assemblies in this context are those that do not meet requirements of the specific design basis loading curve (i.e., those that have a higher fissile material content than the loading curve allows). When k_{eff} exceeds the preclosure upper subcriticality limit, the margin of safety is reduced to below an acceptable level.

If underburned or fresh assemblies are present in the SNF pool at the loading site, one or more of these assemblies could be misloaded into a TAD canister. An assembly misload could be caused by an error in the assembly reactor records, misapplied reactor records, an error in the assembly identification, or human error in assembly identification and/or handling. Neutronic reactivity effects of single or multiple misloads are investigated in different studies (Bechtel SAIC Company, LLC, 2004a; Electric Power Research Institute, 2003; NRC, 2008). DOE estimates that if an assembly with an initial enrichment of 3.5 percent and burnup of 4 GWd/MTU is loaded into the waste package instead of the required 24 GWd/MTU burnup, the k_{eff} of the system would increase from 0.92808 to 0.95487, corresponding to a Δk_{eff} of about 0.02679 (Bechtel SAIC Company, LLC, 2004a). Misloading of underburned assemblies, therefore, could potentially exceed the upper subcriticality limit of the loaded waste package.

According to NRC (2008), which investigated reactivity effects of pressurized water reactor assembly misloading in a GBC–32 cask, misloading of a single fresh assembly with 3, 4, or 5 weight percent of U-235 initial enrichment results in a k_{eff} increase of ~0.02, 0.04, or 0.06, respectively. Misloading of two assemblies that are underburned by 75 percent results in a k_{eff} increase of 0.02–0.035 depending on the initial enrichment. Misloading of four assemblies that are underburned by 50 percent results in an increase in k_{eff} of 0.02–0.035 for different enrichments of the fuel. These results were obtained for 29 principal actinides and fission products that are slightly different from the set used in the DOE study (Bechtel SAIC Company, LLC, 2004a). Electric Power Research Institute (2003) reports that if a single fresh assembly with U-235 enrichment of 5.0 weight percent is misloaded into a conceptual 24 pressurized water reactor assembly cask that requires 45 GWd/MTU burnup, the Δk_{eff} is at ~0.061. If an underburned single assembly of 4.0 weight percent of initial U-235 enrichment and 15 GWd/MTU burnup is misloaded into a GBC–32 cask that requires 35 GWd/MTU burnup, the Δk_{eff} is at ~0.02. If an assembly with 5.0 weight percent of initial U-235 enrichment and 25 GWd/MTU burnup is misloaded into the same cask that requires 45 GWd/MTU burnup, the Δk_{eff} is at ~0.012. These results were obtained for a loading curve developed for the conceptual cask with five unidentified fission products (Electric Power Research Institute, 2003). This set of isotopes is likely to differ from the sets in both the DOE (Bechtel SAIC Company, LLC, 2004a) and NRC (2008) staff studies. DOE (Bechtel SAIC Company LLC, 2004a), NRC (2008), and Electric Power Research Institute (2003) analyses assume the use of neutron absorber plates and the presence of fresh water inside the casks. None of these analyses involve a TAD canister placed inside a waste package.

To gauge the effect of misloading underburned or fresh assemblies on neutronic reactivity of the system involving the TAD canister inside the waste package, the Center for Nuclear Waste Regulatory Analyses (CNWRA) conducted preliminary scoping calculations using example loading curves (Sandia National Laboratories, 2007b; DOE, 2007; Bechtel SAIC Company, LLC, 2004a) and a conceptual TAD canister design (DOE, 2007). The results are presented in Figure 3-1 (Povetko, 2008).

Figure 3-1 shows the effect on Δk_{eff} of a single misloaded assembly for a conceptual TAD canister loaded with 20 identical assemblies in accordance with a corresponding loading curve and a single underburned assembly (i.e., in violation of the loading curve requirement) that is placed in one of the central locations of the canister. Three curves correspond to minimum burnups of 20, 30, and 40 GWd/MTU and corresponding enrichments. The analyses are performed for the assumed 0, 20, 50, 80, and 100 percent of minimum burnup required by the loading curve. The 5 data points on each of 3 curves correspond to 0, 20, 50, 80, and 100 percent of minimum required burnup (i.e., degree of the underburn of the single central

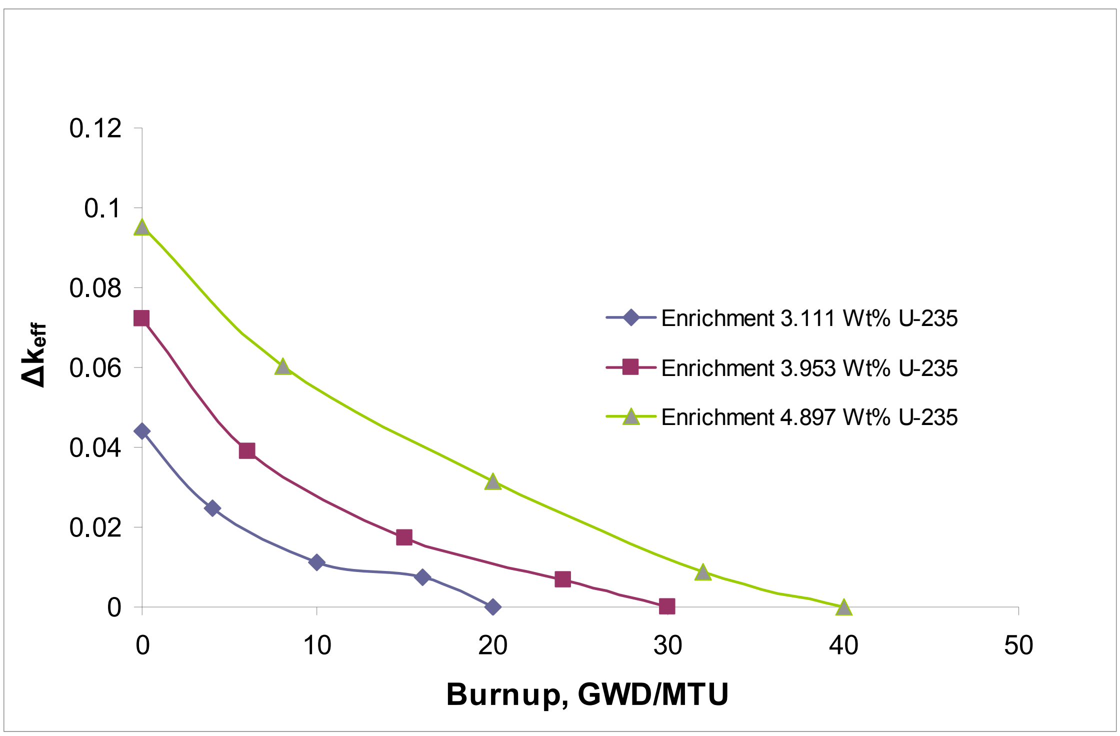

Figure 3-1. The Effect on k_{eff} of a Single Misloaded Assembly as a Function of the Minimum Burnup Required by the Loading Curve (Sandia National Laboratories, 2007b; DOE, 2007; Bechtel SAIC Company, LLC, 2004a)

assembly). These preliminary results for a conceptual TAD canister correlate well with Oak Ridge National Laboratory results for GBC–32 (NRC, 2008), DOE results for uncanistered fuel in a waste package cask (Bechtel SAIC Company, LLC, 2004a), and Electric Power Research Institute results (2003) for a conceptual 24 pressurized water reactor assembly cask discussed previously. The designs, configurations, materials, neutron absorbers, and sets of isotopes are different in each of these studies.

Because the values of Δk_{eff} caused by the probability of misloading fresh and/or underburned assemblies are relatively high, DOE has investigated the probability of assembly misloading. DOE indicates that a waste package misload is a credible preclosure event (Bechtel SAIC Company, LLC, 2005) and estimates the probability of one misload of an underburned assembly over the entire preclosure period as 0.23 with independent checking of the assembly data as an operational control (Bechtel SAIC Company, LLC, 2004b). DOE also reports the overall probability of assembly misload at 2.7×10^{-4} per assembly movement (CRWMS M&O, 2001). The estimated number of SNF assembly movements used for this calculation is 1,199,000, and the number of misloads is 327. This estimated number of moved spent fuel assemblies is based on the 1985–1999 study period. For pre-TAD system design, when the entire inventory of SNF was planned to be loaded at the repository surface facilities, DOE estimates the probability of misloading of an underburned pressurized water reactor assembly into a 21 pressurized water reactor absorber plate waste package as 1.18×10^{-5} per waste package and the probability of an underburned boiling water reactor assembly misloaded into a 44 boiling water reactor absorber plate waste package as 1.73×10^{-5} per waste package (Bechtel SAIC Company, LLC, 2005). Electric Power Research Institute (2006) estimates the overall likelihood of a misloaded SNF cask as 6×10^{-7} per cask and discusses actions to further reduce the likelihood of assembly misloading.

A number of fuel misloads were documented in the NRC Licensee Event Report database from 1980–2006. The review of these data is outside the scope of this report, but the incidence of misloading events is expected to be on the same order of magnitude per fuel handling activity as in the analyses discussed previously. DOE (Bechtel SAIC Company, LLC, 2004c) estimated the probabilities of the neutron absorber misload, but this topic is outside of the scope of this report.

Once a TAD canister is loaded and sealed at the loading site, it will not be opened again and fuel will not be reloaded through permanent disposal in the repository. About 90 percent of assemblies are expected to be loaded into TAD canisters at the utility sites. The fuel characteristics relied on in TAD canister loading curves are, therefore, the important parameters requiring preloading verification. These parameters include fuel initial enrichment, burnup, and cooling time. The American National Standards Institute/American Nuclear Society (2004) recommends determining assembly burnup by either burnup measurements or analysis of power history. U.S. Nuclear Regulatory Commission (NRC) staff (1998) takes exception to this recommendation and instead recommends taking credit for fuel burnup only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored. DOE suggests that burnup measurements are unnecessary and may rely solely on reactor records for verification of burnup values. DOE does not plan to verify these records by assembly burnup measurements (Bechtel SAIC Company, LLC, 2004b). The utilities may quantify assembly burnup and burnup record uncertainty and account for this uncertainty prior to TAD canister loading (CRWMS M&O, 1998).

Electric Power Research Institute (1999) compares the reaction rates measured by moveable in-core instruments during reactor operations and later converted into average assembly burnup values with the results of average individual assembly burnup calculations for an unidentified Westinghouse pressurized water reactor. The differences between these two values (measured converted and calculated) are defined as burnup uncertainties. The study concludes that in the fuel assemblies in instrumented locations in a first cycle, the burnup uncertainty is 2.49 percent; for assemblies in instrumented locations in the second cycle, the burnup uncertainty is 1.67 percent; and for assemblies in instrumented locations in the third cycle, it is 1.99 percent. The study warns that the data used in the comparison represent current reactor software technology and may not represent older or future technologies. The study does not address batch average assembly measurements, differences in methods that different utilities routinely use to calculate assembly burnup, and older instrumental technologies used for past in-core measurements. The Oak Ridge National Laboratory prepared a report on SNF burnup verification and measurements prior to loading, tentatively titled "Review of Information for Spent Nuclear Fuel Burnup Confirmation." The report was not available for examination at the time of this survey.

DOE may rely on the performance of neutron absorber plates as criticality controls during pre- and postclosure periods. Borated stainless steel SS304B4 and Ni-Gd alloy may be used as neutron absorber materials for commercial and DOE SNF, respectively. If the TAD canister manufacturer misses the plate or places a plate with insufficient or heterogeneous distribution of neutron-absorbing isotopes, the system neutronic reactivity will increase and the calculated k_{eff} may exceed the preclosure upper subcriticality limit and/or postclosure criticality limit. The verification of neutron absorber plates is therefore important during loading. However, the issue is outside the scope of this study because the manufacturer(s) will install the neutron absorber plates as a part of the TAD canister system.

DOE indicates that it might classify the TAD canister as important to safety for providing for moderator control as the primary criticality safety control and for containing radionuclides during event sequences (Bechtel SAIC Company, 2007b). The moderator control function of the TAD canister is an important issue to consider for TAD canister loading but is outside the scope of this study.

3.2 Thermal Parameters

Repository preclosure safety analysis and postclosure performance assessment rely on certain fuel and waste package characteristics. If these characteristics are not met during TAD canister loading, the results of the DOE analyses may not be accurate and, consequently, the safety case may be compromised. For example, if the total waste package thermal output exceeds the design basis value, it may negatively affect the long-term capability of the engineered and natural barriers beyond the range of their capabilities accounted for by DOE postclosure performance assessment. Numerical design basis limits for thermal parameters are not included in 10 CFR Part 63; therefore, DOE derived these design basis limits based on 10 CFR Part 63 performance objectives (CRWMS M&O, 2000). As CRWMS M&O (2000) initially developed and recently reiterated by DOE (2007), peak cladding temperature should remain below 400 °C [752 °F] under normal conditions and 350 °C [662 °F] for certain specified thermal conditions outlined in Table 3-1. To satisfy this peak cladding temperature design c repository thermal design strategy for pre-TAD design assumes an emplacement linear thermal power of 1.45 kW/m, a waste package spacing of 0.1 m [0.33 ft], and that the 21 pressurized riterion, the

Table 3-1. Thermal Conditions for Cladding Temperature Determination	
Thermal Output (kW)	**Canister Surface Temperature Boundary Conditions °C [°F]**
11.8	274 [525]
18	232 [450]
25	181 [358]

water reactor absorber plate waste package thermal output is below 11.8 kW (CRWMS M&O, 2000). These criteria may not change significantly in a new repository design involving the TAD system. The total waste package thermal output limit of 11.8 kW allows flexibility in the assembly loading pattern into the TAD canister. These design basis limits would not contradict the thermal requirements of the current NRC staff practice under 10 CFR Parts 71 and 72—specifically, the cladding limits of 380 °C [716 °F] for 5-year-old fuel and 570 °C [1,058 °F] for short-term accidents of fuel transfer (NRC, 2006a). The thermal and criticality requirements, however, may present conflicting conservatisms for loading patterns. While loading of a relatively young and higher burnup assembly would satisfy the criticality loading curve, it might potentially cause a local increase in the cladding temperature.

3.3 Radiological and Shielding Parameters

DOE (2007) indicates that the contact dose rate at any point on the top surface of the TAD canister does not exceed 10 mSv/hr [1,000 mrem/hr] and average dose rate over the top surface remains below 8 mSv/hr [800 mrem/hr]. The majority of TAD canisters are expected to be loaded with 21-pressurized water reactor or 44-boiling water reactor assemblies; therefore, these integrated specifications allow loading flexibility of assemblies with different external radiation intensities and spectra. These specifications are pertinent to the external TAD canister surface condition, not to the source term of the loaded waste. While loading of a relatively young and higher burnup assembly would satisfy the criticality loading curve, it might affect the peak dose rate for the top surface of the TAD canister. If the TAD canister shielding analyses are not conducted for the maximum SNF (i.e., fuel having bounding direct dose rates), then the dose effect of preferential loading of high dose rate assemblies should be investigated further.

4 DETERMINATION OF HEAT OUTPUT PARAMETERS FOR COMMERCIAL SPENT NUCLEAR FUEL EXPECTED TO BE LOADED INTO TRANSPORTATION, AGING, AND DISPOSAL CANISTERS

4.1 Purpose

This calculation determines the heat output generated during the radioactive decay of discharged fuel from light water nuclear power reactors for a wide range of spent nuclear fuel (SNF)[1] parameters. The range of parameters extends beyond those available in the literature and guidance (NRC, 1998, 1994) because there are indications that utilities continue increasing fuel utilization. The calculation output supports the transportation, aging, and disposal (TAD)[2] canister thermal evaluation.

4.2 Methodology

The method used to evaluate the SNF heat output involves simulating of the burnup and decay for fuel assemblies. The calculation methodology is as follows:

(1) Review published media that pertain to determining inventory and heat output of light water reactor SNF.

(2) Identify the types and materials of fuel elements that would produce a representative heat output following irradiation in the reactor.

(3) Identify reactor and reactor campaign parameters that would produce a representative heat output.

(4) Develop a reactor and fuel campaign model based on the selected fuel type and reactor campaign.

(5) Select representative ranges of the SNF parameters and breakdown points in the ranges for depletion computations.

(6) Conduct a series of the point-depletion and decay computations to calculate radionuclide inventory and decay heat output rate by the end of the fuel cooling period. These computations are performed for all combinations of discrete parameter values in Table 4-1.

[1]Spent nuclear fuel is referenced frequently throughout this document. The abbreviation SNF will be used.
[2]Transportation, aging, and disposal is referenced frequently throughout this document. The abbreviation TAD will be used.

Table 4-1. Description of Modeled Pressurized Water Reactor Spent Nuclear Fuel Assembly and Its Power History. In the Calculation, These Parameters Assume Multiple Discrete Values. Results for High Burnup Values 50, 60, 70, and 78.26 GWd/MTU Are Presented for Illustration Purpose Only (Values Are Italized).	
Parameter	**Value**
Initial Enrichment, Weight Percent of U	1.5, 2.0, 2.5, 3.0, 3.5, 4.0, 4.5, 5.0, 5.5
Burnup, GWd/MTU	10, 20, 30, 40, *50, 60, 70, 78.26*
Cooling Time, Years	5, 15, 25
Specific Power, kWt/kgU	28, 31, 29, 40, 60, 80

4.3 Use of Computer Software and Models

The Center for Nuclear Waste Regulatory Analyses (CNWRA) used the SCALE/ORIGEN Version 5.1 code system (Oak Ridge National Laboratory, 2006a) to calculate radionuclide inventory and radiation heat output rate corresponding to this inventory for the light water reactor SNF. The software was qualified in accordance with the Geosciences and Engineering Division Technical Operating Procedure–018 Development and Control of Scientific and Engineering Software.

The ORIGEN-ARP sequence is a part of the SCALE/ORIGEN code system. ORIGEN-ARP performs point-depletion calculations with the ORIGEN-S code using problem-dependent cross sections. Problem-dependent cross section libraries are generated by the automatic rapid processing module using an interpolation algorithm that operates on pregenerated libraries created for a range of fuel properties and operating conditions. The sequence interpolates cross sections based on enrichment, burnup, and optionally, moderator density involving a set of standard basic cross section libraries for light water reactor and mixed oxide fuel assembly designs. The interpolated cross sections are passed to ORIGEN-S. The ORIGEN-ARP was extensively validated; the validation reports are available from Oak Ridge National Laboratory (2008). Table 4-1 lists a burnup range of 10–78.26 GWd/MTU. The ORIGEN-ARP module contains cross-section libraries for this burnup range. According to the code developers, however, the validation of ORIGEN-ARP is based primarily on the validation of the TRITON/NEWT SCALE module, which is used to generate the cross section libraries (Bowman, 2008). The highest burnup for which these cross section libraries are validated is 47 GWd/MTU (Sanders, 2003); therefore, in Table 4-1 and in Appendix 1, the results corresponding to 50, 60, 70, and 78.26 GWd/MTU are presented for illustration purposes only and are italicized to distinguish these values from those for which the current validation of the cross section libraries currently extends.

