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Impact investigation of reactor fuel operating parameters on reactivity for use in burnup credit applications

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IMPACT INVESTIGATION OF REACTOR FUEL OPERATING PARAMETERS ON REACTIVITY FOR USE IN BURNUP CREDIT APPLICATIONS

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ABSTRACT

Impact Investigation of Reactor Fuel Operating Parameters on Reactivity for Use in Burnup Credit Applications

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When representing the behavior of commercial spent nuclear fuel (SNF), credit is sought for the reduced reactivity associated with the net depletion of fissile isotopes and the creation of neutron-absorbing isotopes, a process that begins when a commercial nuclear reactor is first operated at power. Burnup credit accounts for the reduced reactivity potential of a fuel assembly and varies with the fuel burnup, cooling time, and the initial enrichment of fissile material in the fuel. With regard to long-term SNF disposal and transportation, tremendous benefits, such as increased capacity, flexibility of design and system operations, and reduced overall costs, provide an incentive to seek burnup credit for criticality safety evaluations.

The Nuclear Regulatory Commission issued Interim Staff Guidance 8, Revision 2 in 2002, endorsing burnup credit of actinide composition changes only; credit due to actinides encompasses approximately 30% of exiting pressurized water reactor SNF inventory and could potentially be increased to 90% if fission product credit were accepted. However, one significant issue for utilizing full burnup credit, compensating for actinide and fission product composition changes, is establishing a set of depletion

parameters that produce an adequately conservative representation of the fuel's isotopic inventory. Depletion parameters can have a significant effect on the isotopic inventory of the fuel, and thus the residual reactivity.

This research seeks to quantify the reactivity impact on a system from dominant depletion parameters (i.e., fuel temperature, moderator density, burnable poison rod, burnable poison rod history, and soluble boron concentration). Bounding depletion parameters were developed by statistical evaluation of a database containing reactor operating histories. The database was generated from summary reports of commercial reactor criticality data. Through depletion calculations, utilizing the SCALE 6 code package, several light water reactor assembly designs and in-core locations are analyzed in establishing a combination of depletion parameters that conservatively represent the fuel's isotopic inventory as an initiative to take credit for fuel burnup in criticality safety evaluations for transportation and storage of SNF.

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CHAPTER 1

INTRODUCTION

Nuclear light water reactor (LWR) power plants have been commercially operating for over 50 years. As fuel is burned to generate power, it eventually no longer contains enough fissionable isotopes to effectively produce power; therefore, it is removed from the reactor. Now classified as spent nuclear fuel (SNF), it is placed in a temporary storage pool to cool thermally and radioactively. In recent years, storage pools have reached capacity, requiring power plants to develop dry storage. Eventually, SNF is to be transported and stored at a repository, as mandated by United States (U.S.) Congress in Nuclear Waste Policy Act of 1982. For spent nuclear fuel to be transported and stored, several analyses are completed, including a criticality safety evaluation. Historically, criticality analysis for transportation and storage has utilized a fresh-fuel assumption. This bounding approach assumes fresh (unirradiated) fuel with uniform isotopic compositions corresponding to the maximum allowable enrichment. The fresh-fuel assumption eliminates all concerns of variability related to the fuel operating history and simplifies safety analysis. However, the assumption is a significant conservatism in the system reactivity. The decrease in reactivity due to the irradiation of the fuel is unaccounted for, leading to a large decrease in the capacity of transportation and storage casks and increasing associated costs.

Hence, when representing the behavior of commercial SNF, credit is sought for the reduced reactivity associated with the net depletion of fissile isotopes and the creation of neutron-absorbing isotopes, a process that begins when a commercial nuclear reactor is first operated at power. Burnup credit is the application for which credit for the reduction

in reactivity associated with the change in fuel material composition from irradiating the fuel is sought. The reduction in reactivity that occurs with fuel burnup is due to the change in concentration (net reduction) of fissile nuclides and the production of parasitic neutron absorbing nuclides (i.e., non-fissile actinides and fission products). Credit for the burnup in SNF, within criticality safety evaluation for transportation and storage scenarios, has several benefits, including increased package capacity, enhanced flexibility for system operations, and significant reduction of overall system costs.

Issuance by the Nuclear Regulatory Commission (NRC) in 2002 of Interim Staff Guidance – 8 (ISG-8) Rev. 2 for actinide-only burnup in pressurized water reactor (PWR) analysis was a significant step forward in a multi decade burnup credit study within the U.S. Recommendations in Revision 2 of ISG-8 included limits and assumptions for the licensing basis, guidance on code validation, preparation of loading curves and values, and benefits in reactivity margin beyond that substantiated through the validation process (NRC 2002). However, under this current guidance only about 30% of the current PWR SNF inventory can be transported in high-capacity (32-assembly) casks (Parks et al. 2006). Investigations have been done to demonstrate that the allowable inventory percentage could potentially increase to nearly 90% if burnup credit for actinides and fission products were allowed (Parks et al. 2006).

As for the evaluation of boiling water reactor (BWR) fuel, little has been studied for burnup credit. Benefits of burnup credit for BWR fuel include increase in allowable enrichments to safely accommodate all current and foreseeable assemblies and reduction in costly fixed neutron poison loading in canisters. Unlike PWR burnup credit, increased cask capacity is not as desirable since current designs are capable of accepting

approximately 68 assemblies with assembly-averaged initial enrichments up to 4 wt% ^{235}U (Parks et al. 2006).

As LWR fuel designs advance allowing for higher enrichments and burnups, the application of burnup credit for criticality safety analyses of transportation and storage is important in taking advantage of the benefits burnup credit provides, such as decreased financial burden and operational flexibility. For burnup credit implementation to be most effective, different assembly designs and reactor operational histories should be utilized in establishing a set of depletion parameters that produce an adequately conservative representation of the fuel's isotopic inventory. Depletion parameters have a significant effect on the isotopic inventory of the fuel, and thus the residual reactivity of the SNF. These parameters including fuel temperature, moderator density, and poison concentration are just a few that are impacted in the power generation of the nuclear fuel cycle.

Nuclear Fuel Cycle

Currently in the United States, commercial reactors operate under a once-through fuel cycle. At the end of the power generation life cycle, SNF is categorized as high-level waste (HLW) and is currently being stored temporarily at on-site facilities. As described in Wilson 1996 (Wilson 1996), the LWR nuclear fuel cycle comprises a number of interrelated activities, summarized in Figure 1. Beginning with uranium mining and milling, then the resultant uranium oxide compound is largely refined and converted to uranium hexafluoride (UF_6) for enrichment. Once enriched, UF_6 is manufactured into uranium dioxide (UO_2) fuel pellets and arranged into assemblies for nuclear reactors.

After completing its power generation cycle(s), the assemblies are removed for cooling and eventual storage, reprocessing, or recycling.