The software is appropriate for this calculation and is used within its range as described in the software documentation.

4.4 Assumptions

These analyses are based on several assumptions.

- Pressurized water reactor SNF is assumed to be more representative of heat output than boiling water reactor SNF, and this calculation is performed for pressurized water reactor fuel only.

  Rationale: The values for heat output that Oak Ridge National Laboratory (NRC,1994) and NRC (1999b) reported are generally higher for pressurized water reactor SNF assemblies than for boiling water reactor SNF assemblies. According to DOE projections (Bechtel SAIC Company, LLC, 2004d), pressurized water reactor SNF will be loaded into 4,557 waste packages, whereas boiling water reactor SNF will be in 2,915 waste packages.

- The 15 × 15 Babcock & Wilcox Mark B pressurized water reactor fuel assembly (Bechtel SAIC Company, LLC, 2004e) is chosen for the depletion calculation.

  Rationale: DOE (Bechtel SAIC Company, LLC, 2004e) found that the selection of a particular assembly is not sensitive to the resulting radionuclide inventory. The 15 × 15 Babcock & Wilcox Mark B pressurized water reactor fuel assembly contains high initial heavy metal content and a large amount of hardware and therefore generates more fission and activation products than other fuel assemblies. This assembly provides a conservative basis for the pressurized water reactor waste high heat output for the pressurized water reactor waste stream.

- One power cycle is assumed for irradiation of assemblies. Nonstop, steady-state power is assumed during irradiation until desired burnup is reached.

  Rationale: The power histories of assemblies can be very complicated. The simple one-cycle approach is used followed by a sensitivity analysis investigating the effect of the elevated specific power on the heat output rate. Such a one-cycle reactor operation would conservatively (i.e., overestimate) estimate decay heat. Sterbentz (1997) indicates that the duration of burnup has little effect on mass, activity concentrations of radionuclides, and decay heat after approximately 3 years of decay.

- The 15 × 15 ORIGEN-ARP library pregenerated for the Westinghouse 15 × 15 assembly and provided with the software is used in the depletion calculation.

  Rationale: The reliability of the ORIGEN-ARP results depends mainly on the appropriateness of the neutron cross section libraries used in the calculation. The burnup-dependent cross sections depend highly on the core neutron spectrum. The spectrum is a dynamic function of the reactor features including the uranium loadings, fuel enrichment, moderator and cladding materials, lattice configuration and pitch, in-core absorbers, and operational temperature. SCALE developers have pregenerated cross sections for the pressurized water reactor fuel using the two-dimensional lattice physics code NEWT as applied in the TRITON depletion analysis module of the SCALE system (Oak Ridge National Laboratory, 2006b). The parameters of the various assembly designs used for these pregenerated libraries are analyzed, and the 15 × 15 library based on the Westinghouse 15 × 15 assembly is selected as the closest match to the 15 × 15 Babcock & Wilcox Mark B assembly design (Bechtel SAIC Company, LLC, 2004e) because 15 × 15 Babcock & Wilcox Mark B assembly libraries are not included in the ORIGEN-ARP set of libraries.

4.5 Calculation

The calculation consists of determining the isotopic composition of a pressurized water reactor assembly after a continuous full-power reactor campaign and cooling period, which consequently enables determination of the heat output rate. The single value fuel and assembly power history parameters used in the calculation are presented in Table 4-2 and multiple discrete value parameters in Table 4-1.

4.6 Results

The per-assembly heat output rates are presented in Appendix A. The output data in the tables are interpolated and presented as graphical maps for better visualization. Note that color scales are different for each graphical map. The color scales span from minimum to maximum values for the particular set of cases in the corresponding table. These minimum and maximum values are shown at the bottom of each table as P_{min} and P_{max}, respectively.

4.7 Sensitivity Analysis

The assembly heat generation rate depends on the specific power during assembly irradiation. To investigate the scope of this effect, the value of 28 kW/kgU is a selected basecase specific power. The heat generation rates are calculated for 4 specific elevated power levels (31.3, 40, 60, and 80 kW/kgU), and the ratios of the elevated heat generation rates to the basecase rate are tabulated in Appendix B. The values between data points are interpolated, and Appendix B figures visually present these interpolated data as colored maps.

Table 4-2. Description of Modeled Pressurized Spent Nuclear Fuel Assembly and Its Power History (Single Value Parameters)

Parameter	Value*
Assembly Type	15 × 15 Babcock & Wilcox Mark B
Beginning of Life Uranium Content	463.63 kg [1,120 lb]
Moderator Temperature	579 °K* [582 °F]
Density of the Moderator	0.7136 tonne/m^3 [2.578 × 10^{-2} lb/in^3]
Average Boron Concentration	553 ppm

*Bechtel SAIC Company, LLC. "PWR Source Term Generation and Evaluation." 000–00C–MGRO–00100–000–00B. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2004.

5 SUMMARY AND RECOMMENDATIONS

The misloads of fresh and/or underburned assemblies into transportation, aging, and disposal (TAD)[1] canisters are expected to be credible events in the U.S. Department of Energy (DOE) geologic repository preclosure safety analyses. The reactivity effect of a misload of a single underburned assembly may reach a large percentage of k_{eff}.

Based on the literature survey of the spent nuclear fuel (SNF)[2] parameters, the burnup and cooling time have a large effect on dose rates (NRC, 2001a).

The following are the most important parameters that influence criticality safety and radiation doses:

- Fuel Type
  - Array size; number of fuel rods, including the number of partial length rods (where applicable); and cladding type
  - Number and material of guide and instrument tubes
- Enrichment (maximum for criticality safety, minimum for radiation shielding)
- Maximum burnup
- Minimum cooling time
- Maximum uranium mass
- Maximum cobalt level

Other criticality safety parameters (e.g., pitch, pellet outer diameter, clad thickness, clad outer diameter, guide tube and water rod thickness, and fuel stack density) and radiation shielding parameters (e.g., maximum poison loading, minimum boiling water reactor moderator density, and maximum specific power) are less important (NRC, 2001a,b).

Recurring requests for additional information (RAIs)[3] for 10 CFR Parts 71 and 72 applications in the thermal review area center on the computational modeling; specifically, on the use of computational fluid dynamics methods. Recurring RAIs in the confinement review area include definition of damaged fuel, inappropriate use of the term *leak tight*, leakage-rate testing methodology, and testing of vent/drain port welds on cask lids. Recurring RAIs in the criticality safety area occur in computer modeling and margin to criticality credit taken for the boron neutron absorber content. Recurring RAIs in the shielding review area include fuel burnup and use of computer codes.

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[1]Transportation, aging, and disposal is referenced frequently throughout this document. The abbreviation TAD will be used.
[2]Spent nuclear fuel is referenced frequently throughout this document. The abbreviation SNF will be used.
[3]Requests for additional information is referenced frequently throughout this document. The abbreviation RAI will be used.

Because there are indications that utilities continue to increase fuel utilization, this study presents assembly heat output rates for parameter ranges extending beyond those available in the literature and guidance (NRC, 1998, 1994).

To ensure compliance with 10 CFR Part 63 requirements, DOE has indicated that they will rely on the nuclear power plant operators to perform offsite SNF assembly loading according to the TAD canister system performance specifications. After they are loaded at the loading sites and shipped to the disposal site, canisters will be inspected but are not expected to be opened. Such parameters like canister gas composition, canister surface dose rate, and thermal output can be easily measured without opening the canisters using external instrumentation. In case of noncompliance with the quality assurance requirements upon receipt, the remediation could be achieved by aging the canister without opening it at the aging facility or, if necessary, by reloading the canister content in the pool facility. In contrast, the burnup and initial enrichment of the loaded assemblies cannot be verified or measured without opening the canisters. These two parameters must comply with the loading curve values to ensure overall compliance with 10 CFR Part 63 requirements. DOE is not planning to perform any assembly burnup measurements upon receipt of the SNF and is expected to rely solely on reactor records to determine burnup and initial enrichments of the assemblies that would number approximately 221,000 (DOE, 2008, Section 1.5.1). Therefore, burnup and initial enrichment are more risk significant than the other SNF parameters. Special attention should be placed on determining these parameters, the parameter uncertainties, pertinent quality assurance programs, loading protocols, and case frequencies of the recorded assembly misloadings.

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7 GLOSSARY

burnup: A measure of nuclear reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission or as the amount of energy produced per unit weight of fuel.

burnable poisons: Materials found in fuel assemblies that absorb neutrons and are depleted (burned) in the process.

burnup credit: An approach used in criticality evaluations that accounts for the reduction in criticality potential associated with spent nuclear fuel relative to that of fresh fuel. Burnup credit reflects the net depletion of fissionable isotopes and the creation of neutron absorbing isotopes during reactor operations. Burnup credit also accounts for variations in the criticality potential of spent nuclear fuel produced by radioactive decay because the fuel was discharged from a reactor. For geologic disposal, burnup credit [if accepted by the U.S. Nuclear Regulatory Commission (NRC)] will account for the reduction in reactivity associated with 29 isotopes (principal isotopes) from commercial light water reactor spent nuclear fuel. This credit applies specifically to the ceramic form of commercial spent nuclear form.

canister: A cylindrical metal receptacle that facilitates handling, transportation, storage, and/or disposal of high-level radioactive waste. It may serve as (i) a pour mold and container for vitrified high-level radioactive waste or (ii) a container for loose or damaged fuel rods, nonfuel components and assemblies, and other debris containing radionuclides.

cask: A container for shipping or storing spent nuclear fuel and/or high-level waste that meets all applicable regulatory requirements.

chain reaction: A continuing series of nuclear fission events. Neutrons produced by a split nucleus collide with and split other nuclei causing a chain of fission events.

cladding: The metal outer sheath of a fuel rod generally made of a zirconium alloy, and in the early nuclear power reactors of stainless steel, intended to protect the uranium dioxide pellets, which are the nuclear fuel, from dissolution by exposure to high temperature water under operating conditions in a reactor.

commercial spent nuclear fuel: Nuclear fuel rods, forming a fuel assembly, that have been removed from a nuclear power plant after reaching the specified burnup.

critical event: See criticality.

criticality: (i) A condition that would require the original waste form, which is part of the waste package, to be exposed to degradation, followed by conditions that would allow concentration of sufficient nuclear fuel, the presence of neutron moderators, the absence of neutron absorbers, and favorable geometry; (ii) The condition in which a fissile material sustains a chain reaction. It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle. The state is considered critical when a self-sustaining nuclear chain reaction is ongoing.

criticality accident: The release of energy as a result of accidental production of a self-sustaining or divergent neutron chain reaction.

critical limit: A limiting value of k_{eff} at which a configuration is considered potentially critical, as characterized by statistical tolerance limits. Criticality analysis is a mathematical estimate, usually performed with computer software, of the neutron multiplication factor of a system or configuration that contains material capable of undergoing a self-sustaining chain reaction.

criticality control: The suite of measures taken to control the occurrence of self-sustaining nuclear chain reactions in fissionable materials, including spent fuel. For postclosure disposal applications, criticality control is ensuring that the probability of a criticality event is so small that the occurrence is unlikely and the risk that any criticality will violate repository performance objectives is negligible.

disposal: The emplacement of radioactive waste in a geologic repository with the intent of leaving it there permanently.

disposal container: A cylindrical metal receptacle designed to contain spent nuclear fuel and high-level radioactive waste that will become an integral part of the waste package when loaded with spent nuclear fuel or high-level radioactive waste. In a waste package, the inner container will have spacing structures or baskets to maintain fuel assemblies, shielding components, and neutron-absorbing materials in position to control the possibility of criticality.

enrichment: The act of increasing the concentration of fissile isotopes from their value in natural uranium. The enrichment (typically reported in atom percent) is a characteristic of nuclear fuel.

events: (i) Occurrences that have a specific starting time and, usually, a duration shorter than the time being simulated in a model; (ii) Uncertain occurrences that take place within a short time relative to the timeframe of the model. For the purposes of screening features, events, and processes for the total system performance assessment, an event is defined to be a natural or human-caused phenomenon that can potentially affect disposal system performance and that occurs during an interval that is short compared with the period of performance.

features: Physical, chemical, thermal, or temporal characteristics of the site or potential repository system. For the purposes of screening features, events, and processes for the total system performance assessment, a feature is defined to be an object, structure, or condition that can potentially affect disposal system performance.

fissile materials: Materials that will fission with slow neutrons (e.g., U-235, Pu-239).

fissionable materials: Materials that will fission if neutrons have enough energy. Note all fissile materials are fissionable, but not all fissionable materials are fissile.

frequency: The number of occurrences of an observed or predicted event during a specific time period, or the annual probability of occurrence of an initiating event or an event sequence.

geologic repository: A system that is intended to be used for, or may be used for, the disposal of radioactive wastes in excavated geologic media. A geologic repository includes the engineered barrier system and the portion of the geologic setting that isolates the radioactive waste.

half-life: The time required for a radioactive substance to lose its activity due to radioactive decay. At the end of one half-life, 50 percent of the original radioactive material has decayed.

k_{eff}: Effective neutron multiplication factor for a system. It provides a measure of criticality potential for a system ($k_{eff} \geq 1.0$ for criticality).

loading curve: The relationship between the required minimum burnup and fuel assembly initial enrichment.

nuclear criticality safety: Protection against the consequences of a criticality accident, preferably by prevention of the accident.

probabilistic: (i) Based on or subject to probability; (ii) Involving a variant, such as temperature or porosity. At each instance of time, the variant may take on any of the values of a specified set with a certain probability. Data from a probabilistic process are an ordered set of observations, each of which is one item from a probability distribution. Statistical probability examines actual events and can be verified by observation or sampling. Knowledge of the exact probability of an event is usually limited by the inability to know, or compile, the complete set of possible outcomes over time or space; a degree of belief.

processes: Phenomena and activities that have gradual, continuous interactions with the system being modeled. For the purposes of screening features, events, and processes for the total system performance assessment, a process is defined as a natural or human-caused phenomenon that can potentially affect disposal system performance and that operates during all or a significant part of the period of performance.

preclosure: A period of time before permanent closure of the geologic repository.

postclosure: A period of 10,000 years after the permanent closure of the geologic repository.

reactivity: Relative deviation of the neutron multiplication factor of the system from unity (i.e., reactivity = $(k_{eff}-1)/k_{eff}$).

repository: An NRC-licensed system that is intended to be used for, or may be used for, the permanent deep geologic disposal of high-level radioactive waste and spent nuclear fuel. Such a term includes both surface and subsurface areas at which high-level radioactive waste and spent nuclear fuel handling activities are conducted.

risk: The probability that an undesirable event will occur, multiplied by the consequences of the undesirable event.

spent nuclear fuel: Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. Spent fuel that has been burned (irradiated) in a reactor to the extent that it no longer efficiently contributes to a nuclear chain reaction. This fuel is more radioactive than it was before irradiation and releases significant amounts of heat from the decay of its fission product radionuclide. See burnup.

subcritical limit: The value that the calculated k_{eff} for a system/configuration of fissionable material must be shown to be below to be considered subcritical. The subcritical limit is dependent upon the computer system being used to calculate k_{eff}, the configuration being evaluated, and the regulatory margins specified for the application.

uncertainty: How much a calculated or measured value varies from the unknown true value.

waste package: The waste form and any containers, shielding, packing, and other absorbent materials immediately surrounding an individual waste container.

APPENDIX A

THE PER-ASSEMBLY HEAT OUTPUT RATES

Table A–1. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 28 kW/kgU, and Cooling Time of 5 Years*

Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*2,787*	*2,697*	*2,608*	*2,523*	*2,442*	*2,366*	*2,295*	*2,230*	*2,170*
70	*2,404*	*2,318*	*2,235*	*2,157*	*2,084*	*2,017*	*1,957*	*1,903*	*1,854*
60	*1,959*	*1,883*	*1,812*	*1,748*	*1,690*	*1,638*	*1,593*	*1,553*	*1,518*
50	*1,522*	*1,460*	*1,404*	*1,356*	*1,314*	*1,278*	*1,248*	*1,222*	*1,201*
40	1,127	1,083	1,045	1,014	988	967	950	936	923
30	782	758	739	724	711	701	692	685	679
20	491	480	471	465	459	455	451	447	444
10	238	234	231	229	227	226	225	224	223

*P_{min} = 223 watts P_{max} = 2,787 watts
Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

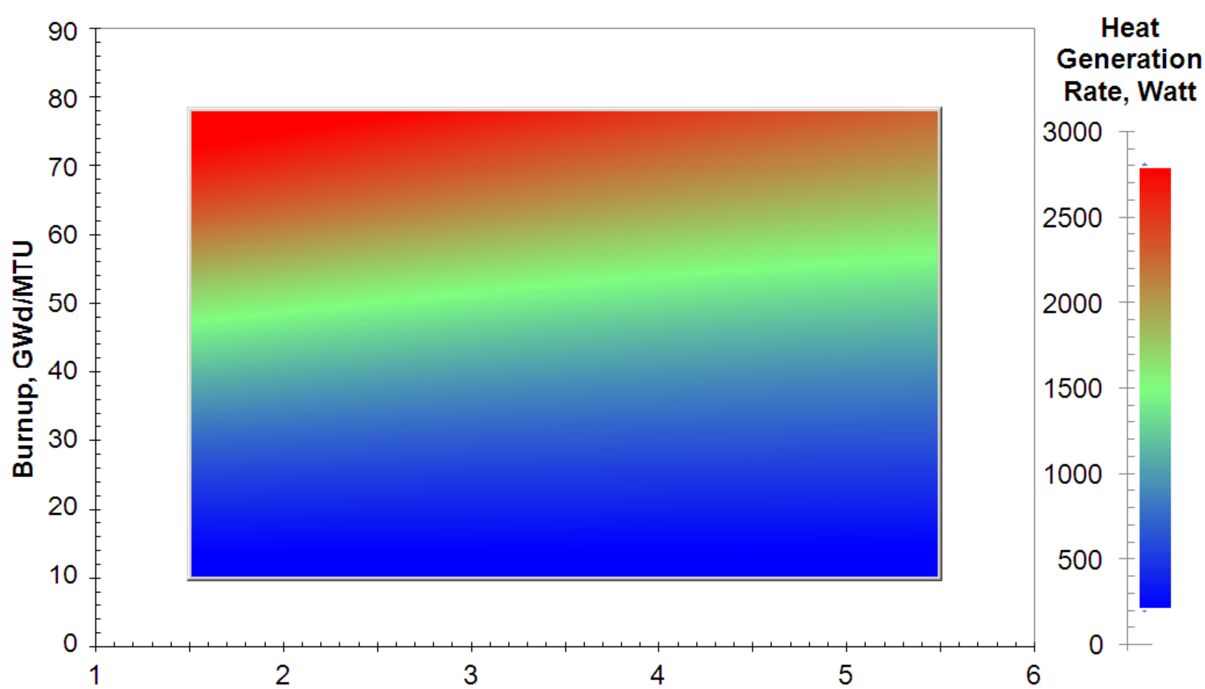

Figure A–1. Heat Generation Rates Thermal Map for Cooling Time of 5 Years and Assembly-Specific Power of 28 kW/kgU

Table A–2. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 28 kW/kgU, and Cooling Time of 15 Years*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,640*	*1,587*	*1,535*	*1,487*	*1,443*	*1,403*	*1,368*	*1,336*	*1,308*
70	*1,397*	*1,348*	*1,302*	*1,260*	*1,222*	*1,189*	*1,161*	*1,137*	*1,116*
60	*1,117*	*1,075*	*1,039*	*1,007*	*980*	*958*	*939*	*923*	*910*
50	*852*	*822*	*796*	*776*	*759*	*746*	*736*	*728*	*721*
40	618	600	587	577	570	564	560	557	554
30	422	417	414	412	411	409	408	407	406
20	265	266	266	266	265	265	264	264	263
10	130	130	130	130	129	129	129	129	129

*P_{min} = 129 watts P_{max} = 1,684 watts
Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

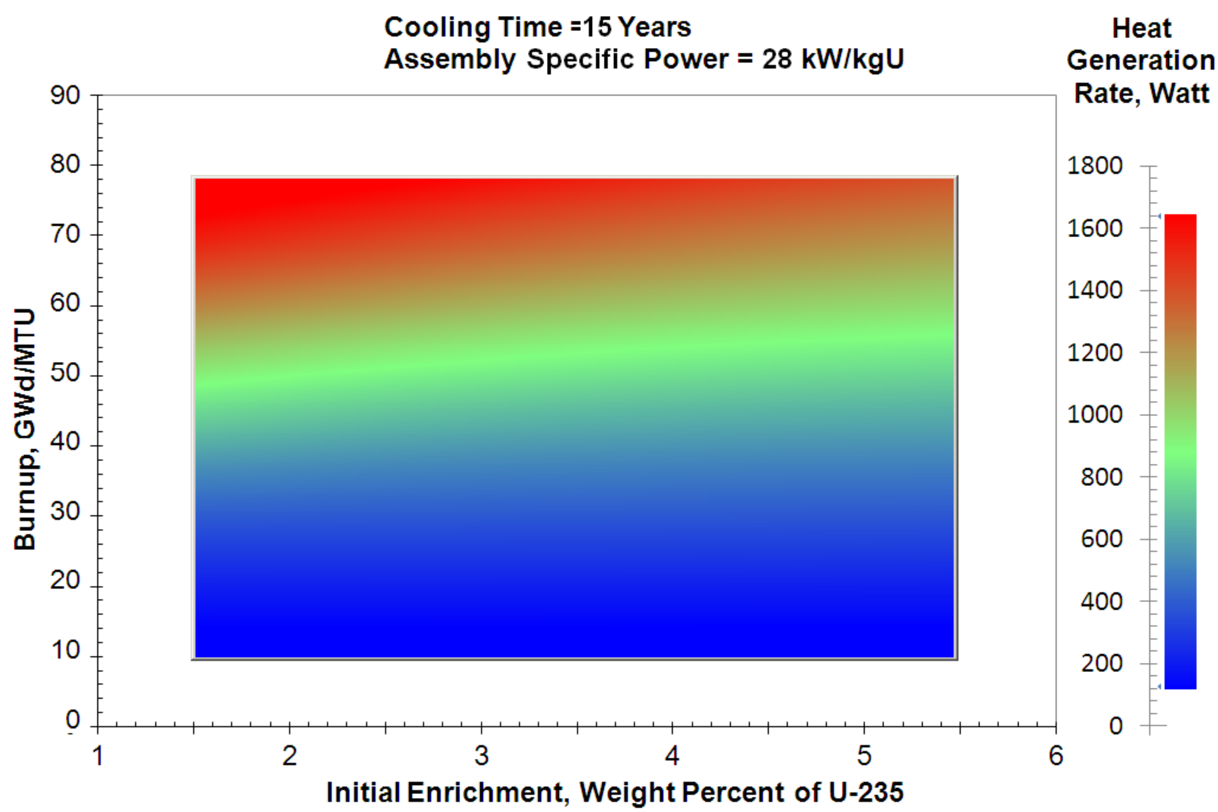

Figure A–2. Heat Generation Rates Thermal Map for Cooling Time of 15 Years and Assembly-Specific Power of 28 kW/kgU

Table A–3. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 28 kW/kgU, and Cooling Time of 25 Years*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,235*	*1,205*	*1,176*	*1,149*	*1,124*	*1,101*	*1,081*	*1,063*	*1,047*
70	*1,062*	*1,034*	*1,008*	*984*	*963*	*944*	*927*	*913*	*901*
60	*861*	*837*	*817*	*799*	*784*	*771*	*760*	*751*	*742*
50	*669*	*652*	*638*	*626*	*617*	*609*	*603*	*597*	*592*
40	497	487	479	474	469	466	462	460	457
30	349	347	345	344	343	341	339	338	336
20	224	224	223	222	221	220	218	217	216
10	110	108	107	106	105	105	104	104	103

*P_{min} = 103 watts P_{max} = 1,235 watts
Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

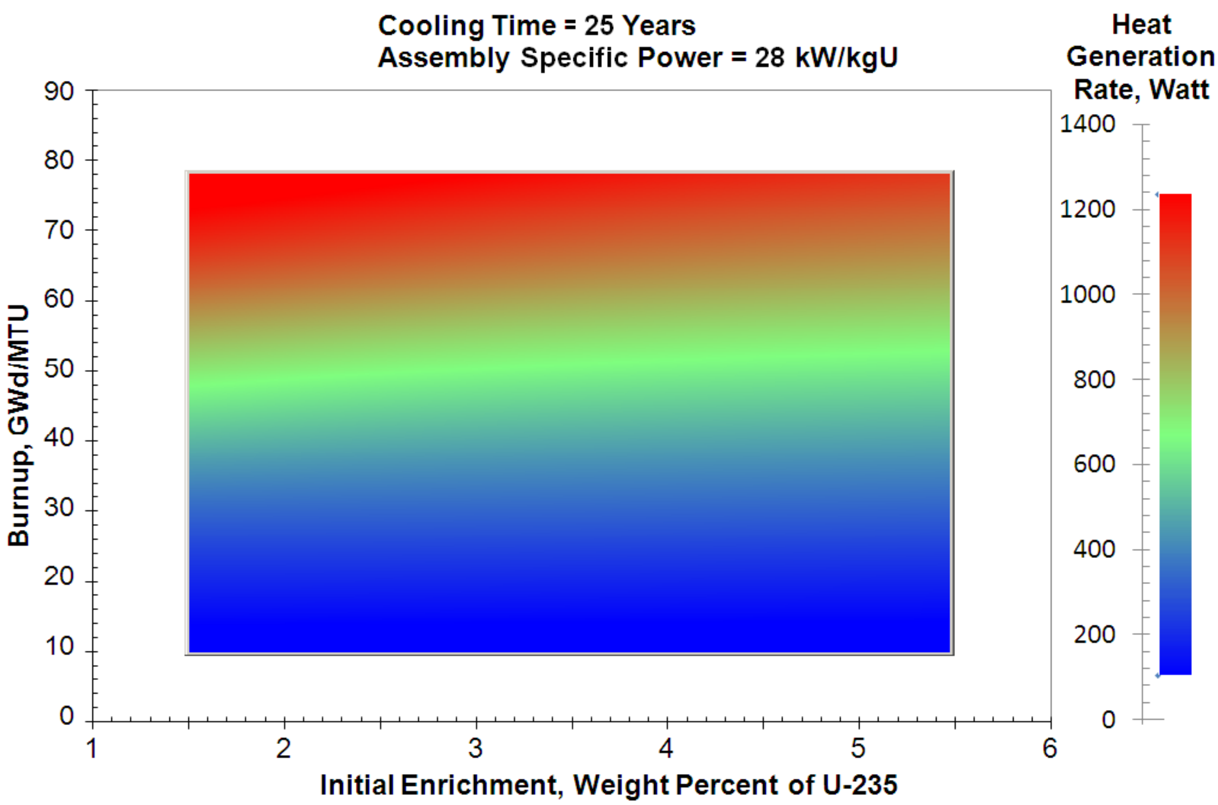

Figure A–3. Heat Generation Rates Thermal Map for Cooling Time of 25 Years and Assembly-Specific Power of 28 kW/kgU

Table A–4. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 31 kW/kgU, and Cooling Time of 5 Years*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*2,844*	*2,752*	*2,661*	*2,574*	*2,491*	*2,413*	*2,340*	*2,274*	*2,213*
70	*2,454*	*2,366*	*2,281*	*2,201*	*2,127*	*2,058*	*1,996*	*1,941*	*1,891*
60	*1,993*	*1,916*	*1,843*	*1,777*	*1,718*	*1,665*	*1,619*	*1,578*	*1,543*
50	*1,553*	*1,489*	*1,432*	*1,382*	*1,339*	*1,303*	*1,272*	*1,245*	*1,223*
40	1,149	1,103	1,064	1,032	1,006	984	966	951	939
30	794	770	749	734	721	710	701	694	687
20	498	487	478	471	466	461	457	453	451
10	241	237	234	232	230	228	227	226	225

*P_{min} = 225 watts P_{max} = 2,844 watts
Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

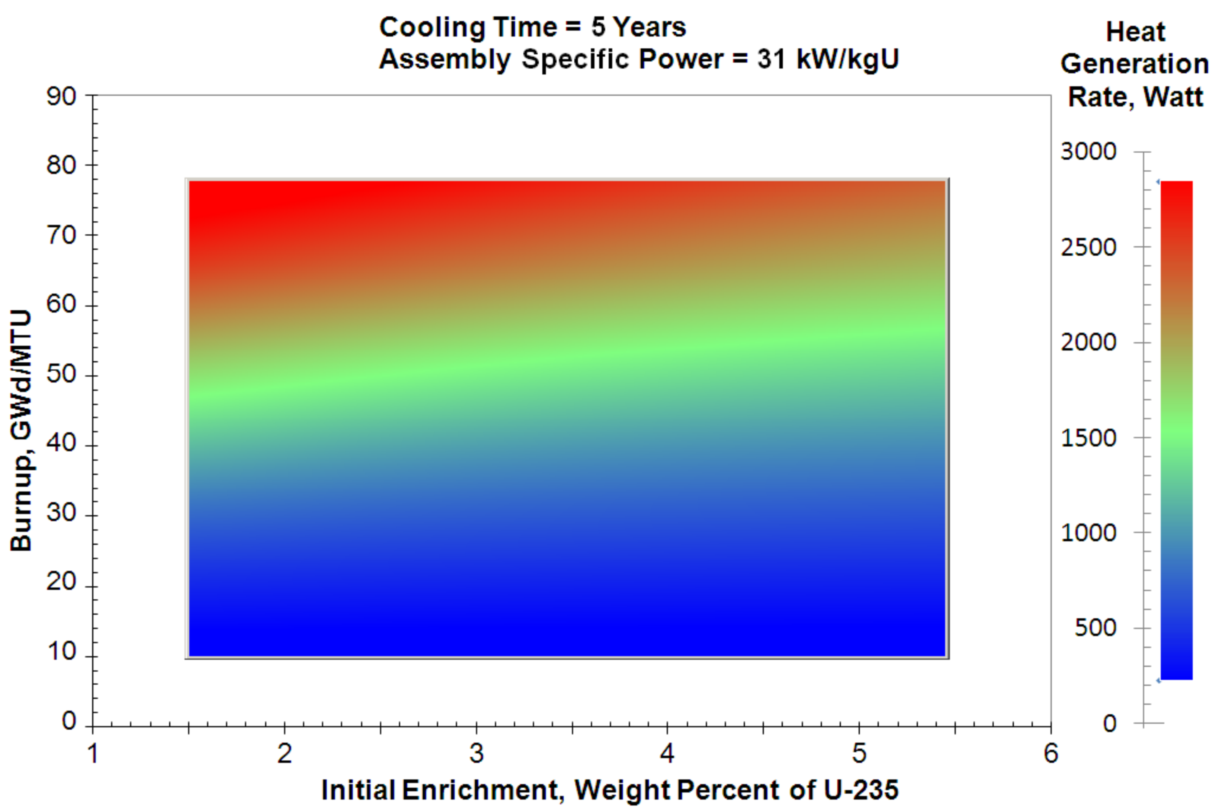

Figure A–4. Heat Generation Rates Thermal Map for Cooling Time of 5 Years and Assembly-Specific Power of 31 kW/kgU

Table A–5. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 31 kW/kgU, and Cooling Time of 15 Years*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,649*	*1,596*	*1,544*	*1,495*	*1,450*	*1,410*	*1,374*	*1,342*	*1,313*
70	*1,405*	*1,355*	*1,308*	*1,266*	*1,228*	*1,195*	*1,166*	*1,141*	*1,120*
60	*1,121*	*1,079*	*1,042*	*1,010*	*983*	*960*	*941*	*926*	*912*
50	*855*	*824*	*798*	*778*	*761*	*748*	*738*	*730*	*723*
40	619	601	587	578	570	565	561	557	554
30	422	417	414	413	411	410	409	407	406
20	265	266	266	266	266	265	265	264	264
10	130	130	130	130	130	130	129	129	129

*P_{min} = 129 watts P_{max} = 1,649 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

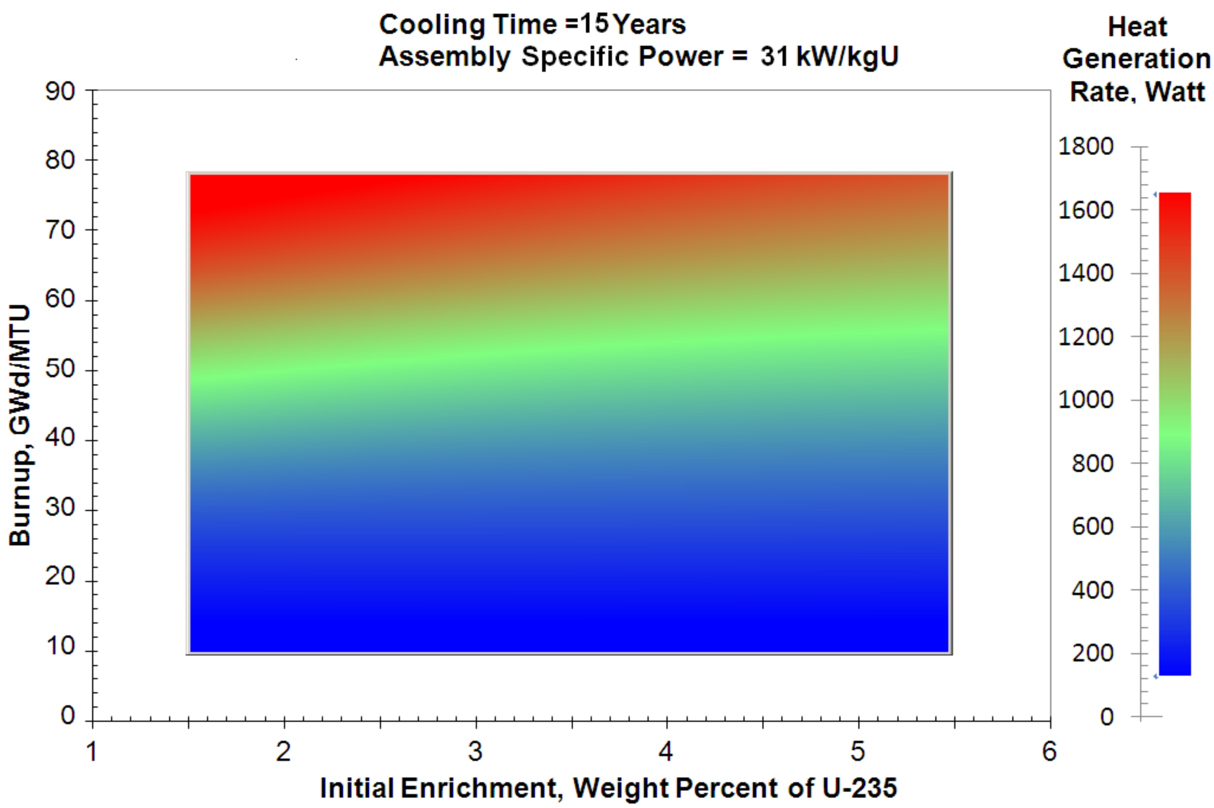

Figure A–5. Heat Generation Rates Thermal Map for Cooling Time of 15 Years and Assembly-Specific Power of 31 kW/kgU

Table A–6. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 31 kW/kgU, and Cooling Time of 25 Years*									
Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,241*	*1,210*	*1,181*	*1,153*	*1,128*	*1,105*	*1,084*	*1,066*	*1,050*
70	*1,067*	*1,039*	*1,012*	*987*	*966*	*947*	*930*	*916*	*903*
60	*863*	*839*	*818*	*800*	*785*	*772*	*761*	*752*	*744*
50	*671*	*653*	*639*	*627*	*618*	*610*	*604*	*598*	*594*
40	497	487	479	474	469	466	462	460	457
30	349	347	346	344	343	341	340	338	336
20	224	224	223	222	221	220	218	217	216
10	110	109	108	107	106	105	104	104	104

*P_{min} = 104 watts P_{max} = 1,241 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

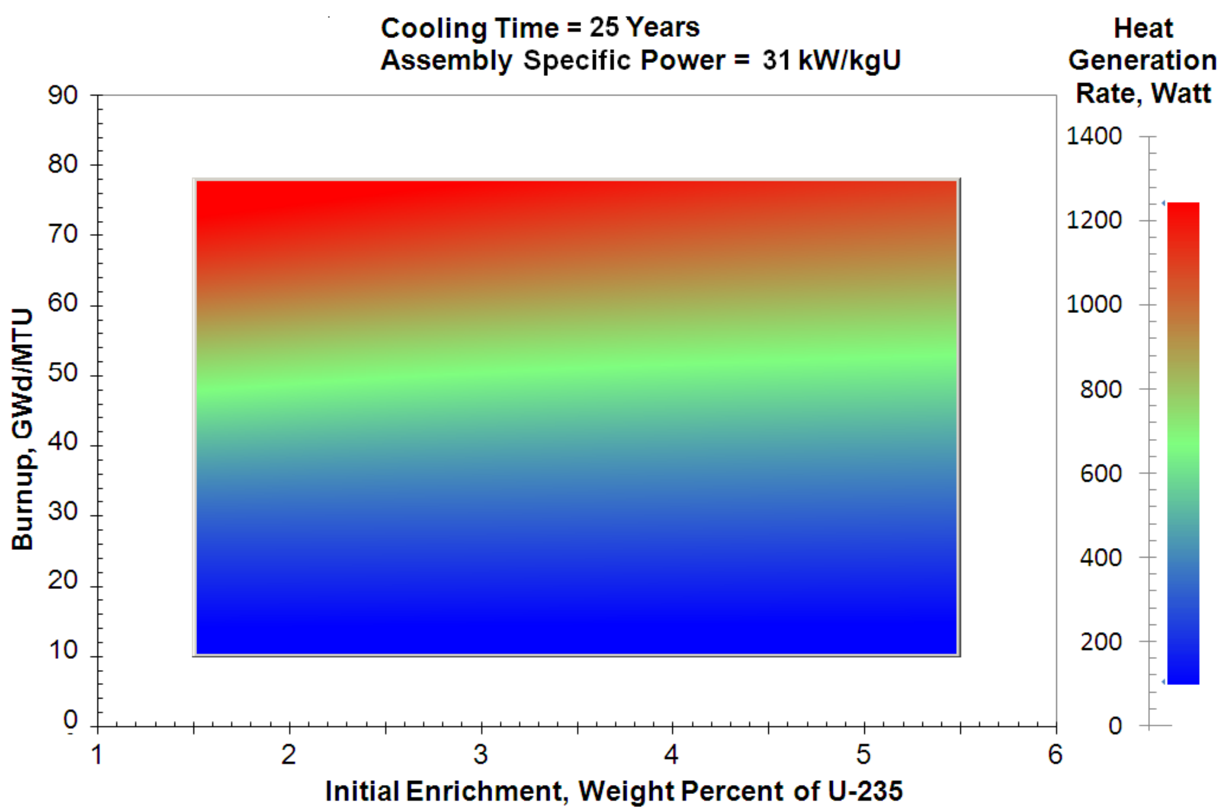

Figure A–6. Heat Generation Rates Thermal Map for Cooling Time of 25 Years and Assembly-Specific Power of 31 kW/kgU