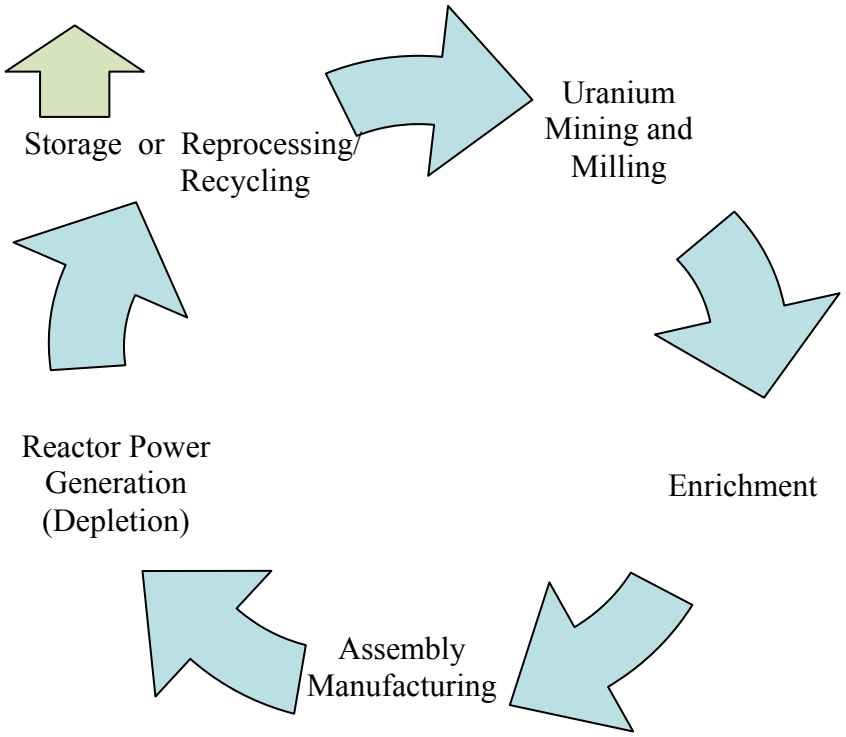

Figure 1 Nuclear Fuel Cycle

LWR fuel assemblies are composed of slightly enriched UO_2 in the form of short, cylindrical fuel pellets stacked to construct fuel rods encased by long, sealed zirconium-alloy cladding tubes. A rectangular array of the rods, with one-quarter symmetry, forms the fuel assembly, for which designs are specific to the manufacturer.

A reactor core is cylindrical and loaded with assemblies in one-eighth symmetry. The loading pattern of a core is designed to maintain a horizontal uniform flux and avoid peaking, which may occur near the center. A PWR consists of a compact core in a pressure vessel capable of containing ordinary water at high pressure. Water serves as the moderator and heat-transfer medium, under a pressure high enough to prevent boiling. The main difference in BWRs is a lower operating pressure, which allows the water passing through the core to boil. Additionally, assemblies contain an internal water channel, which helps maintain the coolant in a liquid state longer before flashing to steam near the top of the reactor.

Upon ending the power generation cycle, fuel assemblies, no longer viable to produce desired power, are removed from the core to storage pools for extended cooling. Now at the end of the nuclear fuel cycle, as LWR spent nuclear fuel, options for storage or reuse require quantification of the spent fuel characteristics for long-term solutions to be managed. The characterization of LWR spent fuel is defined by the reactor operations which occurred during its power generation cycle.

Reactor Operations with Respect to Depletion

The action of regulating and sustaining a chain reaction defines reactor operation. Each nuclide in a reactor system obeys a simple balance equation of the form: net rate of production equals rate of creation minus rate of loss. The chain reaction is evaluated by six factors, which multiplied together compose the effective multiplication factor (k_{eff}), defined as the ratio of the neutrons produced by fission in one generation to the number of neutrons lost through absorption and leakage in the preceding generation (shown in

Equation 1) (DOE 1993). The first four factors are independent of size and shape of the reactor and define the multiplication ability of the fuel and moderator materials, while the second two factors define the leakage of neutrons limited by a finite system of a reactor.

$$k_{eff} = \varepsilon\, p\, f\, \eta\, L_f\, L_t$$

Equation 1

where,
ε = fast fission factor
p = resonance escape probability
f = thermal utilization factor
η = reproduction factor
L_f = non-leakage of fast neutrons
L_t = non-leakage of thermal neutrons

Beginning life as a fast neutron then slowing down to a thermal energy, this life cycle is described by the effective neutron multiplication factors and the interactions with the reactor operation parameters. First, the fast fission factor (ε) is defined as the ratio of the net number of fast neutrons produced by all fissions to the number of fast neutrons produced by thermal fissions. The value of ε is not significantly affected by variables such as temperature, pressure, enrichment, or neutron poison concentrations but rather the fuel-to-moderator ratio (DOE 1993). Since fuel pellets are close-packed, neutrons have a high chance of passing another fuel element without significantly slowing down and generating fast fission. Next, the resonance escape probability (p) is defined as the ratio of the number of neutrons that reach thermal energies to the number of fast neutrons that start to slow down. The value of the resonance escape probability is determined largely by the fuel-to-moderator arrangement and the amount of enrichment of U-235 (DOE 1993). Additionally, changes in the fuel and moderator temperature may increase neutron resonance absorption, hence, decreasing the resonance escape probability. The

third component, the thermal utilization factor (f), is defined as the ratio of the number of thermal neutrons absorbed in the fuel to the number of thermal neutrons absorbed in any reactor material including moderator and poisons. The enrichment of U-235, moderation, and the poison concentration will affect the thermal utilization factor, as well as, temperature effects (DOE 1993). The reproduction factor (η) is defined as the ratio of the number of fast neutrons produced by thermal fission to the number of thermal neutrons absorbed in the fuel. Within the thermal energy range, fission and absorption cross-sections vary with a 1/v relationship, where v is the neutron speed; hence, η changes as U-235 enrichment changes as a result of changing the U-238 concentration and amount of absorption (DOE 1993). Neutrons near the outer edge of the core are likely to escape into the surroundings without propagating the chain reaction. This leakage effect is represented by the two final factors, fast and thermal non-leakage probabilities. The fast non-leakage probability (L_f) is defined as the ratio of the number of fast neutrons that do not leak from the reactor core to the number of fast neutrons produced by all fissions. The thermal non-leakage probability (L_t) is defined as the ratio of the number of thermal neutrons that do not leak from the reactor core to the number of neutrons that reach thermal energies. Both factors are affected by a change in moderator temperature, as an increase in temperature will cause an increased slowing down length and therefore an increased chance of leakage (DOE 1993). These six factors define the chain reaction of the system.

Stability of the chain reaction is essential and controlled by feedback mechanisms that are defined by reactor operation parameters. To quantify the effect on reactivity of the system, a parameter (i.e., fuel or moderator temperature, control rod movement,

neutron poison, etc.) is continuously monitored and controlled to ensure safe and stable operation of the reactor. The specific effects of variations in these parameters are greatly inter-related and characterize the neutron life cycle and depletion of the fuel.

First, in order to even maintain a chain reaction, the moderator-to-fuel ratio (N_m/N_f), is balanced. As the amount of moderator in the core increases (N_m/N_f increases), neutron leakage decreases. This leads to an increase in neutron absorption in the moderator and causes a decrease in the thermal utilization factor (DOE 1993). Having insufficient moderator in the core (N_m/N_f decreasing) causes an increase in the neutron slowing down time and results in a greater loss of neutrons by resonance absorption and will cause an increase in neutron leakage (DOE 1993).

Temperature effects provide two feedback mechanisms. Within the fuel, neutrons may be absorbed by U-238 or induce fission in U-235. If the rate of fission increases, the result is an increase in heat production, therefore an increase in fuel temperature. Raising the temperature of the fuel will raise the resonance absorption in U-238 due to the Doppler broadening effect. The increase in resonance absorption lowers the resonance escape probability, and since the fuel temperature coefficient for resonance escape is negative, the system reactivity will decrease (DOE 1993). The increase in U-238 resonance absorption cross-section, due to Doppler broadening, will also generate more fissile Pu-239, through beta decay of the reaction U-238(n,γ)U-239. Pu-239 is formed as neutrons are absorbed both at resonance and thermal energies; the increased amount of Pu-239 at discharge will increase the reactivity of the spent fuel (Parks et al. 2000a, Lamarsh et al. 2001). For BWR fuel, the reactivity response to fuel temperature has shown to be linear, as fuel temperature increases during depletion the reactivity of the

SNF increases (DeHart et al. 1999). Magnitude of the temperature reactivity effect for BWR fuels is relatively small, as the effect is minimized by the magnitude of the moderator density effects (DeHart et al. 1999). Also in LWRs, when the temperature rises, the water moderator expands, causing the moderator density to decrease, making it less likely for a neutron to be absorbed in the moderator. This reduction in the amount of moderator results in an increase in thermal utilization as moderator temperature increases because a neutron is now more likely to cause fission (DOE 1993).