Table A–7. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 40 kW/kgU, and Cooling Time of 5 Years*									
Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*2,973*	*2,876*	*2,781*	*2,689*	*2,602*	*2,519*	*2,443*	*2,373*	*2,309*
70	*2,564*	*2,472*	*2,383*	*2,298*	*2,220*	*2,147*	*2,082*	*2,023*	*1,972*
60	*2,083*	*2,001*	*1,925*	*1,855*	*1,792*	*1,737*	*1,688*	*1,645*	*1,608*
50	*1,621*	*1,553*	*1,493*	*1,440*	*1,394*	*1,355*	*1,323*	*1,295*	*1,272*
40	1,196	1,148	1,107	1,073	1,046	1,023	1,004	988	975
30	826	799	777	760	746	735	726	718	711
20	513	501	492	485	479	474	470	466	463
10	245	241	238	235	234	232	231	230	229

*P_{min} = 229 watts P_{max} = 2,973 watts
Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

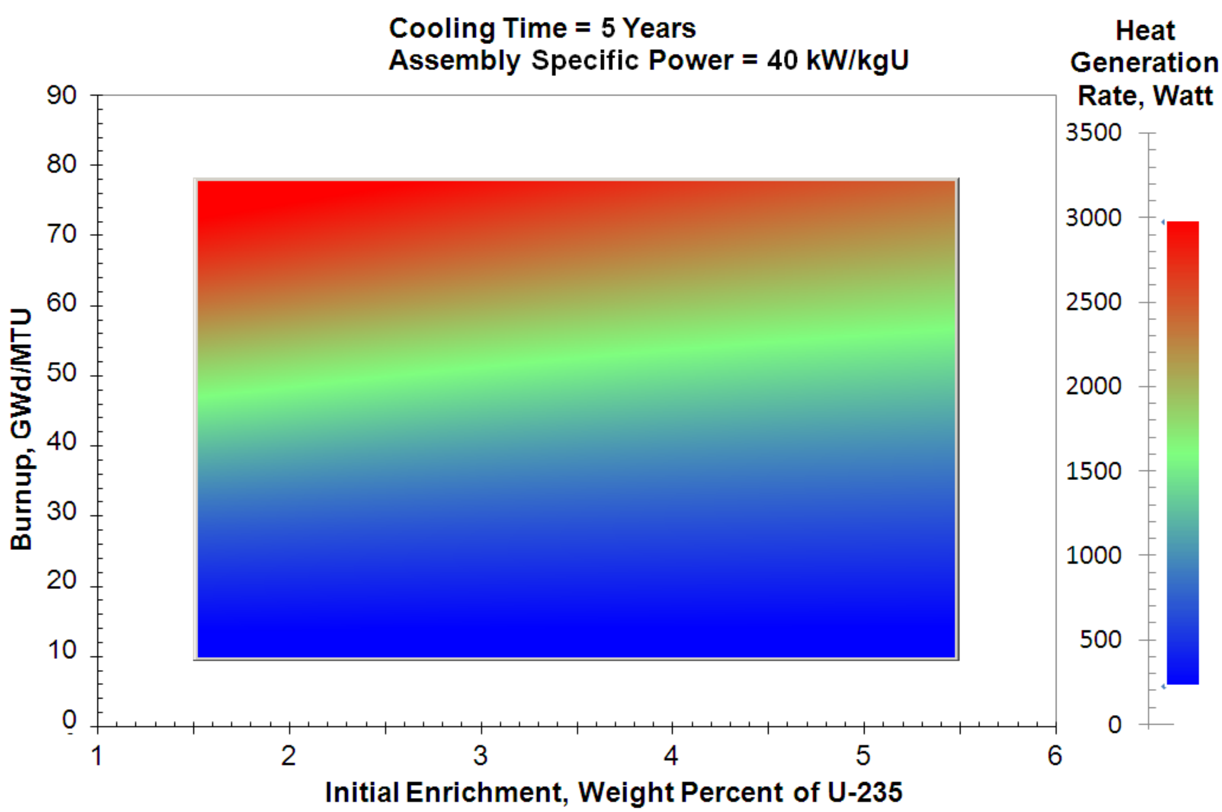

Figure A–7. Heat Generation Rates Thermal Map for Cooling Time of 5 Years and Assembly-Specific Power of 40 kW/kgU

Table A–8. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 40 kW/kgU, and Cooling Time of 15 Years*									
Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,671*	*1,615*	*1,561*	*1,511*	*1,465*	*1,423*	*1,386*	*1,353*	*1,324*
70	*1,420*	*1,369*	*1,321*	*1,277*	*1,238*	*1,203*	*1,174*	*1,149*	*1,128*
60	*1,131*	*1,088*	*1,050*	*1,017*	*990*	*967*	*948*	*932*	*919*
50	*860*	*828*	*802*	*781*	*765*	*751*	*741*	*733*	*726*
40	622	604	590	581	573	568	564	561	558
30	423	418	416	414	413	411	410	409	408
20	266	266	267	267	267	266	266	265	265
10	130	130	130	130	130	130	129	129	129

*P_{min} = 129 watts P_{max} = 1,671 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

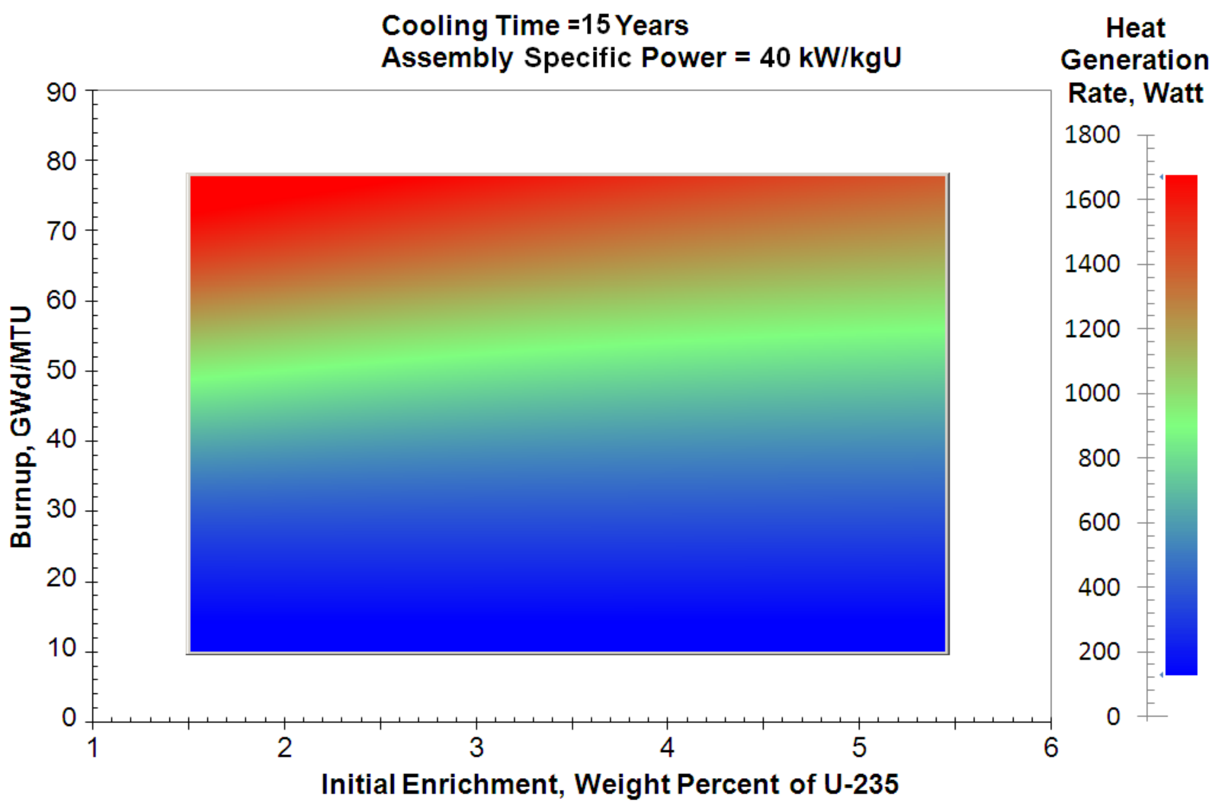

Figure A–8. Heat Generation Rates Thermal Map for Cooling Time of 15 Years and Assembly-Specific Power of 40 kW/kgU

Table A–9. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 40 kW/kgU, and Cooling Time of 25 Years*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,254*	*1,222*	*1,191*	*1,162*	*1,136*	*1,112*	*1,091*	*1,072*	*1,055*
70	*1,076*	*1,047*	*1,019*	*993*	*971*	*951*	*934*	*920*	*907*
60	*869*	*844*	*823*	*804*	*789*	*776*	*765*	*755*	*747*
50	*673*	*655*	*640*	*629*	*619*	*611*	*605*	*600*	*595*
40	499	489	481	475	471	467	464	462	459
30	350	348	346	345	344	342	341	339	337
20	225	225	224	223	222	220	219	218	217
10	110	109	108	107	106	105	104	104	104

*P_{min} = 104 watts P_{max} = 1,254 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

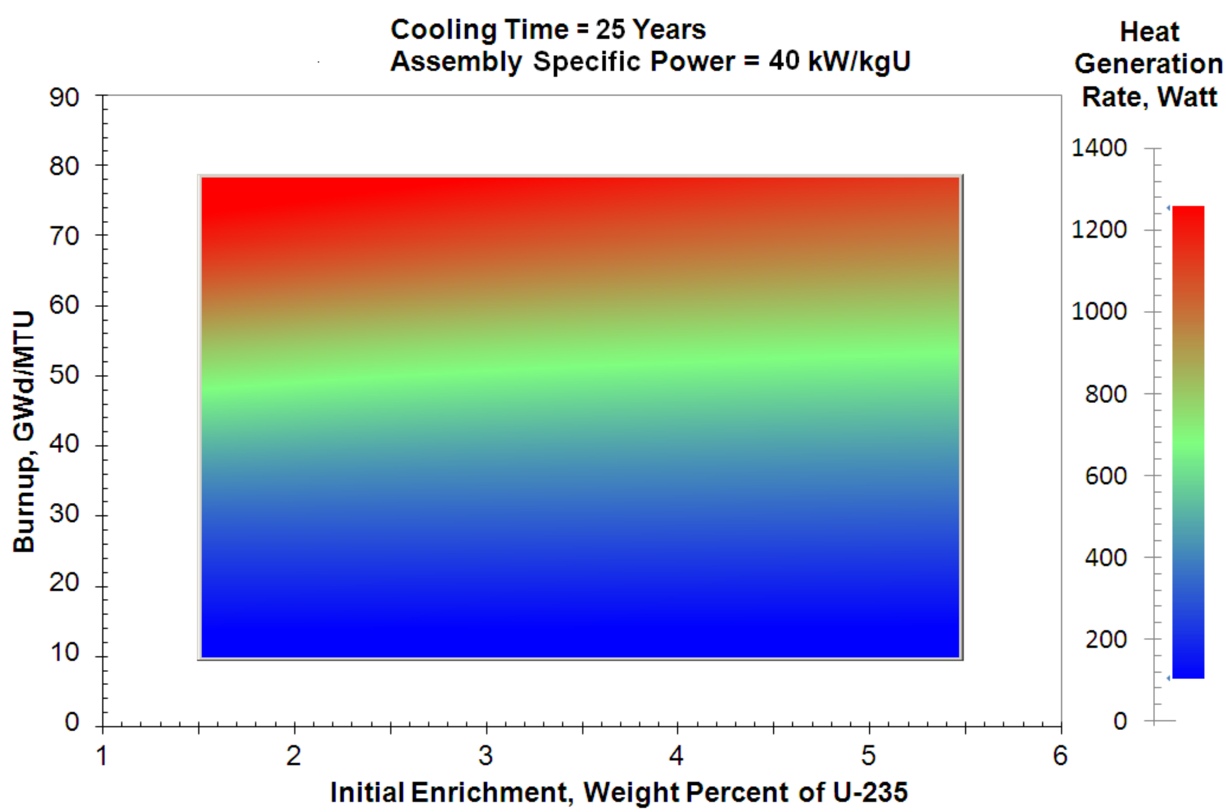

Figure A–9. Heat Generation Rates Thermal Map for Cooling Time of 25 Years and Assembly-Specific Power of 40 kW/kgU

Table A–10. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 60 kW/kgU, and Cooling Time of 5 Years*

Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*3,173*	*3,070*	*2,968*	*2,869*	*2,776*	*2,688*	*2,606*	*2,530*	*2,462*
70	*2,736*	*2,638*	*2,542*	*2,451*	*2,367*	*2,289*	*2,219*	*2,156*	*2,101*
60	*2,222*	*2,134*	*2,051*	*1,976*	*1,908*	*1,848*	*1,796*	*1,750*	*1,710*
50	*1,722*	*1,650*	*1,584*	*1,527*	*1,478*	*1,437*	*1,402*	*1,372*	*1,347*
40	1,270	1,218	1,173	1,136	1,105	1,081	1,060	1,043	1,029
30	869	839	816	797	782	770	760	752	744
20	535	522	512	503	497	492	487	483	480
10	250	246	243	240	239	237	236	235	234

*P_{min} = 234 watts P_{max} = 3,173 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

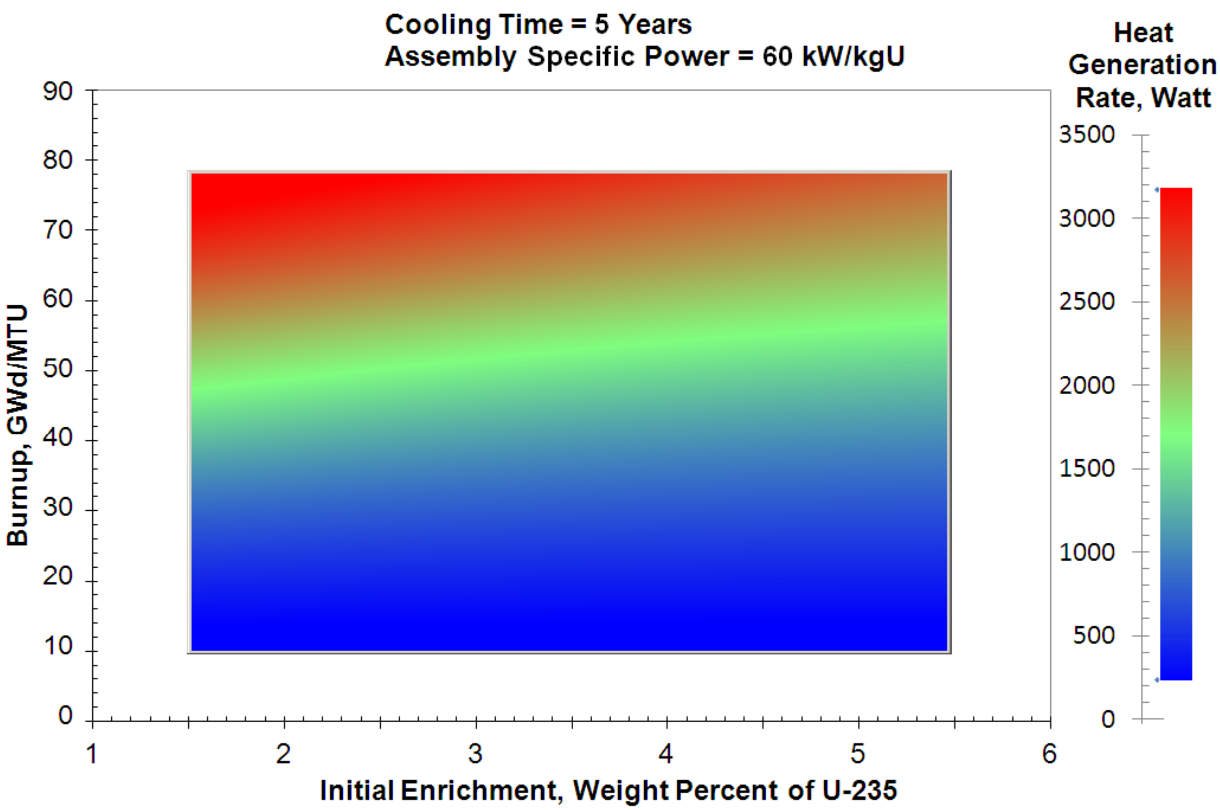

Figure A–10. Heat Generation Rates Thermal Map for Cooling Time of 5 Years and Assembly-Specific Power of 60 kW/kgU

Table A–11. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 60 kW/kgU, and Cooling Time of 15 Years*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,699*	*1,641*	*1,584*	*1,532*	*1,484*	*1,440*	*1,402*	*1,368*	*1,338*
70	*1,442*	*1,388*	*1,338*	*1,292*	*1,251*	*1,216*	*1,186*	*1,160*	*1,138*
60	*1,146*	*1,101*	*1,061*	*1,027*	*999*	*975*	*955*	*939*	*926*
50	*867*	*834*	*807*	*786*	*769*	*756*	*745*	*737*	*730*
40	626	607	593	583	576	571	567	563	561
30	424	419	417	415	414	413	412	411	410
20	266	267	267	268	267	267	267	266	265
10	130	130	130	130	130	130	130	129	129

*P_{min} = 129 watts P_{max} = 1,699 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