The pressure applied to the reactor system can also affect depletion by causing changes in reactivity resulting from pressure response changes in the density of the moderator. Decreases in moderator density cause a hardening of the neutron energy spectrum, resulting from an increased Pu production and fission, and a concurrent reduction of U-235 depletion (DeHart et al. 1999).

A rise in moderator temperature increases the energy of thermalized neutrons, making them less likely to cause fission and so reducing the power level (Wilson 1996). As water temperature increases, water density decreases; therefore, neutrons must travel farther while slowing down. This effect increases the probability of leakage and thus decreases the non-leakage probability (DOE 1993). Additionally, the decrease in water density allows more resonance energy neutrons to enter the fuel and be absorbed; therefore, the increase in moderator temperature causes a decrease in the resonance escape probability (DOE 1993).

Absorption of neutrons may be intentional, through use of burnable absorber poisons, or unintentional caused by other core structural materials and fission product generation from decay. As the fuel is burned, burnable poison concentrations must be

reduced to compensate for the negative reactivity effect of burnup. Burnable poisons are materials that have a high neutron absorption cross section that are converted into materials of relatively low absorption cross section as the result of neutron absorption and depletion (DOE 1993). Due to the burnup of the poison material, the negative reactivity of the burnable poison decreases over core life. However, burnable absorber present during depletion hardens the neutron spectrum due to the removal of thermal neutrons through poison capture and displacement of moderator (DOE 1993). This results in a lower U-235 depletion and higher production of fissile Pu isotopes, which increases the reactivity of fuel at discharge and beyond (Parks et al. 2000a). Consequently, SNF assemblies exposed to burnable absorbers will have a higher reactivity than an assembly not exposed to burnable absorbers.

There are two burnable poisons most commonly used in commercial LWRs, soluble poison in the moderator and fixed or integral burnable absorber rods in the assembly. The most common soluble poison in commercial PWRs is boric acid, which is often referred to as "soluble boron". The boric acid in the moderator decreases the thermal utilization factor, causing a negative reactivity (DOE 1993). Rods of neutron-absorbing material are installed in most current LWR fuel designs to provide adjustable control of reactivity. PWRs currently operate with control rods withdrawn or nearly withdrawn from the active fuel region and use soluble boron to control changes in reactivity with burnup (Parks et al. 2000a). Where as it is common in BWRs for control blades to be finely moved within the reactor for flux shaping. The presence of control rods or axial power shaping rods increases the reactivity of burned fuel by hardening the neutron spectrum and suppressing burnup in localized regions (Parks et al. 2000a). The

suppression of burnup in localized regions can lead to axial-burnup distributions characterized by under-burned regions, specifically near the end where leakage is higher. Computational studies, performed by Oak Ridge National Laboratory (ORNL), have shown that if control rods are deeply inserted into the active fuel region for an extended period of burnup, they have a notable positive impact on the reactivity of SNF (Wagner et al. 2003a). Hence it is important to quantify the burnable poison impact based on operational history data.

Radial variations in the neutron flux across the core are due to two main effects. First, leakage is greatest at the core periphery causing burnup to drop off rapidly near the core periphery. Second, since the thermal and higher energy neutron fluxes are greatest at the center of the reactor, fuel is consumed, fertile material converted, and fission product poisons are produced more rapidly in the center region of the core than other parts (Lamarsh et al. 2001). This, has the effect of reducing the flux in the center of the reactor relative to that on the outside, resulting in an increased burnup at the center. To counteract these effects, fuel assemblies are shuffled during refueling, creating a reduced horizontal burnup gradient and enhanced fuel utilization. A reduction or leveling of the horizontal burnup gradient has a decreased effect on SNF reactivity, and is a relative minor effect as compared to other depletions parameters, such as fuel and moderator parameters (Wagner et al. 2003a). Therefore horizontal burnup is not analyzed within the scope of this research.

Axial burnup profiles are dependent on fuel assembly design, burnup, and operating conditions. It is commonly defined by highly burned center region and lower burned end regions due mainly to leakage. As a reactor operates, the axial variation of

the flux profile shifts with partial length absorbers, varied control rod/blade insertions, non-uniform axial enrichment loadings, and for BWRs axially and time varying moderator density, among other parameters (Parks et al. 2000a). An axial-profile database evaluated by ORNL demonstrates the application of axial burnup profile statistically bounding outliers can conservatively represent the fuel profile for criticality safety analyses of storage and transportation scenarios; however, ORNL recommends updates to the database as longer burned and high enriched fuel operation data becomes available to ensure outlying data points are still bounding (Wagner et al. 2003b). The complexity of axial burnup profiles is not evaluated in this research.

Upon shutdown of a reactor, stopping the chain reaction generates decay heat and also has significant by-products. Fission products are of concern in reactors primarily because they become parasitic absorbers of neutrons which generate majority of the radioactivity of discharged nuclear fuel, and result in long term sources of heat. Although some are stable, the majority are radioactive with half-lives ranging from a fraction of a second to thousands of years.

The decay heat generation rate diminishes to less than 1% of the thermal rating of the reactor approximately one hour after shutdown (DOE 1993). However, even at these low levels, the amount of heat generated requires the continued removal of heat for an appreciable time after shutdown. Decay heat is a long-term consideration and impacts spent fuel handling, reprocessing, waste management, and reactor safety.

Cooling time is important in consideration of the decay heat and radiation source terms. Upon discharge, reactivity of the SNF will increase for approximately 100 hours due to the decrease in neutron absorption caused by the decay of very short-lived fission

products (Wagner et al. 2003a). Afterwards, reactivity decreases to approximately 100 years, which is driven by the decay of the ^{241}Pu fissile nuclide and the buildup of the neutron absorbers ^{241}Am and ^{155}Gd (Wagner et al. 2003c). Then reactivity slowly increases to a second peak around 30,000 years; this slow increase is governed primarily by the decay of two major neutron absorbers, ^{241}Am and ^{240}Pu, and mitigated by a decrease in the fissile inventory as ^{239}Pu decays and causes in increase in ^{235}U (Wagner et al. 2003c). After 30,000 years reactivity decreases as the decay of ^{239}Pu dominates the process.

The reaction and inter-action of operation parameters, including but not limited to fuel temperature, moderator density, moderator temperature, and burnable poison content, as used to operate a reactor define the depletion of the fuel. The main effect of depletion is the hardening of the neutron spectrum, which increases the reactivity of the fuel at discharge. Evaluations of depletion parameters are to characterize parameter impacts on nuclear criticality safety for SNF management.

Spent Fuel Management

There has been a renewed interest in nuclear energy with demands not only for the future but for final solutions of such issues like nuclear spent fuel management. Spent fuel management is a common and costly activity for all operators of nuclear power plants. Per the Nuclear Waste Policy Act of 1982, the U.S. Congress has mandated the disposal of HLW in a geological repository serving as an end-of-cycle solution, allowing for eventual or possible reprocessing of the HLW. As delays are incurred in implementing reprocessing and in plans for geologic repositories, spent fuel

storage for extended durations has become a progressive reality. In the short-term, HLW is stored in spent fuel pools as well as interim dry storage casks at reactor plant sites. With new fuel and material designs, spent fuel storage technology, with focus on dry interim storage, is undergoing evolution. Economic considerations in spent fuel storage projects rise in importance as spent fuel storage quantities increase. Implementation of burnup credit offers the possibility to increase realism in analyses.