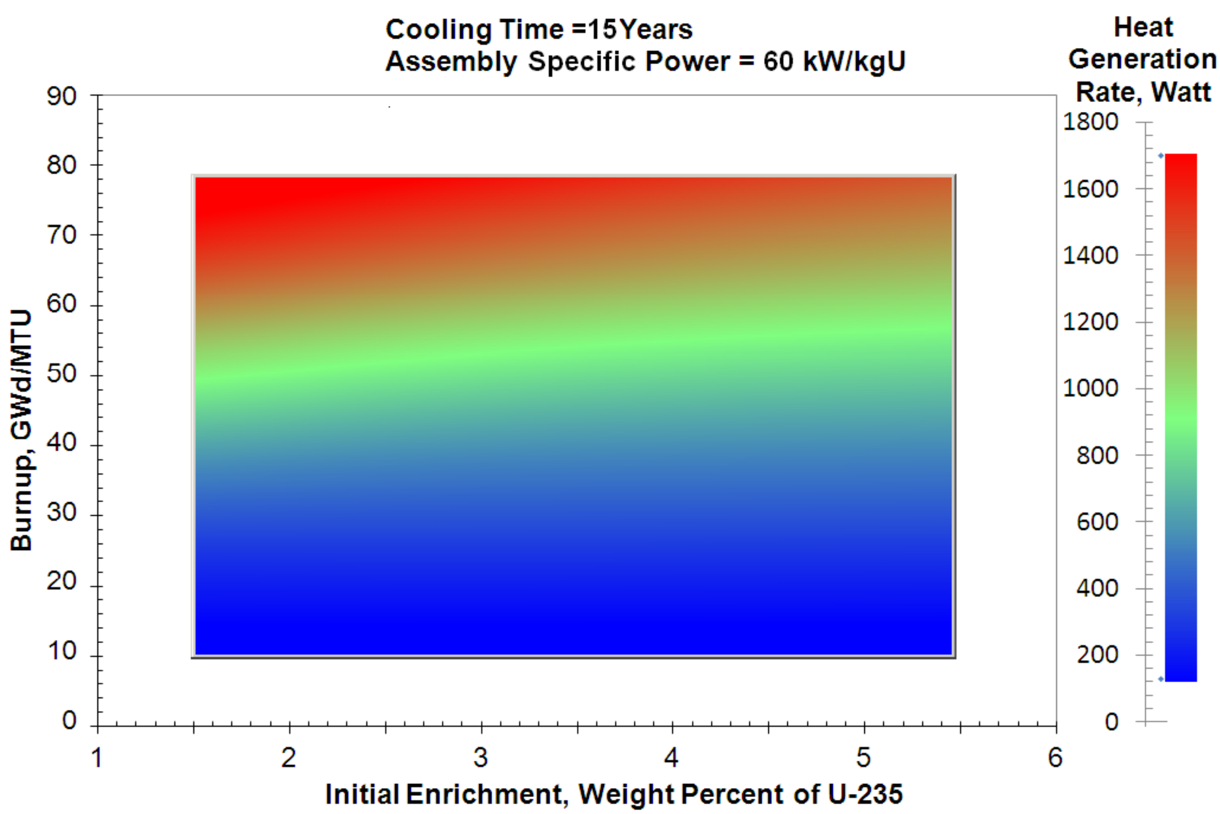

Figure A–11. Heat Generation Rates Thermal Map for Cooling Time of 15 Years and Assembly-Specific Power of 60 kW/kgU

Table A–12. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 60 kW/kgU, and Cooling Time of 25 Years*									
Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,272*	*1,238*	*1,205*	*1,175*	*1,147*	*1,122*	*1,100*	*1,080*	*1,063*
70	*1,089*	*1,058*	*1,028*	*1,002*	*978*	*958*	*940*	*926*	*913*
60	*878*	*852*	*829*	*809*	*793*	*780*	*768*	*759*	*751*
50	*677*	*658*	*643*	*631*	*621*	*613*	*607*	*602*	*597*
40	500	490	482	476	472	469	466	463	461
30	350	348	346	345	344	343	341	340	338
20	225	225	224	223	222	221	220	218	217
10	110	109	108	107	106	105	105	104	104

*P_{min} = 104 watts P_{max} = 1,272 watts
Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

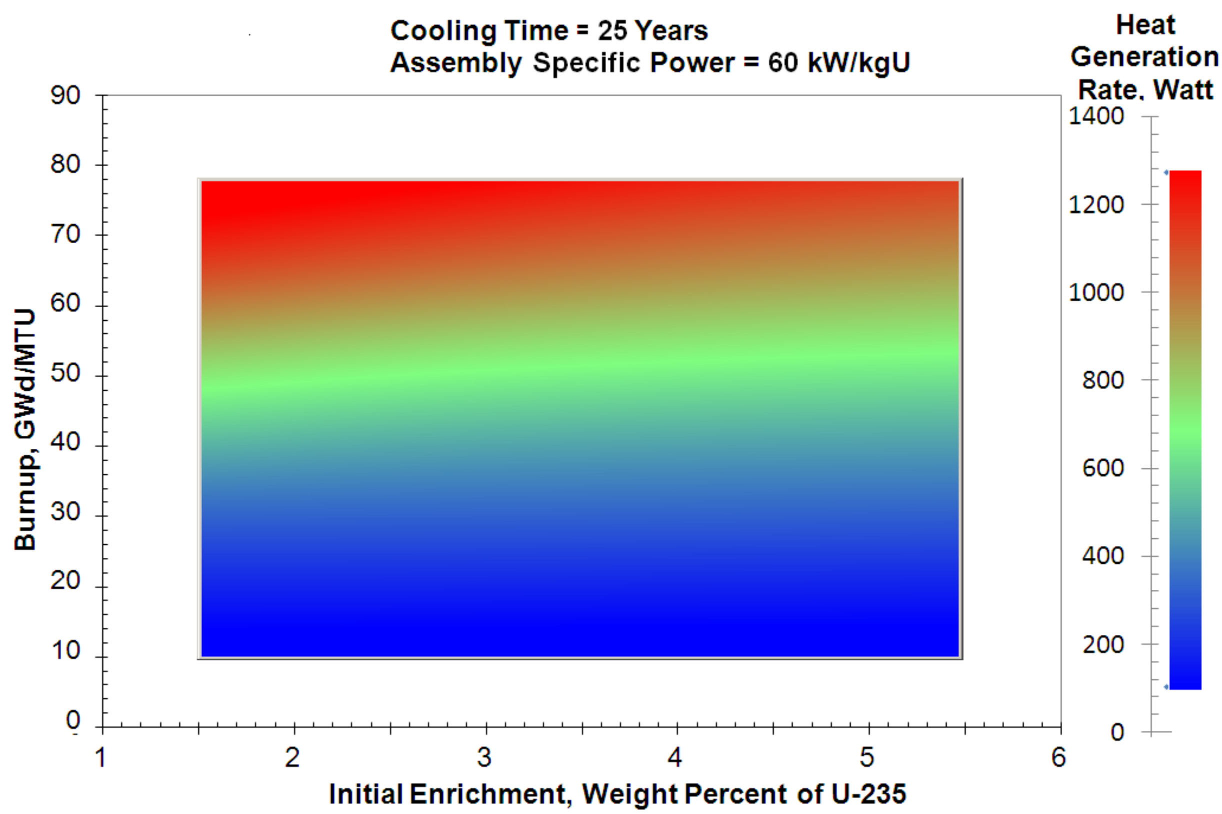

Figure A–12. Heat Generation Rates Thermal Map for Cooling Time of 25 Years and Assembly-Specific Power of 60 kW/kgU

Table A–13. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 80 kW/kgU, and Cooling Time of 5 Years*									
Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*3,316*	*3,206*	*3,098*	*2,995*	*2,896*	*2,803*	*2,717*	*2,638*	*2,566*
70	*2,856*	*2,751*	*2,650*	*2,554*	*2,465*	*2,384*	*2,310*	*2,245*	*2,186*
60	*2,314*	*2,221*	*2,133*	*2,054*	*1,983*	*1,919*	*1,864*	*1,816*	*1,775*
50	*1,793*	*1,716*	*1,647*	*1,587*	*1,535*	*1,491*	*1,454*	*1,423*	*1,397*
40	1,316	1,261	1,213	1,174	1,142	1,116	1,095	1,077	1,062
30	897	865	840	821	805	792	781	772	765
20	549	534	523	515	508	502	498	494	490
10	252	248	245	242	241	239	238	237	236

*P_{min} = 236 watts P_{max} = 3,316 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

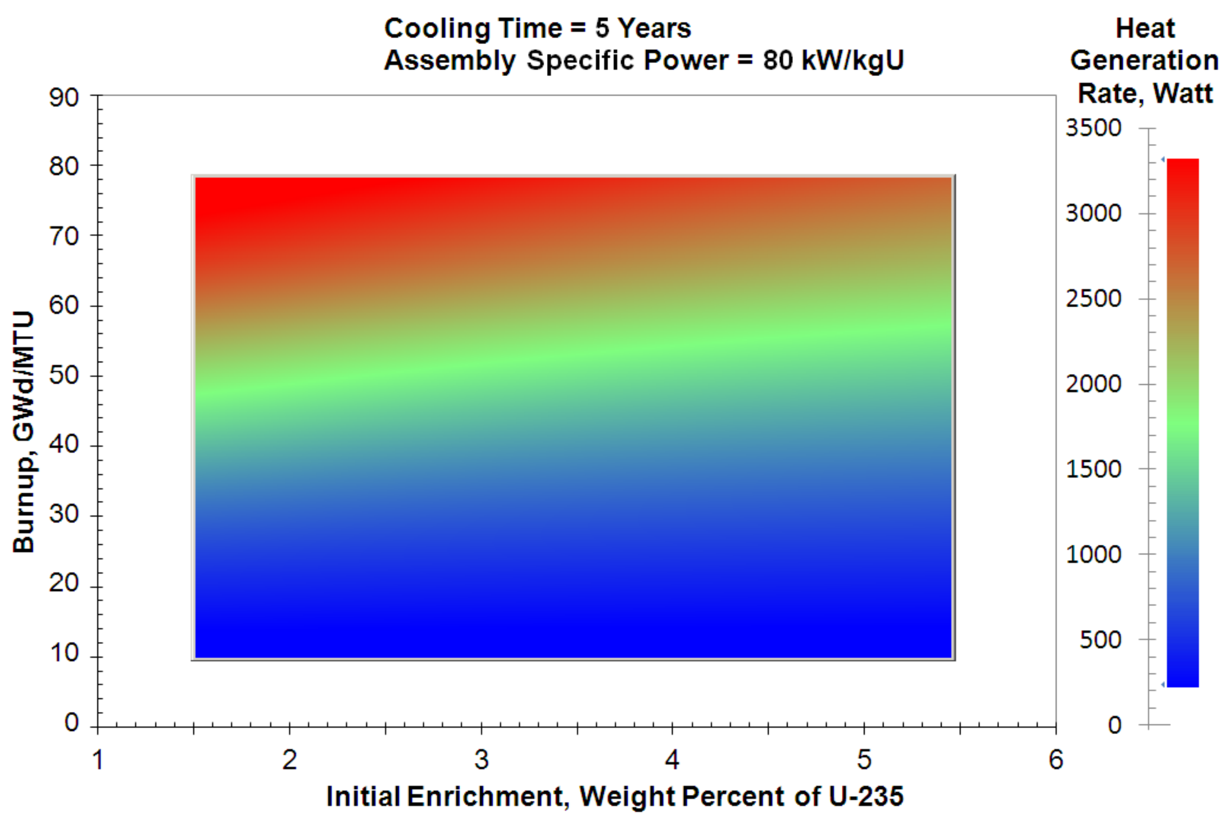

Figure A–13. Heat Generation Rates Thermal Map for Cooling Time of 5 Years and Assembly-Specific Power of 80 kW/kgU

Table A–14. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 80 kW/kgU, and Cooling Time of 15 Years*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,716*	*1,656*	*1,598*	*1,544*	*1,494*	*1,450*	*1,410*	*1,376*	*1,345*
70	*1,454*	*1,398*	*1,347*	*1,299*	*1,258*	*1,222*	*1,191*	*1,165*	*1,143*
60	*1,153*	*1,106*	*1,066*	*1,031*	*1,002*	*978*	*958*	*941*	*928*
50	*872*	*839*	*811*	*789*	*772*	*758*	*748*	*740*	*733*
40	628	608	594	584	577	572	568	565	562
30	426	420	418	416	415	414	413	412	411
20	267	268	268	268	268	268	267	267	266
10	130	130	130	130	130	130	129	129	129

*P_{min} = 129 watts P_{max} = 1,716 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

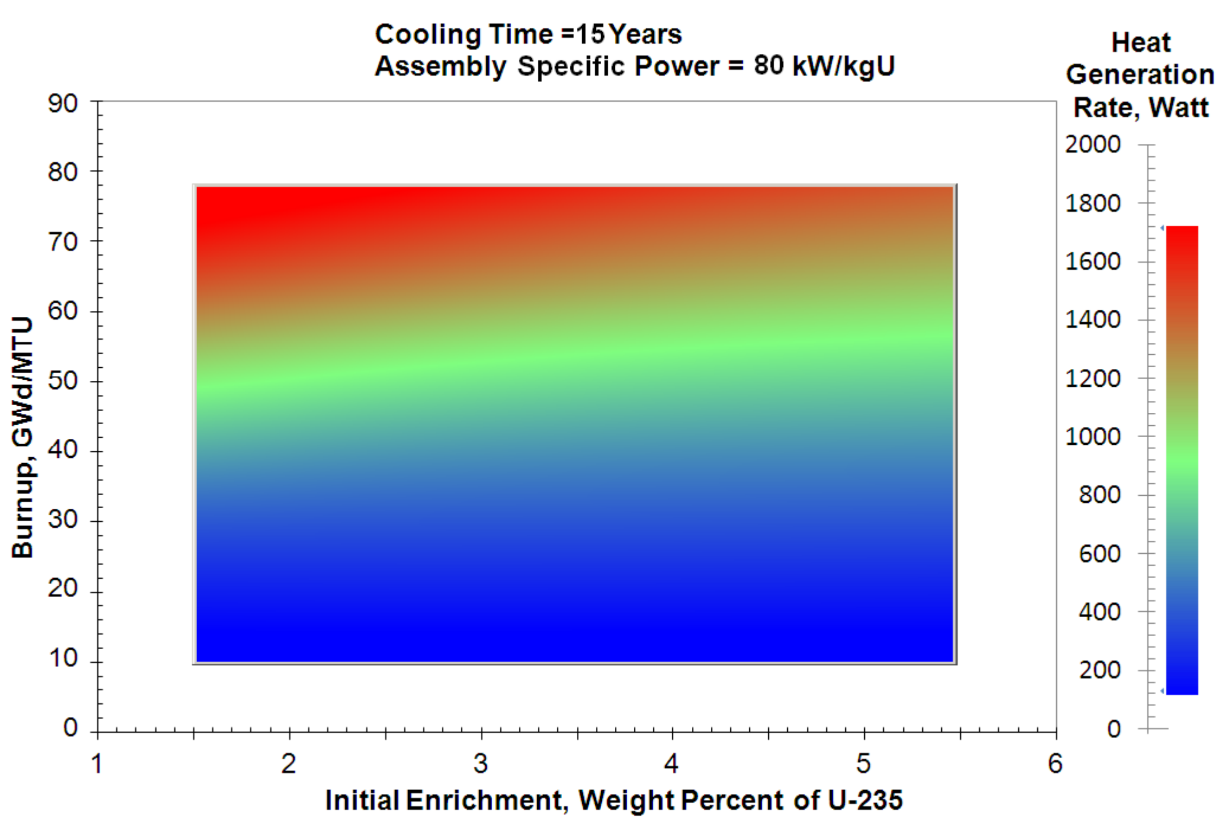

Figure A–14. Heat Generation Rates Thermal Map for Cooling Time of 15 Years and Assembly-Specific Power of 80 kW/kgU

Table A–15. Heat Generation Rates in Watts Per Single 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU, Specific Power of 80 kW/kgU, and Cooling Time of 25 Years*									
Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1,282*	*1,247*	*1,213*	*1,181*	*1,152*	*1,127*	*1,104*	*1,084*	*1,066*
70	*1,096*	*1,064*	*1,033*	*1,006*	*982*	*961*	*943*	*928*	*915*
60	*881*	*854*	*831*	*811*	*794*	*780*	*769*	*759*	*751*
50	*680*	*660*	*645*	*632*	*622*	*615*	*608*	*603*	*599*
40	501	490	482	476	472	469	466	464	462
30	350	348	347	346	345	343	342	340	339
20	226	225	225	224	223	221	220	219	218
10	109	108	107	106	106	105	104	104	104

*P_{min} = 104 watts P_{max} = 1,282 watts

Minimum (P_{min}) and maximum (P_{max}) values of heat generation rates are in blue and red font, respectively. Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

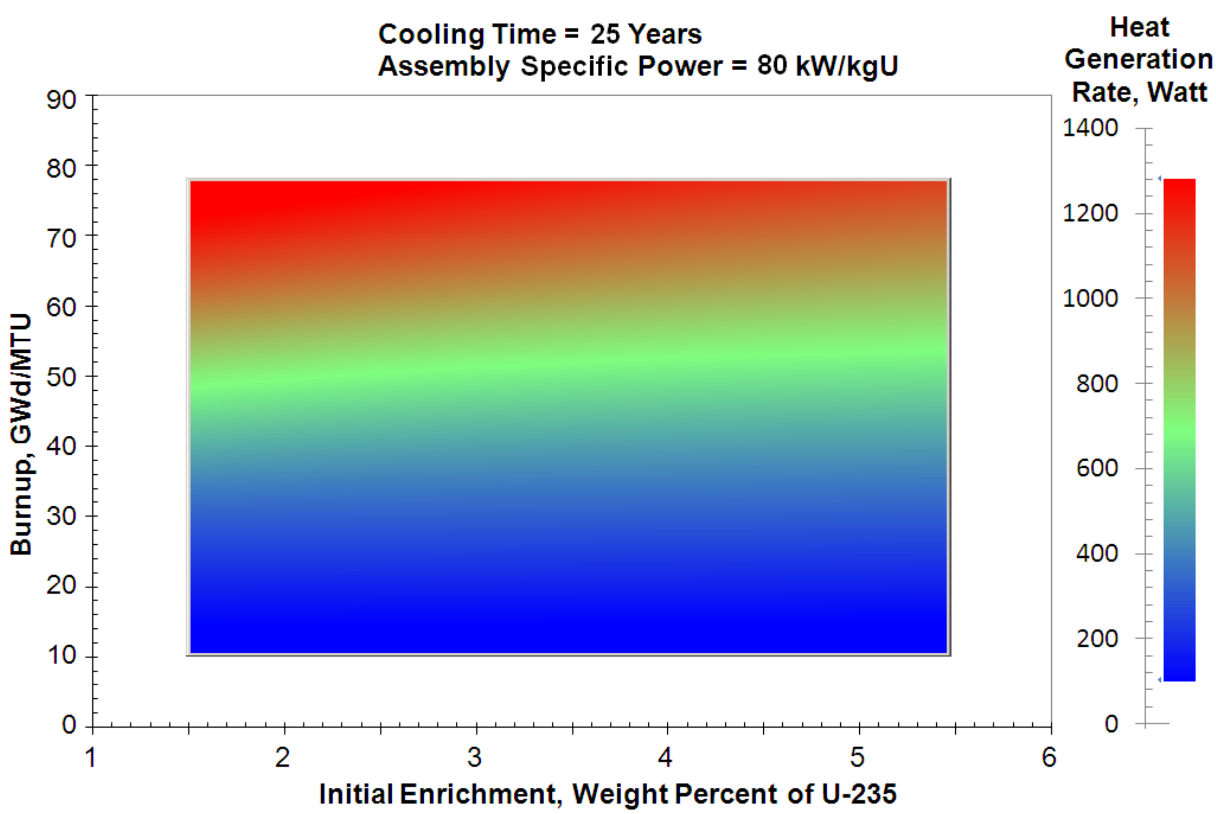

Figure A–15. Heat Generation Rates Thermal Map for Cooling Time of 25 Years and Assembly-Specific Power of 80 kW/kgU