Transportation and storage casks are utilized for transporting between and use during short and long term storage. Several safety criteria exist for casks, including: cask load limits, criticality safety evaluation, and other requirements for shielding, mechanical, thermal, operation, etc. These criteria provide a safety barrier, however the restrictive levels for analyses are what derail the economics of waste management. Therefore it is important that regulatory guidance follow the research results to allow for more realistic designs and waste management specifications.

Regulatory Guidance

Per regulatory guidance, the most common assumption for criticality safety analyses of dry transportation and storage scenarios of SNF is based on a fresh, unirradiated fuel with uniform isotopic compositions corresponding to the maximum allowable enrichment. Although, this approach ensures maximum conservatism, it is unrealistic and has several drawbacks, including decreased cask capacity, higher costs, and over engineered designs.

Currently under another revision, in September 2002, ISG-8 Rev. 2 was issued providing a step towards regulatory guidance that enables industry to effectively proceed

with design and licensing of a burnup-credit cask. ISG-8 provides recommendations for the use of burnup credit with PWR spent fuel in transport and dry storage casks, including guidance on criteria to determine whether SNF is eligible for burnup credit consideration, the experimental data needed and the general approach to take for establishing the bias and uncertainty in the analysis codes, modeling assumptions to consider in performing analyses for the safety basis, and loading operations (NRC 2002).

Final recommendations within the ISG-8 Rev. 2 (NRC 2002) limit the burnup credit to that available from actinide-only nuclides for SNF with an assembly-average burnup of 40 Gigawatt-days per metric ton heavy metal (GWd/MTHM) or less and a cooling time of five years. Although burnup values of greater than 40 GWd/MTHM may be loaded into a cask, only burnup to 40 GWd/MTHM may be credited in criticality safety analysis (NRC 2002). Initial enrichments up to 5.0 wt% ^{235}U are allowed, but a burnup loading penalty is required for enrichments above 4.0 wt% (NRC 2002). The loading offset accounts for the lack of measured data for assemblies above 4.0 wt% initial enrichment. An "adequate representation of the physics" is also recommended for analysis, as the axial and horizontal burnup profile varies within a spent fuel assembly (NRC 2002). Additionally, assemblies with burnable absorbers are not allowed (NRC 2002). Analysis methods for calculating k_{eff} and isotopic compositions should be verified against measured data for validation (NRC 2002). Potential uncertainties created by the lack of physical data and variability of operating history need to be quantified and/or bounded in safety analysis (NRC 2002). Difficulties in accommodating these recommendations, pose large costs and time for cask design companies. Although additional modifications are required to further the acceptance of burnup credit, ISG-8

has a set a precedence showing that an adequate safety margin can be maintain while still accrediting reactivity decreases from burnup for transportation and storage of SNF.

American National Standard Institute and American Nuclear Society identify ANSI/ANS-8.27-2008 as guidance on burnup credit for LWR fuel. The standard states that for validation of the depletion and decay analysis, adequacy is demonstrated by comparison to measured data (ANSI/ANS 2008). Considering the number of SNF assemblies located around the country, it is not a feasible option to measure reactivity of all assemblies, especially since there is a high background radiation. Current regulations are moving towards risk-informed, performance-based licensing strategies which makes an expansion of analysis and verification of fuel exposure history feasible. However, obtaining the detailed operating histories needed to model all LWR fuel assemblies to which burnup credit would be applied is a tedious and costly task.

A bounding set of depletion parameters needs to be developed that would be acceptable for transport and storage, where each assembly design would be used as the bounding model that would produce conservative effective multiplication factor values for assemblies in the commercial SNF inventory. Comparison of the bounding models to an up-to-date database of operating history conditions could represent a good indication that of a sufficient conservatism in criticality safety analyses for storage and transportation of SNF.

Research Objectives

One of the significant issues yet to be resolved for using burnup credit for SNF is establishing a set of depletion parameters that produce an adequately conservative

representation of the fuel's isotopic inventory. Depletion parameters (such as fuel temperature, moderator density, burnable poison rod history, and soluble boron concentration) can have a significant effect on the isotopic inventory of the fuel, and thus the residual reactivity.

ISG-8 Rev. 2 (NRC 2002) states the following:

"… a value of k-effective for the licensing safety basis should be calculated using fuel design and in-reactor operating parameter values that appropriately encompass the range of design and operating conditions for the proposed contents."

This research quantifies the reactivity impact on a system from dominant depletion parameters (i.e., fuel temperature, moderator density, burnable poison rod history, and soluble boron concentration). Bounding depletion parameters are developed by statistical evaluation of a generated database containing operating history data from several U.S. commercial nuclear power plants of several LWR designs. The database was generated from summary reports of commercial reactor criticality data. Through depletion calculations in conjunction with a criticality evaluation, utilizing the SCALE 6 code package, several assembly designs and in-core locations are analyzed in establishing a combination of depletion parameters that conservatively represent the fuel's reactivity as an initiative to take credit for fuel burnup in criticality safety evaluations for transportation and storage of SNF.

The database of reactor operating history includes information such as fuel temperature, moderator density, moderator temperature, burnable poison rod history, and soluble boron concentration. A statistical analysis of the various parameters is completed utilizing the Minitab software. Utilizing the statistically determined bounding values,

each parameter is evaluated independently and collectively through depletion calculation. The k_{eff} of the systems are compared to quantify the effects of each parameter independently and collectively. The outcome of the analysis recommends representative values for each parameter that will produce bounding, most reactive values for burnup credit criticality safety evaluations.

CHAPTER 2

LITERATURE RESEARCH REVIEW

In an effort to credit burnup of SNF, two significant types of research include measurement and computations methods. Often to minimize uncertainties, these two research methods are used interchangeable to verify and/or validate one another. Research in the realm of burnup credit, has been conducted domestically and internationally for several decades. With an initial attempt to reduce the large conservatism in the calculated value of the system reactivity implicated by the "fresh fuel" assumption, research methods have advanced in many aspects, including reviewing current SNF inventories with measurements and determining bounding depletion parameter values with computational tools. All research seeks to take credit for the burnup of SNF, hence reducing the analysis conservatism while maintaining an adequate criticality safety margin while providing flexibility and cost savings.

ISG-8 Rev. 2 ties acceptance of burnup credit methodology to a verification by measurement of the burnup of each assembly before loading, with the requirement to adjust the verified reactor record burnup value by a combination of the uncertainties in the reactor value and the measurement (NRC 2002). Often this is considered an unnecessary disadvantage, in terms of impact on operations and costs, to the implementation of burnup credit, given the very low probability of transportation accidents with the potential for re-flooding of the cask cavity, and the very low conditional probability of a critical configuration assuming that re-flooding occurs. Electric Power Research Institute (EPRI) evaluated two types of misloadings: loading of fresh fuel and loading of assemblies with less burnup than required (Wells 2003).

However unlikely or readily detectable through audit procedures, the potential misload events showed to have no safety consequences, as reactivity increase does not lead to a critical configuration (Wells 2003). Additionally, cask size does not significantly alter the reactivity effects of misloading with under-burned spent fuel (Wells 2003).

Operational experience indicates that reactor records provide reliable information with regard to assembly burnup characteristics. Studies sponsored by EPRI were performed to quantify the magnitude of uncertainties that can be present in burnup value estimates by a PWR utility for their discharged fuel (EPRI 1999). Examination of measured data for a typical Westinghouse PWR revealed that the uncertainty in the assembly average burnup in instrumented locations was approximately 2.5% at most (EPRI 1999). This value was derived from a direct comparison between measurement and calculation of the burnup in instrumented locations for all three cycles evaluated. Commercial company COGEMA's La Hague Reprocessing Plant experience indicates that the value of the burnup as measured at La Hague and the value of the burnup as reported by the French operators differ by a mean average of 5%, which is consistent with the EPRI study (IAEA 2002). In conclusion, a high precision burnup measurement does not seem to be required for safety purposes and burnup verification for compliance with technical specifications could be met with audit procedures. Hence, operational data is deemed acceptable for computational evaluations of fuel depletion.