APPENDIX B

THE PRESSURIZED WATER REACTOR SPENT NUCLEAR FUEL HEAT GENERATION RATE RATIOS FOR FOUR ELEVATED SPECIFIC POWER LEVELS

Table B–1. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 5 Years and Specific Power of 31.3 kW/kgU*

Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.020*	*1.020*	*1.020*	*1.020*	*1.020*	*1.020*	*1.020*	*1.020*	*1.020*
70	*1.021*	*1.021*	*1.021*	*1.020*	*1.020*	*1.020*	*1.020*	*1.020*	*1.020*
60	*1.017*	*1.017*	*1.017*	*1.017*	*1.017*	*1.017*	*1.016*	*1.016*	*1.016*
50	*1.020*	*1.020*	*1.020*	*1.020*	*1.019*	*1.019*	*1.019*	*1.019*	*1.019*
40	1.019	1.019	1.018	1.018	1.018	1.017	1.017	1.017	1.017
30	1.015	1.015	1.014	1.014	1.014	1.013	1.013	1.013	1.013
20	1.016	1.015	1.015	1.014	1.014	1.014	1.014	1.014	1.014
10	1.012	1.012	1.011	1.011	1.011	1.011	1.011	1.011	1.011

*$Ratio_{min}$ = 1.011 $Ratio_{max}$ = 1.021
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

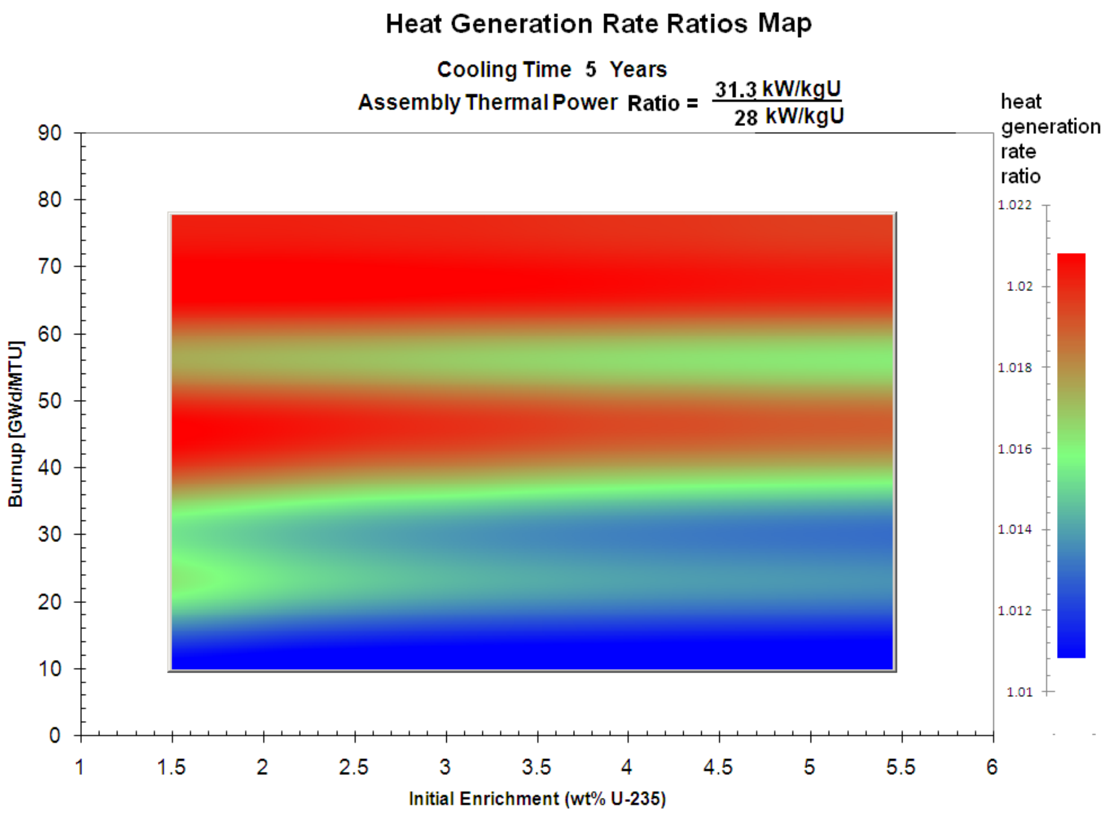

Figure B–1. Heat Generation Rate Ratios Map for the 31.3/28 Assembly Thermal Power Ratio and for 5-Year Cooling Time

Table B–2. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 15 Years and Specific Power of 31.3 kW/kgU*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.006*	*1.006*	*1.005*	*1.005*	*1.005*	*1.005*	*1.004*	*1.004*	*1.004*
70	*1.006*	*1.005*	*1.005*	*1.005*	*1.005*	*1.004*	*1.004*	*1.004*	*1.004*
60	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*
50	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*
40	1.001	1.001	1.001	1.001	1.001	1.001	1.001	1.001	1.001
30	1.001	1.001	1.001	1.001	1.001	1.001	1.001	1.002	1.002
20	1.001	1.001	1.001	1.001	1.001	1.001	1.001	1.001	1.002
10	1.003	1.003	1.003	1.003	1.003	1.003	1.003	1.003	1.003

*$Ratio_{min}$ = 1.001 $Ratio_{max}$ = 1.006
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

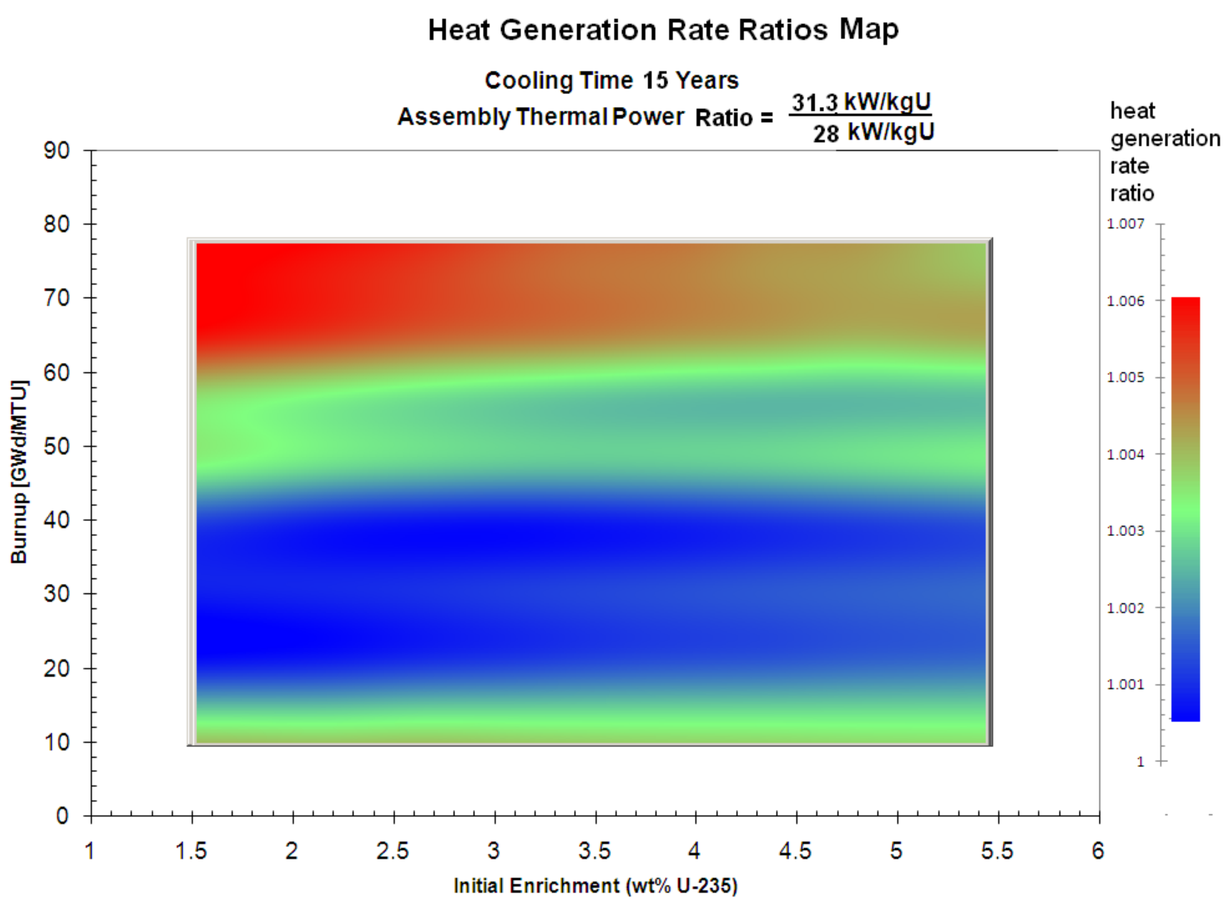

Figure B–2. Heat Generation Rate Ratios Map for the 31.3/28 Assembly Thermal Power Ratio and for 15-Year Cooling Time

Table B–3. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 25 Years and Specific Power Ratio of 31.3 kW/kgU*

Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.005*	*1.005*	*1.004*	*1.004*	*1.004*	*1.003*	*1.003*	*1.003*	*1.003*
70	*1.005*	*1.004*	*1.004*	*1.004*	*1.003*	*1.003*	*1.003*	*1.003*	*1.003*
60	*1.003*	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*
50	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*	*1.002*
40	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.001
30	1.000	1.000	1.001	1.001	1.001	1.001	1.001	1.001	1.001
20	1.000	1.000	1.000	1.001	1.001	1.001	1.001	1.001	1.001
10	1.003	1.003	1.003	1.003	1.003	1.003	1.003	1.003	1.003

*$Ratio_{min}$ = 1.000 $Ratio_{max}$ = 1.005
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

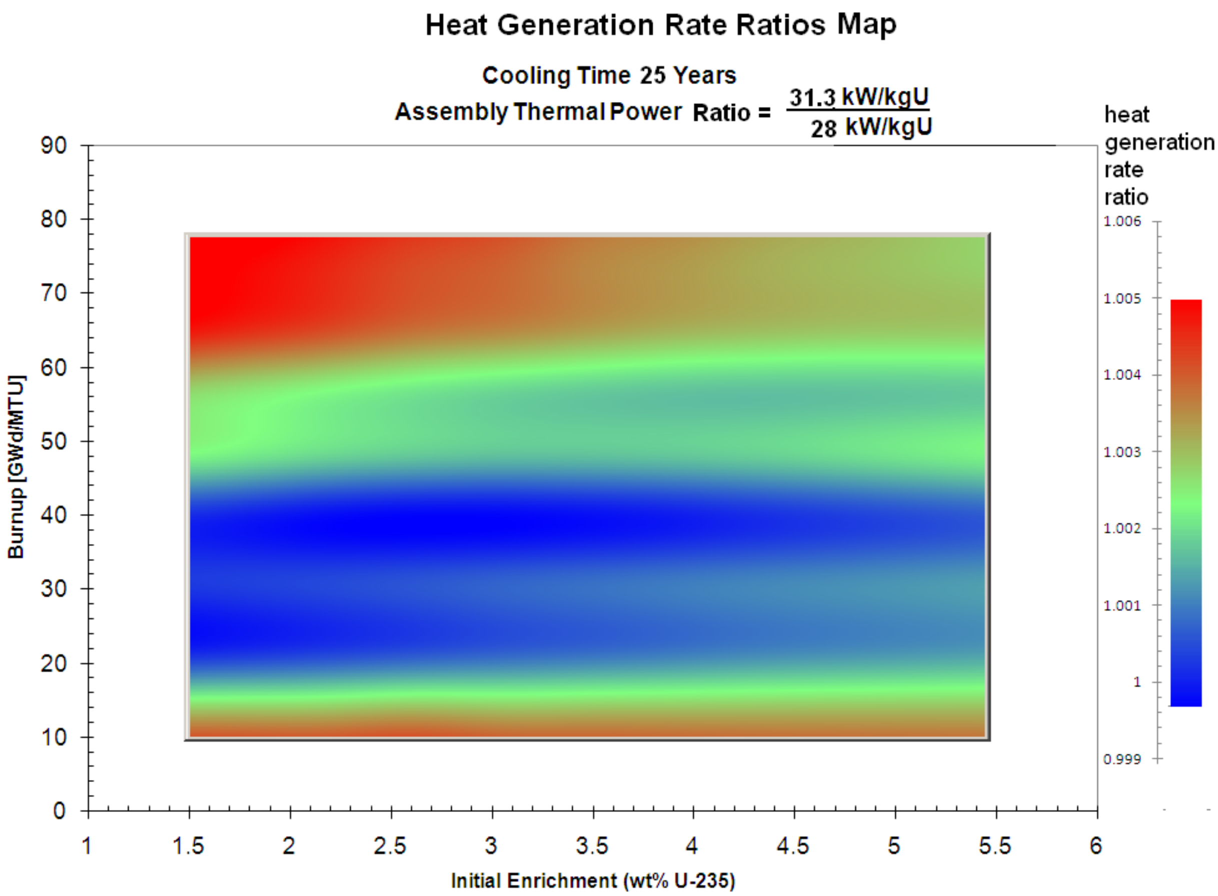

Figure B–3. Heat Generation Rate Ratios Map for the 31.3/28 Assembly Thermal Power Ratio and for 25-Year Cooling Time

Table B–4. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 5 Years and Specific Power of 40 kW/kgU*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.067*	*1.066*	*1.066*	*1.066*	*1.065*	*1.065*	*1.065*	*1.064*	*1.064*
70	*1.067*	*1.066*	*1.066*	*1.065*	*1.065*	*1.064*	*1.064*	*1.063*	*1.063*
60	*1.063*	*1.063*	*1.062*	*1.061*	*1.061*	*1.060*	*1.060*	*1.059*	*1.059*
50	*1.065*	*1.064*	*1.063*	*1.062*	*1.061*	*1.060*	*1.060*	*1.059*	*1.059*
40	1.061	1.060	1.059	1.058	1.058	1.057	1.057	1.056	1.056
30	1.055	1.053	1.052	1.051	1.050	1.049	1.048	1.048	1.048
20	1.047	1.045	1.044	1.043	1.043	1.042	1.042	1.042	1.042
10	1.030	1.029	1.028	1.028	1.028	1.028	1.027	1.027	1.027

*Ratio$_{min}$ = 1.027 Ratio$_{max}$ = 1.067
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

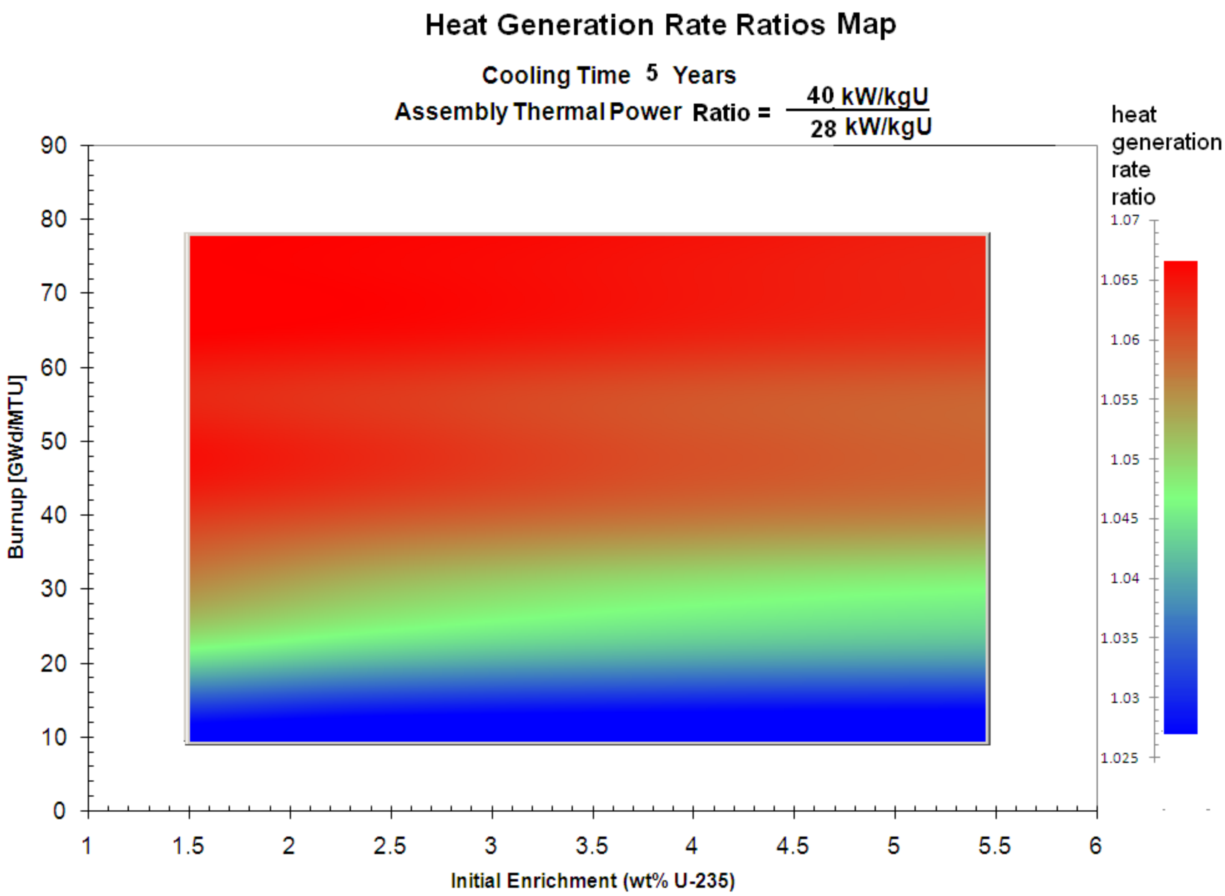

Figure B–4. Heat Generation Rate Ratios Map for the 40/28 Assembly Thermal Power Ratio and for 5-Year Cooling Time