Current data of assay measurements are minimal, and fuel development has surpassed the limitations of older fuel. Hence an obstacle exists between available measured data as it compares to engineering calculations. Present experimental database of public domain actinide assay data consists largely of samples from older fuel assembly

designs with enrichments below 3.5 weight percent (wt%). A single measurement contains fuel above 3.4 wt%, but it has low burnup of 12 GWd/MTHM (Parks et al. 2000b). Only seven of approximately 50 samples had burnable poison rods (BPR) present during irradiation (Parks et al. 2000b). ISG-8 provides recommendations on the use of measurements to confirm reactor records, as to represent a sampling of assemblies. Hence the measurement technique must provide accuracy and precision of the representative measured values. Research has been conducted for the various burnup characterization techniques to quantify the uncertainties in calibration and measurement conditions. Dry fuel measurements have shown an accuracy of fuel mass determination in the range of 8%, while wet fuel measurements of burnup as compared to declared values differed by 3-5% (Simpson et al. 2006). BIL Solutions Inc. shows that operational experience proves key in the development and demonstration of spent fuel measurements (Simpson et al. 2006). However, it is necessary to analyze burnup credit through an engineering approach with computational tools to compensate for the lack of measured data.

A conceptual high-capacity (32-assembly) cask, designated generic burnup credit cask (GBC) – 32, was developed by ORNL, utilizing the SCALE code package, to provide a reference burnup credit cask design for use in establishing the effectiveness of ISG-8 and demonstrating potential benefits that might be gained with negative reactivity credit from actinides and fission products for PWR fuel (Parks et al. 2004). The generic cask design includes features from several U.S. cask vendors' designs (e.g., similar canister inside diameter and Boral™ for fixed neutron poison), as well as features from an internationally specified benchmark cask, and will accommodate 32 PWR fuel

assemblies. Burnup credit within the cask design is evaluated as a function of several parameters to determine the conservative direction for that parameter. Evaluated parameters include fuel temperature, moderator temperature, soluble boron concentration, operating history, specific power, fixed absorbers; the bounding conditions that produce maximum reactivity when examined with all other nominal conditions are highest fuel temperature, highest moderator temperature, highest boron concentration, high power late in life (actinide-only burnup credit), high specific power (actinide-only burnup credit), and the presence of fixed absorbers during depletion (Wagner et al. 2003a). All trends are related to spectral hardening, with the exception of specific power/operating history effects. The results showed that the increase in k_{eff} associated with the use of bounding parameters increases with burnup and decreases with initial fuel enrichment (Wagner et al. 2003a). The GBC-32 burnup cask assessment revealed the major component that would improve the accuracy of burnup credit analyses is the inclusion of fission products (Wagner et al. 2003a). Additional reactivity reductions may be achieved through optimization of cask design and utilization characteristics, such as assembly separation, poison loading, and the use of assembly inserts.

Assemblies that cannot be accommodated in a 32-assembly cask are transported in a 24-assembly cask. Inclusion of fission product burnup credit could potentially reduce the number of shipments by about 22%, as compared to a reduction of about 8% for actinide-only burnup credit as recommended by ISG-8 (Parks et al. 2004). It can be calculated that 315 shipments of PWR SNF to the repository will be eliminated using actinide-only burnup credit; however, an additional 625 shipments could be eliminated if burnup credit accounted for fission products (Parks et al. 2004). These values are

determined based on the percentage of total metric tons of heavy metal (MTHM) from PWRs as of the end of 1998 (64%), the average number of PWR assemblies per MTHM (about 2.33 PWR assemblies/MTHM), and the current 70,000 MTHM repository capacity limit established in the Nuclear Waste Policy Act; these values are utilized to estimate an approximate 100,000 PWR assemblies to be transported to the repository (Parks et al. 2004). Inclusion of fission products will only account for a very small percentage of negative reactivity, as actinide burnup accounts for approximately 2/3 of the negative reactivity (Parks et al. 2004). However, the slight downward shift of the actinide-only loading curve associated with the inclusion of fission products will drastically increase the amount of assemblies acceptable for high-capacity cask loading (Parks et al. 2004).

As seen in Table 1, the effectiveness of ISG-8 based on actinide only burnup credit is minimal. While it shows to be most effective for the Combustion Engineering designs, those designs only represent approximately 28% of the total SNF inventory of discharged data from U.S. PWRs through the end of 1998 (Parks et al. 2004). However, the inclusion of fission products and removal of burnup and enrichment limits can drastically increase the inventory of SNF assemblies available for burnup credit cask storage and transportation, as seen in Table 1, by an almost 40% increase in viable SNF for high-capacity storage.

Table 1 SNF Assemblies Viable for Conceptual Burnup Credit Cask

Assembly type	% acceptable for GBC-32 based on actinide only burnup	% acceptable for GBC-32 based on actinide + fission product burnup
Combustion Engineering 14x14	77%	95%
Babcock & Wilcox 15x15	3%	33%
Combustion Engineering 16x16	62%	99%
Westinghouse 17x17	11%	58%
Total	27%	65%
Reference	(Parks et al. 2004)	(Wagner et al. 2003a)

Benefits of burnup credit include large financial savings. Utility companies with PWRs that are shutdown prior to resolution of interim storage or permanent disposal must decide what to do with their spent fuel. High-capacity casks, approved for actinide-only burnup credit, would decrease the number of casks required to dispose of all SNF at a shutdown PWR and hence decrease costs. TRW Environmental Safety Systems, Inc. in conjunction with Maine Yankee Atomic Power Company computed a potential net savings of $5.5 million for the utility and $1.3 million for the Department of Energy (DOE) through use of the burnup credit Holtec Hi-Star 100/Multi Purpose Canister (MPC) -32 high-capacity cask instead of a typical 24 assembly cask (Lancaster et al. 1998); these cost savings represent a single power plant of the approximately 100 power plants operational today. At the time of the Maine Yankee cost analysis the Holtec Hi-Star 100/MPC-32 high-capacity cask was still under development. In 2008, the final safety analysis report for the Holtec Hi-Star 100/MPC-32 high-capacity cask was approved and issued, allowing storage and transportation efforts involving burnup credit to proceed. Although mostly reserved as propriety data, the safety analysis report displays the complexity of design, maintenance, and operation of a burnup credit cask.

CHAPTER 3

METHODOLOGY

Operating History

In order to establish justifiable assumptions for a burnup credit criticality safety evaluation, the effects on the reactivity of variations in reactor operating conditions and fuel assembly design characteristics of SNF shall be quantified. Using operating histories to accurately represent the values of the depletion parameters, independent and collective impacts on SNF system criticality are determined.

Operating histories from commercial reactors required for performing analytical commercial reactor criticality (CRC) analyses have been prepared by the U.S. Department of Energy Office of Civilian Radioactive Waste Management for the Yucca Mountain site characterization project. Fuel assembly and reactor core follow data, including statepoint and datapoint measurements of several operation parameters, are included in the CRC summary reports. Depletion parameters, fuel temperature, moderator specific volume, and burnup, are provided for axial nodes along several assemblies within the core for several cycles. Additional data includes fuel design specifications, burnable poison presence, cycle lengths, and moderator temperature among other parameters. The commercial reactor criticals included an evaluation of the assembly isotopic inventory as well as the reactivity calculations for each statepoint. The reactivity calculations included isotopic and cross section uncertainties, and have been shown to have similar neutron spectra to spent fuel containers and transport casks (Wells 2010). Thus the endorsed use of the CRC data for burnup credit analyses.

The operating history data for several fuel designs of varying commercial power

plants has been compiled into Minitab, statistical analysis software, for evaluation and statistical analysis. Plant data represented in this research is shown in Table 2. Operational histories are compiled based on LWR type; hence all PWR and BWR data are reviewed separately. Two PWR fuel designs are evaluated, including Babcock and Wilcox 15x15 pin array design and Westinghouse 17x17 pin array design. While, the BWR fuel design is representative of the General Electric 8x8 pin array design.

Table 2 Represented Commercial Nuclear Power Plants

Power Plant	Reactor Type	Assembly Design	Pin Array Size	Reference
Sequoyah Unit 2	PWR	Westinghouse	17x17	(Mays 1998a)
Davis-Besse Unit 1	PWR	Babcock & Wilcox	15x15	(Mays 1998b)
LaSalle Unit 1	BWR	General Electric	8x8	(Henderson 1999)

Depletion parameters quantified include: fuel temperature, moderator density, moderator temperature, and poison inclusion (i.e., control rods, BPRs, or soluble poison). Measurements of fuel temperature and moderator specific volume are provided for axial nodes along several assemblies in the core for several statepoints and datapoints. The axial nodes represent segmented lengths along the vertical axis of the fuel assemblies. The statepoint measurements represent a critical point in the cycle for a k_{eff} of one, and datapoints are used to provide data for assemblies measured that were inserted prior to the statepoint measurements. The BWR data represented by LaSalle Unit 1 was compiled for 11 statepoints and datapoints represented by 790 assembly measurements for 25 axial nodes each. PWR data is represented by Davis-Besse Unit 1 compiled for 9

statepoints and datapoints represented by 181 assembly measurements for 18 axial nodes each and Seqouyah Unit 2 compiled for 5 statepoints and datapoints represented by 103 assembly measurements for 16 axial nodes each. Data represented by the initial insertion of the assembly into the core is excluded from the generated database due to burnup and fuel temperature measurements not available yet, since the fuel is fresh and un-depleted. In order to maintain consistent datapoint measurements, moderator specific volume measurements provided at the fresh assembly initial insertion are also excluded.

All operation history data for fuel temperature is shown in Figure 2 for PWR assemblies and Figure 3 for BWR assemblies. All operation history data for moderator density, calculated as the inverse of the specific volume, is shown in Figure 4 for PWR assemblies and Figure 5 for BWR assemblies. Basic statistics of all data, including the average and one and two standard deviations (sigma) in the conservative direction of the parameter, are shown on the corresponding figure for an overview. Although statistics (i.e., average and sigma) on the figures show the general trend of the data, bounding values including the maximum, minimum, and average were used in the evaluation of the depletion parameter impact to ensure a conservative selection. The data represents measurements of the parameter from beginning of cycle, through, and to end of cycle for several cycles and for all axial nodes.