Table B–5. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 15 Years and Specific Power Ratio of 40 kW/kgU*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.019*	*1.018*	*1.017*	*1.016*	*1.015*	*1.014*	*1.013*	*1.013*	*1.012*
70	*1.017*	*1.015*	*1.014*	*1.013*	*1.013*	*1.012*	*1.011*	*1.011*	*1.011*
60	*1.012*	*1.012*	*1.011*	*1.010*	*1.010*	*1.009*	*1.009*	*1.009*	*1.009*
50	*1.009*	*1.008*	*1.008*	*1.007*	*1.007*	*1.007*	*1.007*	*1.007*	*1.008*
40	1.007	1.006	1.006	1.006	1.006	1.007	1.007	1.007	1.008
30	1.003	1.004	1.004	1.004	1.005	1.005	1.005	1.006	1.006
20	1.003	1.003	1.004	1.004	1.004	1.005	1.005	1.005	1.005
10	1.003	1.003	1.004	1.004	1.004	1.004	1.004	1.004	1.004

*$Ratio_{min}$ = 1.003 $Ratio_{max}$ = 1.019
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

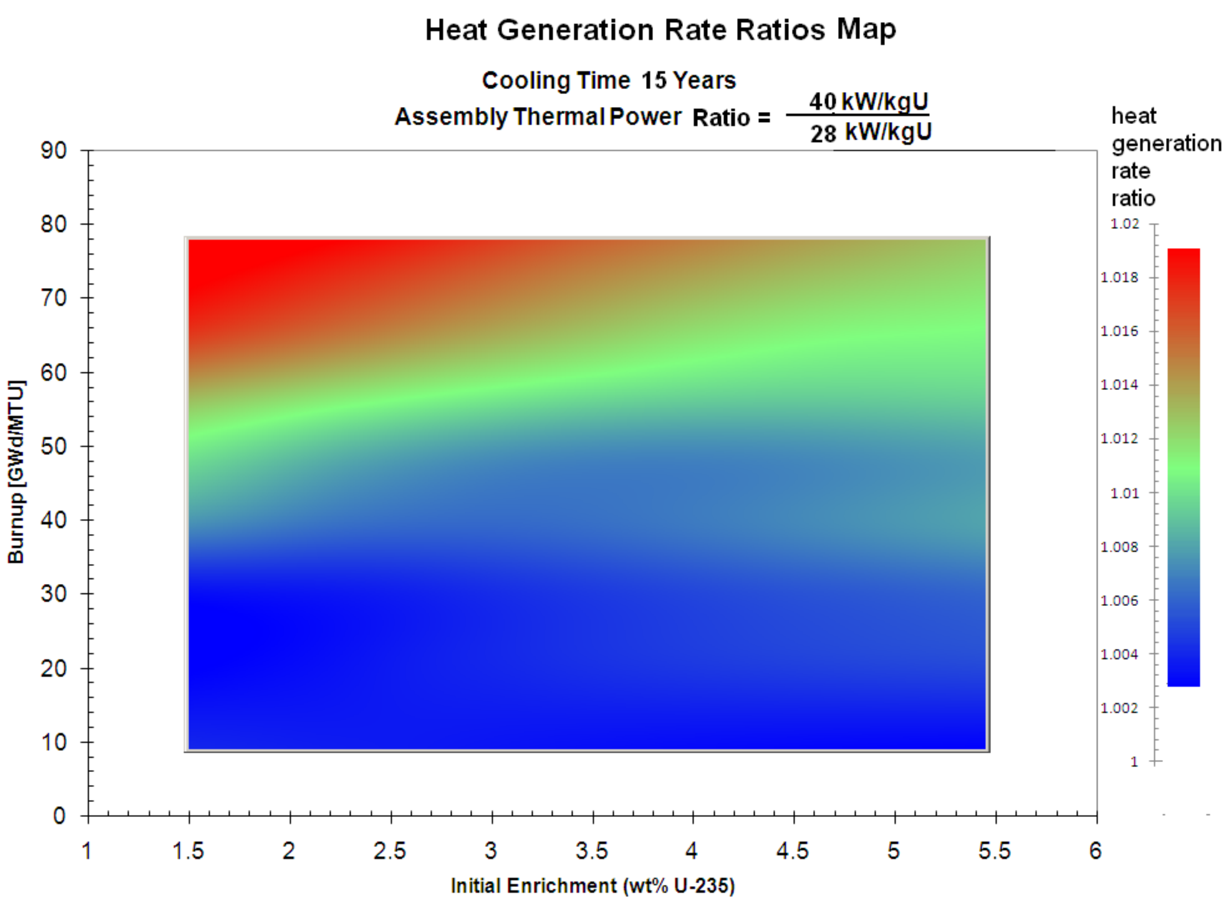

Figure B–5. Heat Generation Rate Ratios Map for the 40/28 Assembly Thermal Power Ratio and for 15-Year Cooling Time

Table B–6. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 25 Years and Specific Power of 40 kW/kgU*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.015*	*1.014*	*1.013*	*1.012*	*1.011*	*1.010*	*1.009*	*1.009*	*1.008*
70	*1.013*	*1.012*	*1.011*	*1.010*	*1.009*	*1.008*	*1.007*	*1.007*	*1.007*
60	*1.009*	*1.008*	*1.007*	*1.007*	*1.006*	*1.006*	*1.006*	*1.006*	*1.006*
50	*1.006*	*1.005*	*1.004*	*1.004*	*1.004*	*1.004*	*1.004*	*1.004*	*1.005*
40	1.004	1.003	1.003	1.003	1.004	1.004	1.005	1.005	1.006
30	1.001	1.002	1.002	1.003	1.003	1.004	1.004	1.004	1.005
20	1.002	1.002	1.003	1.003	1.003	1.004	1.004	1.004	1.004
10	1.003	1.003	1.003	1.003	1.003	1.003	1.003	1.003	1.003

*$Ratio_{min}$ = 1.001 $Ratio_{max}$ = 1.015
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

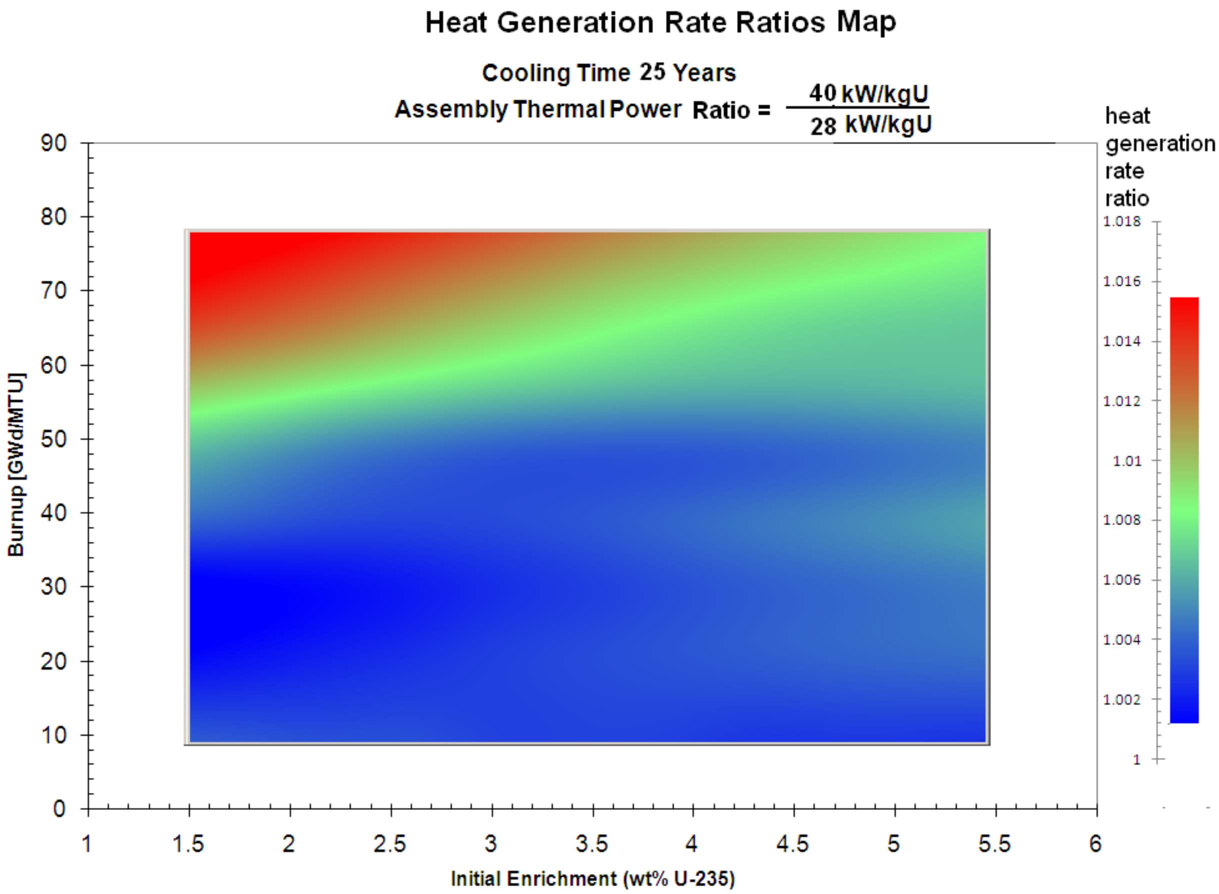

Figure B–6. Heat Generation Rate Ratios Map for the 40/28 Assembly Thermal Power Ratio and for 25-Year Cooling Time

Table B–7. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 5 Years and Specific Power Ratio of 60 kW/kgU*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.139*	*1.138*	*1.138*	*1.137*	*1.137*	*1.136*	*1.135*	*1.135*	*1.134*
70	*1.138*	*1.138*	*1.137*	*1.136*	*1.135*	*1.135*	*1.134*	*1.133*	*1.133*
60	*1.134*	*1.133*	*1.132*	*1.131*	*1.129*	*1.128*	*1.127*	*1.127*	*1.126*
50	*1.132*	*1.130*	*1.128*	*1.127*	*1.125*	*1.124*	*1.123*	*1.122*	*1.122*
40	1.127	1.124	1.122	1.120	1.118	1.117	1.116	1.115	1.114
30	1.110	1.107	1.104	1.102	1.100	1.099	1.098	1.097	1.097
20	1.091	1.088	1.085	1.084	1.082	1.082	1.081	1.080	1.080
10	1.049	1.049	1.049	1.050	1.050	1.050	1.050	1.050	1.050

*$Ratio_{min}$ = 1.049 $Ratio_{max}$ = 1.139
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

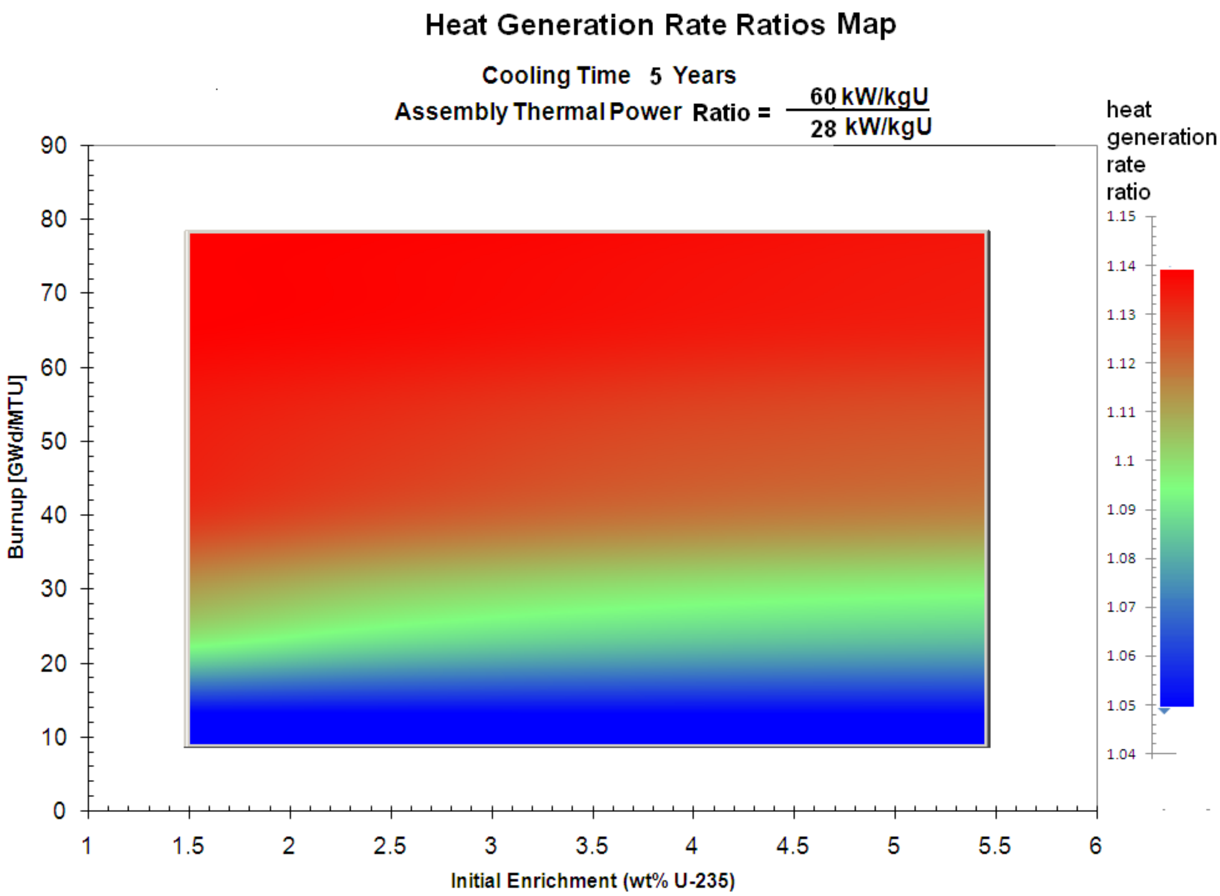

Figure B–7. Heat Generation Rate Ratios Map for the 60/28 Assembly Thermal Power Ratio and for 5-Year Cooling Time

Table B–8. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 15 Years and Specific Power of 60 kW/kgU*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.036*	*1.034*	*1.032*	*1.030*	*1.028*	*1.026*	*1.025*	*1.024*	*1.023*
70	*1.032*	*1.030*	*1.027*	*1.025*	*1.024*	*1.022*	*1.021*	*1.021*	*1.020*
60	*1.026*	*1.024*	*1.022*	*1.020*	*1.019*	*1.018*	*1.017*	*1.017*	*1.017*
50	*1.018*	*1.016*	*1.014*	*1.013*	*1.012*	*1.012*	*1.013*	*1.013*	*1.014*
40	1.012	1.011	1.010	1.010	1.011	1.011	1.012	1.012	1.013
30	1.006	1.006	1.006	1.007	1.008	1.008	1.009	1.009	1.010
20	1.005	1.005	1.006	1.007	1.007	1.007	1.008	1.008	1.008
10	1.001	1.003	1.004	1.004	1.005	1.005	1.006	1.006	1.006

*$Ratio_{min}$ = 1.001 $Ratio_{max}$ = 1.036
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

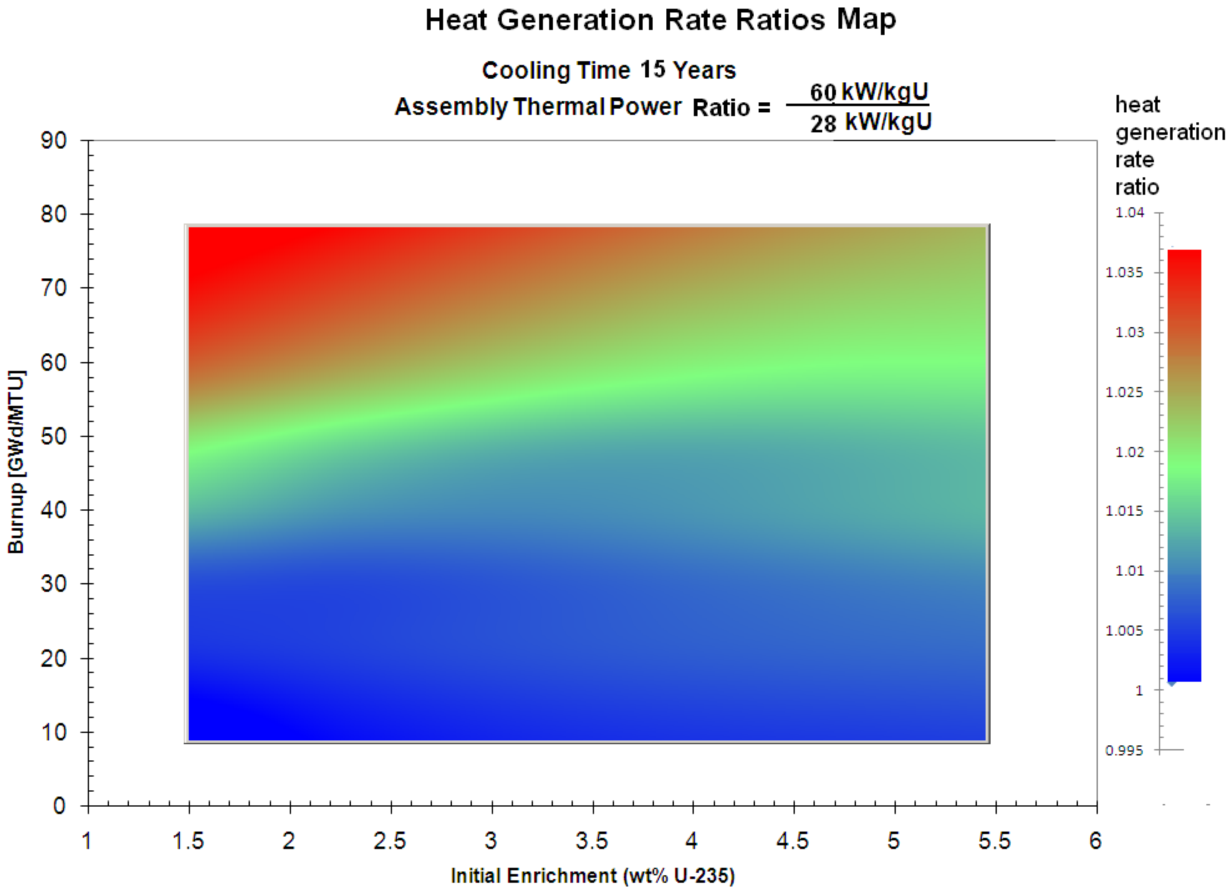

Figure B–8. Heat Generation Rate Ratios Map for the 60/28 Assembly Thermal Power Ratio and for 15-Year Cooling Time