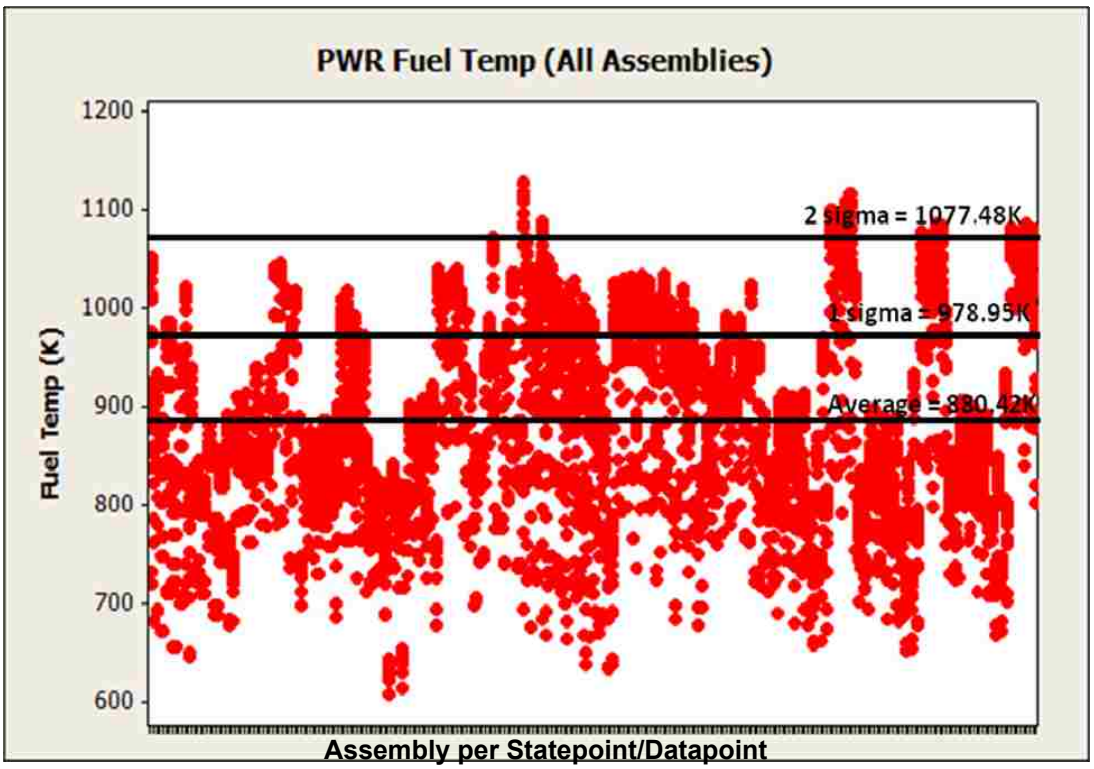

Figure 2 PWR Fuel Temperature per Statepoint/Datapoint

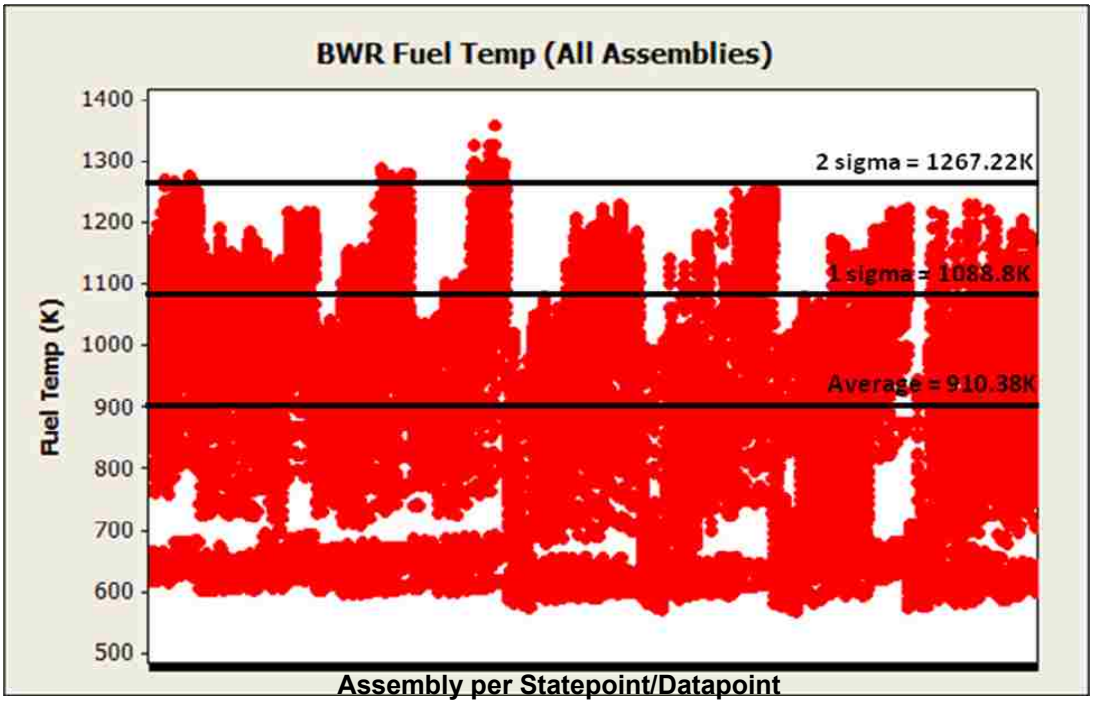

Figure 3 BWR Fuel Temperature per Statepoint/Datapoint

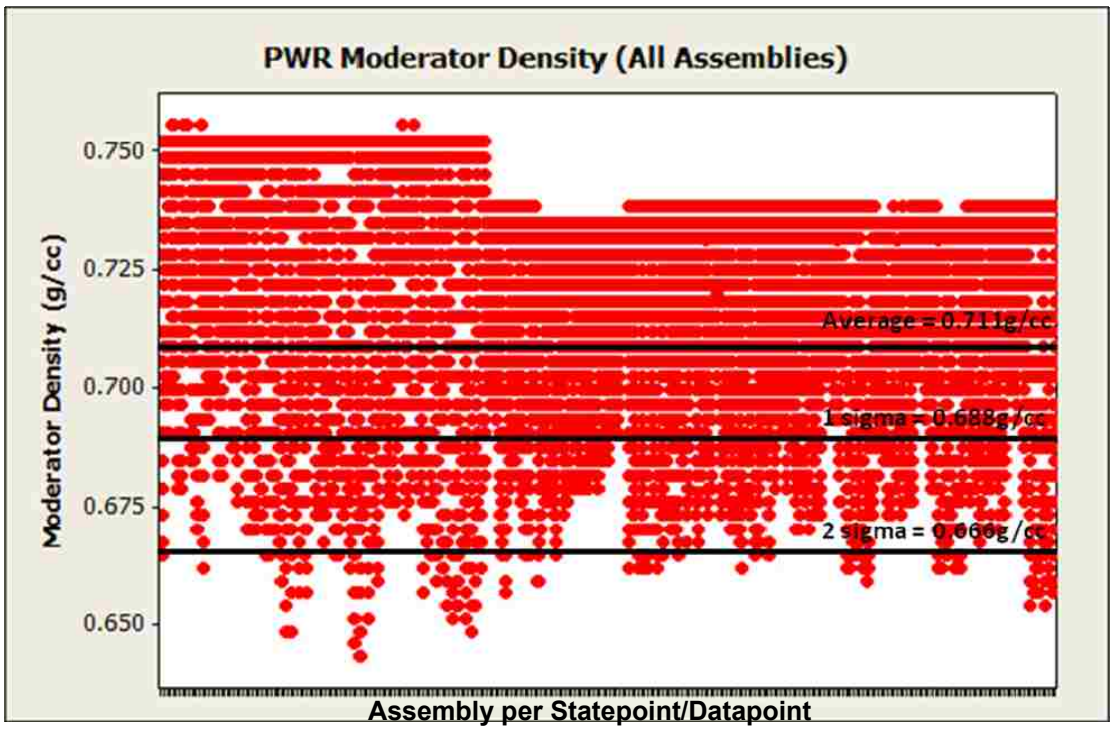

Figure 4 PWR Moderator Density per Statepoint/Datapoint

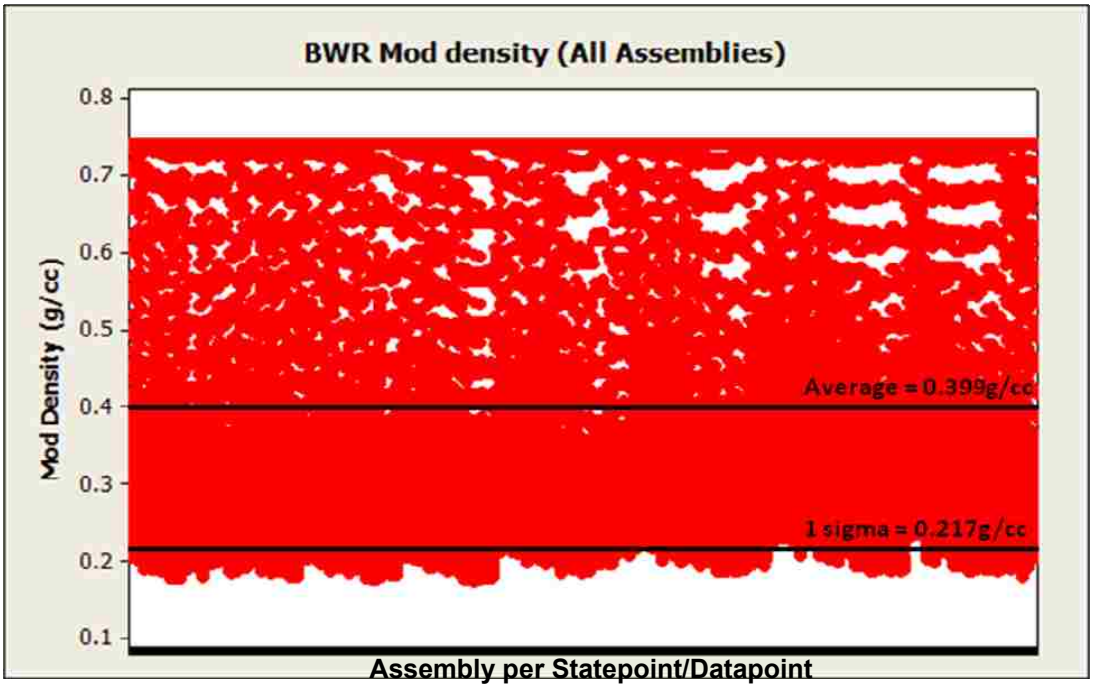

Figure 5 BWR Moderator Density per Statepoint/Datapoint

Statistical Analyses and Selection of Depletion Parameters

Utilizing the generated database in Minitab for all operational histories, basic statistics were computed for the depletion parameters of large datasets, including fuel temperature and moderator density. Additional depletions parameters, including burnable absorber presence and moderator temperature, are evaluated. However, these depletion parameters of importance did not have sufficient detail in the summary reports; hence a technical justification for the selected value is provided. Other parameters utilized for depletion calculations, such as cycle length and decay or cooling time, are quantified and justified by CRC data and other resources.

Depletion parameters fuel temperature and moderator density are evaluated at the average and maximum and/or minimum values statistically determined from the operating histories. Only the conservative direction of each depletion parameter is used, for example fuel temperature is evaluated at the average and maximum. Operation data for the depletion parameters was tabulated for the entire height of the rod, as quantified by the axial nodes. The statistics are performed for all data points for the entire length of the rod, but depletion calculations are based on two-dimensional (2-D) model representation. Hence, for any depletion parameter, the average accounts for the maximum and minimum along the height of the rod. The resultant statistical value for the PWR data and BWR data are shown in Table 3 and Table 4, respectively.

Table 3 PWR Operation History Depletion Parameter Statistics

Parameter	Min	Ave	Max
Fuel Temp (K)	--	880.42	1128.87
Moderator Density (g/cc)	0.6433	0.7109	--

Table 4 BWR Operation History Depletion Parameter Statistics

Parameter	Min	Ave	Max
Fuel Temp (K)	--	910.38	1358.4
Moderator Density (g/cc)	0.1741	0.399	0.7398

The moderator temperature for the measured critical conditions is provided in the CRC data for cycle statepoints. The average of the statepoint measurements is used to represent the average moderator temperature of the system. However, as fuel temperature increases the moderator temperature increases and the moderator density decreases. Due to this effect, a maximum moderator temperature is evaluated with the maximum fuel temperature and minimum moderator density parameter combination. For PWRs, a maximum moderator temperature of 600 Kelvin (K) is used for comparison, as recommended by ORNL to represent the maximum core outlet temperature (Parks et al. 2000a).