Table B–9. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 25 Years and Specific Power Ratio of 60 kW/kgU*									
Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.030*	*1.027*	*1.025*	*1.023*	*1.021*	*1.019*	*1.017*	*1.016*	*1.015*
70	*1.025*	*1.023*	*1.020*	*1.018*	*1.017*	*1.015*	*1.014*	*1.013*	*1.013*
60	*1.019*	*1.017*	*1.015*	*1.013*	*1.012*	*1.011*	*1.011*	*1.011*	*1.011*
50	*1.011*	*1.009*	*1.008*	*1.007*	*1.007*	*1.007*	*1.007*	*1.008*	*1.008*
40	1.007	1.006	1.005	1.005	1.006	1.007	1.007	1.008	1.009
30	1.002	1.002	1.003	1.004	1.005	1.005	1.006	1.007	1.007
20	1.003	1.003	1.004	1.005	1.005	1.006	1.006	1.007	1.007
10	1.000	1.001	1.003	1.003	1.004	1.004	1.005	1.005	1.005

*$Ratio_{min}$ = 1.000 $Ratio_{max}$ = 1.030
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

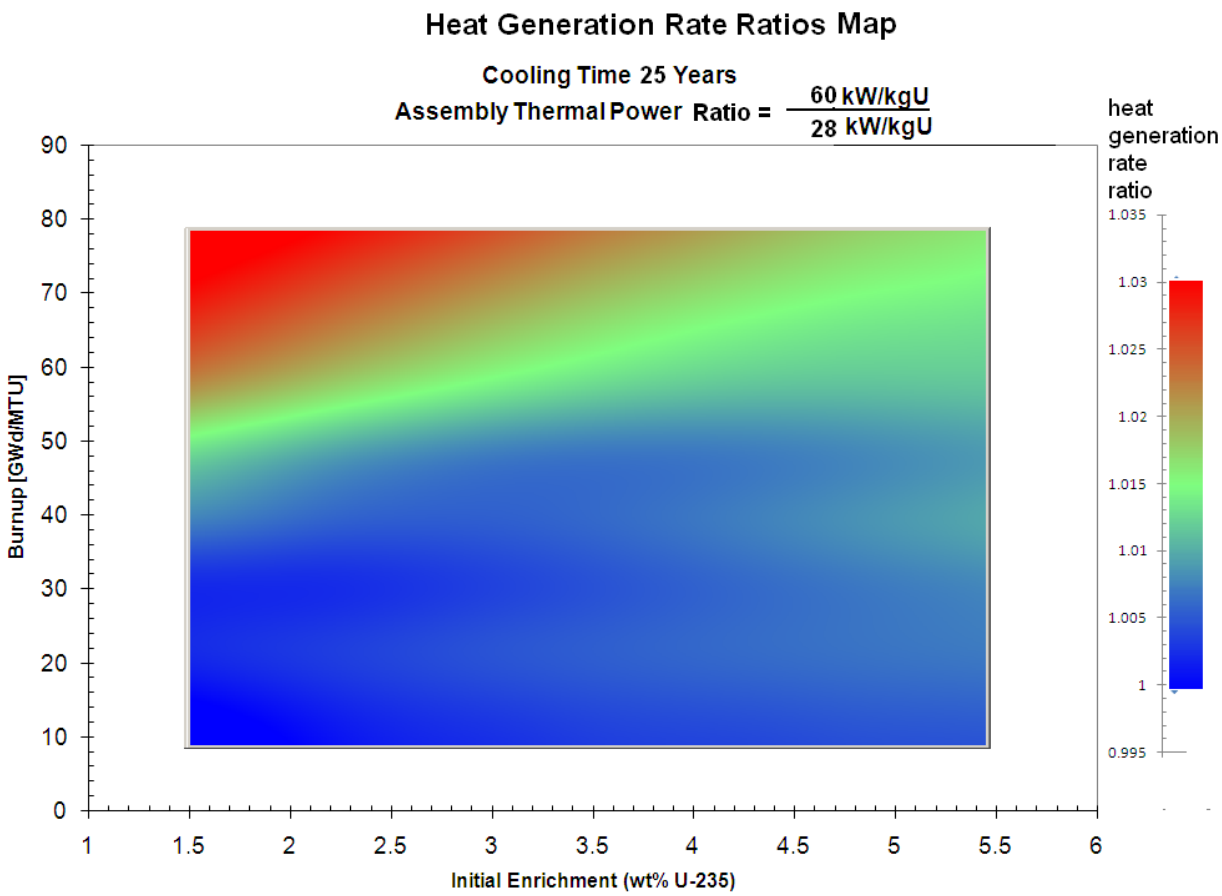

Figure B–9. Heat Generation Rate Ratios Map for the 60/28 Assembly Thermal Power Ratio and for 25-Year Cooling Time

Table B–10. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 5 Years and Specific Power of 80 kW/kgU/28 kW/kgU*

Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.190*	*1.189*	*1.188*	*1.187*	*1.186*	*1.185*	*1.184*	*1.183*	*1.183*
70	*1.188*	*1.187*	*1.186*	*1.184*	*1.183*	*1.182*	*1.181*	*1.180*	*1.179*
60	*1.181*	*1.179*	*1.177*	*1.175*	*1.173*	*1.172*	*1.171*	*1.170*	*1.169*
50	*1.179*	*1.176*	*1.173*	*1.170*	*1.168*	*1.167*	*1.165*	*1.164*	*1.163*
40	1.168	1.164	1.160	1.158	1.155	1.154	1.152	1.151	1.150
30	1.146	1.141	1.137	1.134	1.132	1.130	1.129	1.128	1.127
20	1.118	1.114	1.110	1.108	1.107	1.105	1.104	1.104	1.103
10	1.058	1.058	1.058	1.058	1.058	1.059	1.059	1.059	1.059

*$Ratio_{min}$ = 1.058 $Ratio_{max}$ = 1.190
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

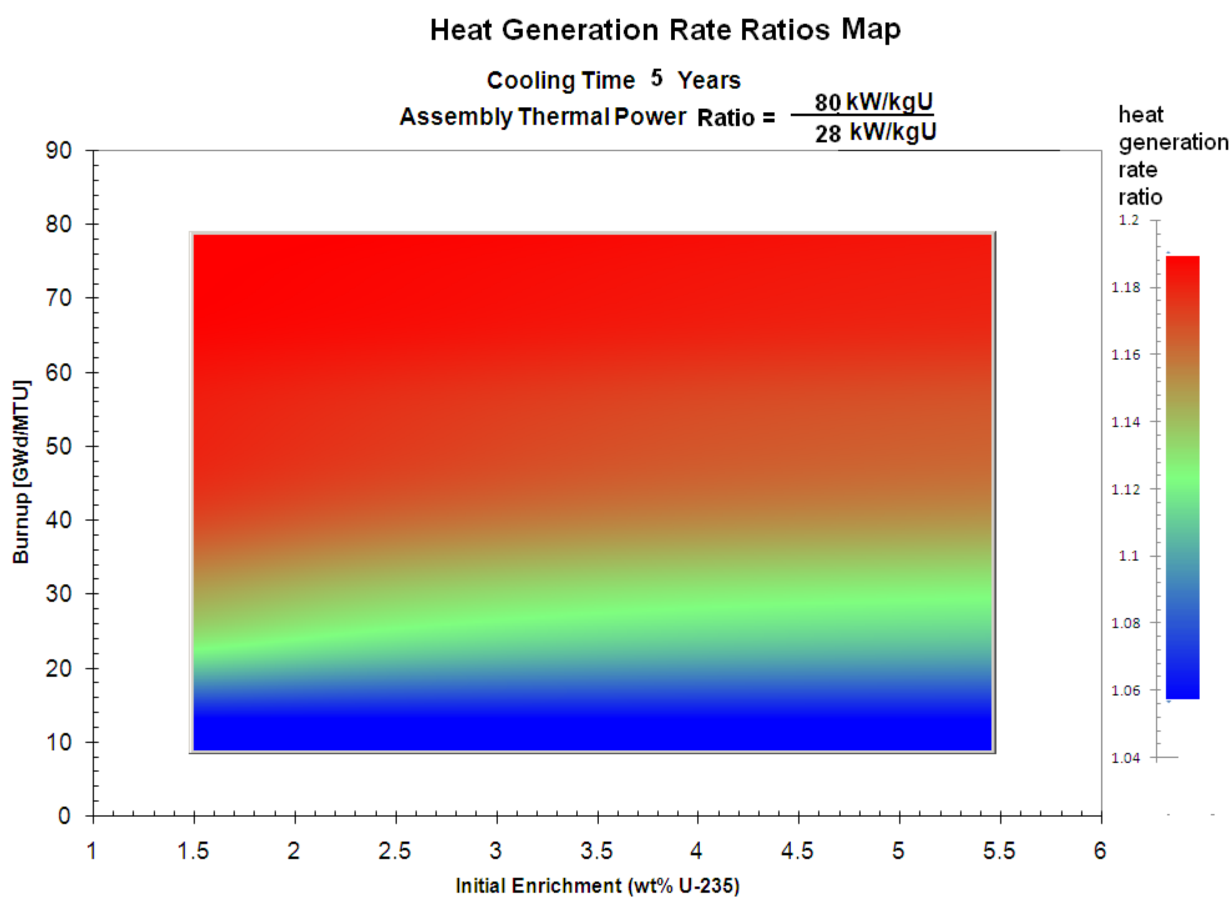

Figure B–10. Heat Generation Rate Ratios Map for the 80/28 Assembly Thermal Power Ratio and for 5-Year Cooling Time

Table B–11. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 15 Years and Specific Power Ratio of 80 kW/kgU*									
Burnup, GWd/MTU	Initial Enrichment, Weight Percent of Uranium								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.047*	*1.044*	*1.041*	*1.038*	*1.035*	*1.033*	*1.031*	*1.030*	*1.028*
70	*1.041*	*1.037*	*1.034*	*1.032*	*1.029*	*1.027*	*1.026*	*1.025*	*1.024*
60	*1.032*	*1.029*	*1.026*	*1.023*	*1.022*	*1.020*	*1.020*	*1.019*	*1.020*
50	*1.024*	*1.021*	*1.019*	*1.017*	*1.016*	*1.016*	*1.016*	*1.017*	*1.017*
40	1.015	1.013	1.012	1.012	1.012	1.013	1.014	1.014	1.015
30	1.009	1.008	1.008	1.009	1.010	1.010	1.011	1.012	1.012
20	1.008	1.008	1.009	1.009	1.010	1.010	1.011	1.011	1.011
10	0.999	1.001	1.002	1.003	1.003	1.004	1.004	1.005	1.005

*$Ratio_{min}$ = 0.999 $Ratio_{max}$ = 1.047
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

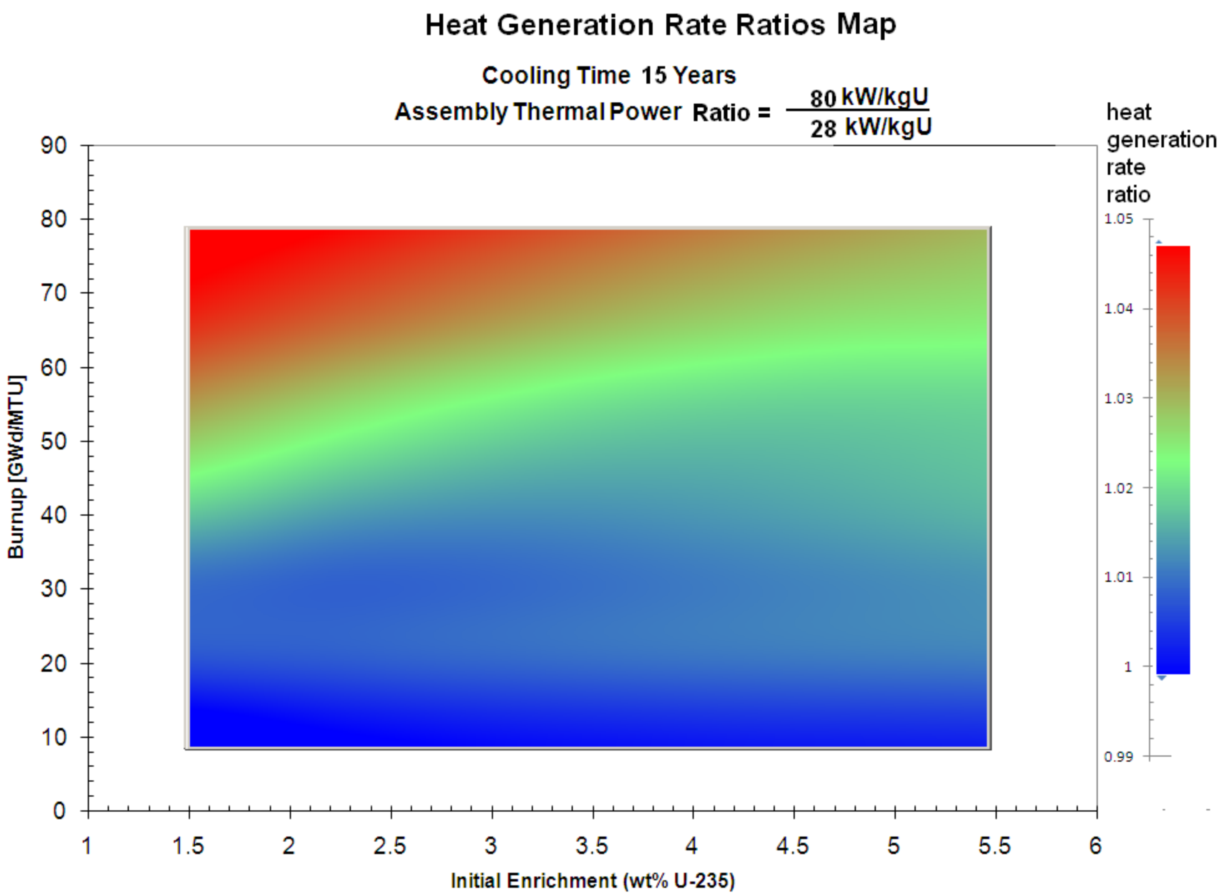

Figure B–11. Heat Generation Rate Ratios Map for the 80/28 Assembly Thermal Power Ratio and for 15-Year Cooling Time

Table B–12. Heat Generation Rate Ratios for 15 × 15 Babcock & Wilcox Assembly With Initial Uranium Loading of 0.46363 MTU for Cooling Time of 25 Years and Specific Power of 80 kW/kgU*									
Burnup, GWd/MTU	**Initial Enrichment, Weight Percent of Uranium**								
	1.5	2	2.5	3	3.5	4	4.5	5	5.5
78.26	*1.038*	*1.035*	*1.031*	*1.028*	*1.026*	*1.023*	*1.021*	*1.020*	*1.019*
70	*1.032*	*1.028*	*1.025*	*1.022*	*1.020*	*1.018*	*1.017*	*1.016*	*1.015*
60	*1.024*	*1.020*	*1.017*	*1.015*	*1.013*	*1.012*	*1.012*	*1.012*	*1.012*
50	*1.016*	*1.013*	*1.011*	*1.009*	*1.009*	*1.009*	*1.009*	*1.010*	*1.011*
40	1.008	1.007	1.006	1.006	1.006	1.007	1.008	1.009	1.010
30	1.004	1.004	1.004	1.005	1.006	1.007	1.008	1.009	1.009
20	1.005	1.006	1.007	1.007	1.008	1.008	1.009	1.009	1.010
10	0.998	1.000	1.001	1.002	1.002	1.003	1.003	1.003	1.004

*$Ratio_{min}$ = 0.998 $Ratio_{max}$ = 1.038
Results for high burnup values 50, 60, 70, and 78.26 GWD/MTU are presented for illustration purpose only (values are in italics; see Section 4.3).

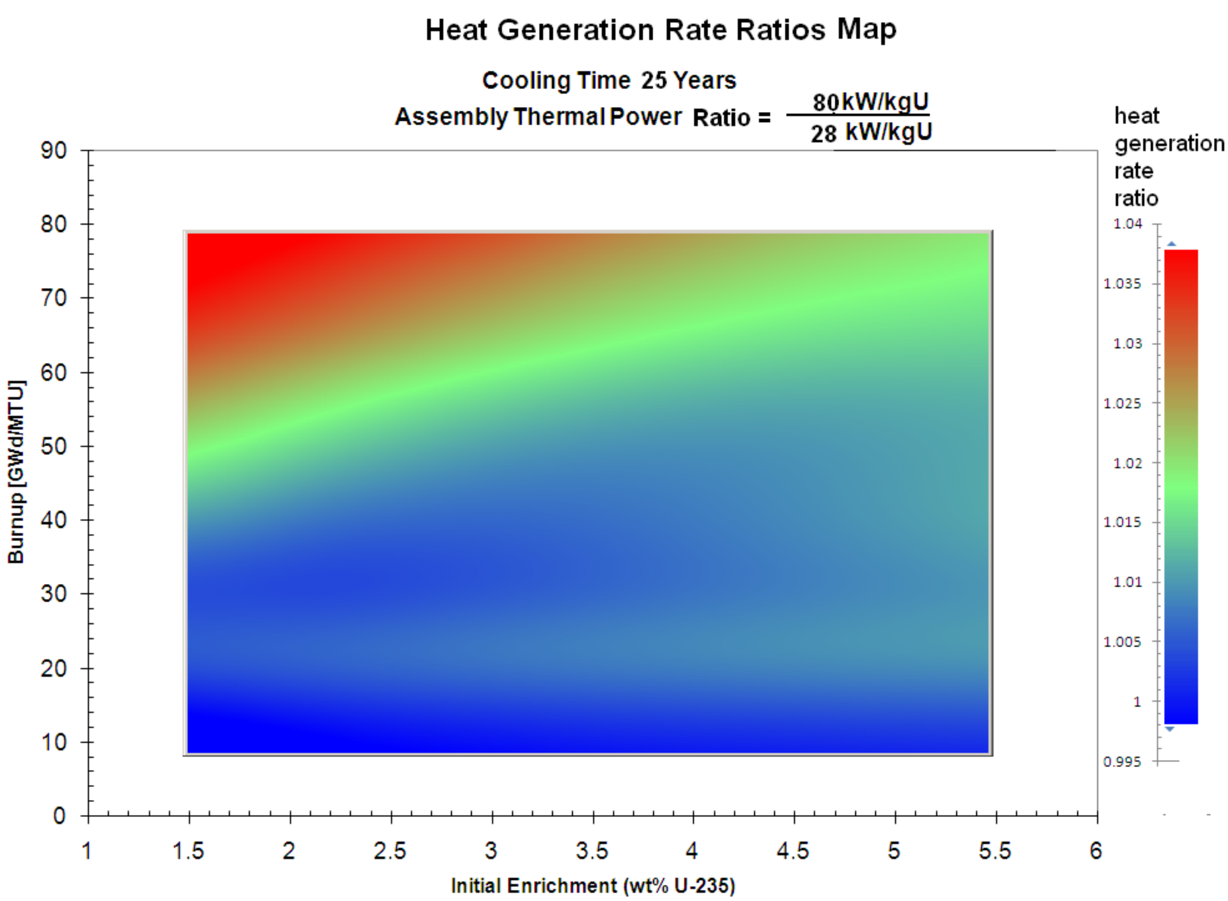

Figure B–12. Heat Generation Rate Ratios Map for the 80/28 Assembly Thermal Power Ratio and for 25-Year Cooling Time