As for BWRs, the moderator temperature changes very little along the upward axial flow direction as the moderator boiling (i.e., voiding) increases (Parks et al. 2000a). Therefore, as voiding varies along the fuel, the moderator impact is controlled by the moderator density, hence, the moderator density for the BWR analysis is evaluated at the minimum, average and maximum and the moderator temperature is maintained at an

average value calculated from CRC data. The modeled average and maximum moderator temperatures are summarized in Table 5.

Table 5 Moderator Temperature Parameter

Fuel Type	Ave	Max [a]	Reference
Westinghouse 17x17	559.3 K	600 K	(Mays 1998a)
Babcock & Wilcox 15x15	551.1 K	600 K	(Mays 1998b)
General Electric 8x8	354.3 K	--	(Henderson 1999)

Reference: [a] (Parks et al. 2000a)

The variability in control rod usage is difficult to estimate in a generic sense. For PWRs, control rod presence data is provided for a few cycles, cycles one through three for Westinghouse 17x17 fuel and one through five Babcock & Wilcox 15x15 fuel. Evaluation of the PWR summary report data shows a preference toward last cycle control rod presence for an assembly producing power for two or three cycles (Mays 1998a, Mays 1998b). Some CRC data is provided for the initial core load, power plant cycle one. Control rod data for cycle one is neglected due to the unique core load for start up of a reactor, which includes large quantities of low average enrichment assemblies and primary source rods. Therefore, judgment of "typical" control rod insertion patterns is based on a presence at end of life or present in the last cycle of a three cycle power generating term. As for BWRs, control blades are not inserted directly into the assembly, instead they are a cross shape that is located between four assemblies in a cell, as shown in Figure 6. Control blade data is included for cycles four through seven, where the use of control blades has a fine movement capability of 3 in. notches (Henderson 1999).

Although the notch control allows for variability in the length of the rod insertion, since depletion evaluations are a 2-D scale, only control blade insertions beyond half-way are accounted as inserted. Review of the BWR control blade history shows a "typical" preference toward presence at end of life or present in the last cycle of a three cycle power generating term (Henderson 1999).

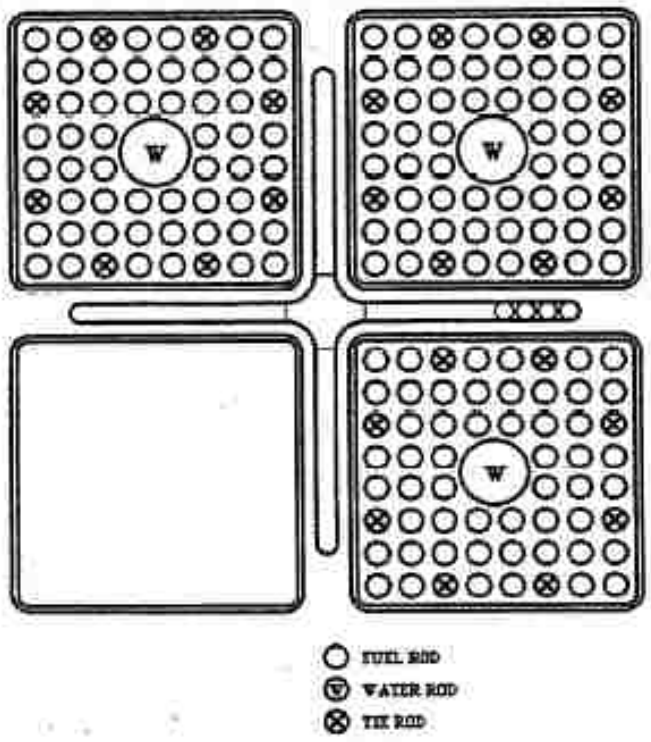

Figure 6 BWR Control Blade Cell (Henderson 1999)

Soluble boron is the preferential burnable absorber for flux control in PWRs. The solution is inserted into the moderator based on a concentration of parts-per-million

(ppm). Soluble boron concentration is conservatively modeled as a constant average for the entire cycle, versus the introduction of boron to the system over time, reduced throughout the cycle as less flux suppression is needed due to burnup. The representation of a constant boron modeling has significantly more boron present late in burnup, hence resulting in slightly more reactive isotopic compositions for discharged SNF (Wagner 2003d).

Summary reports provide boron letdown data per cycle for PWRs as a linear regression equation fit to core operation histories for each cycle data provided. The equation is in the form A+Bx, where A is an initial concentration and B is the slope of the line defined by the reduction in boron concentration per effective full power day. Using the critical boron data per fuel design, the cycle averaged boron concentration is calculated for each cycle and then averaged, which is then utilized to represent the average boron concentration constant over the entire cycle. The maximum boron concentration of 750 ppm is a referenced value, recommended by ORNL to be adequately bounding based on studies performed (Parks et al. 2000a). Additionally, the boron concentration is modeled dependent on the cycle length, adjusted by a needed increase in burnable poisons to compensate for the increased burn length. Hence, as the cycle length is increased to 18 months, used as a representative case, the ratio of cycle lengths is applied to the boron concentration. Therefore, for the 18 month cycle length the boron concentration is multiplied by 1.5 from the 12 month cycle length. Soluble boron concentrations, modeled as a cycle averaged input, are displayed in Table 6.

Table 6 PWR Cycle Averaged Soluble Boron Concentration Statistics

Fuel Design	Cycle Length	Ave Boron Conc. (ppm)	Max Boron Conc. (ppm)	Reference
Westinghouse 17x17	~ 12 months	545.05	750 [a]	(Mays 1998a)
	18 months	817.57	1125.0	1.5 times 12 month cycle
Babcock & Wilcox 15x15	~ 12 months	592.1	750 [a]	(Mays 1998b)
	18 months	888.15	1125.0	1.5 times 12 month cycle

Reference: [a] (Parks et al. 2000a)

BWRs do not use soluble boron in the moderator since BPRs are a common design feature in every assembly. The BPR, also known as Gad rod, is a fixed burnable absorber integrated in the fuel rod, represented by a mixture of uranium oxide and gadolinium oxide (Gd_2O_3). The weight percent (wt%) of gadolinium oxide is defined by the fuel designers and depends upon several items, including the bundle U-235 enrichment and the number of BPRs. The CRC data provides a summary of the average axial enrichment and the number and weight percent of Gd_2O_3 for each axial zone (Henderson 1999). The quantity of BPRs and weight-percent Gd_2O_3 are selected based on a maximum and minimum representation from the CRC data. The weight-percent Gd_2O_3 of the mixture is based on the maximum amount for the number of rods represented. Table 7 displays the BPR data used in the depletion analyses. In assembly locations of BPRs are selected based on an assessment of Gad rod worth in a lattice for a fresh BWR fuel shipping package. (Sloma et al. 2009). The highest worth locations are important to depletion calculations as the larger absorption of thermal neutrons will harden the spectrum, produce more Pu, and hence increase the reactivity of the SNF.

Therefore reviewing the results of worth based on Gd-157, the largest thermal absorption cross-section of the Gd_2O_3 mixture, locations of highest worth are used for BPR positions in the depletion calculations (Sloma et al. 2009). Figure 7 displays the selected BPR positions in the lattice for minimum and maximum loading.

Table 7 BPR Summary for BWR Fuels (Henderson 1999)

Fuel Design	Case	No. of BPRs	wt% Gd_2O_3
General Electric 8x8	Max	12	5
	Min	7	5

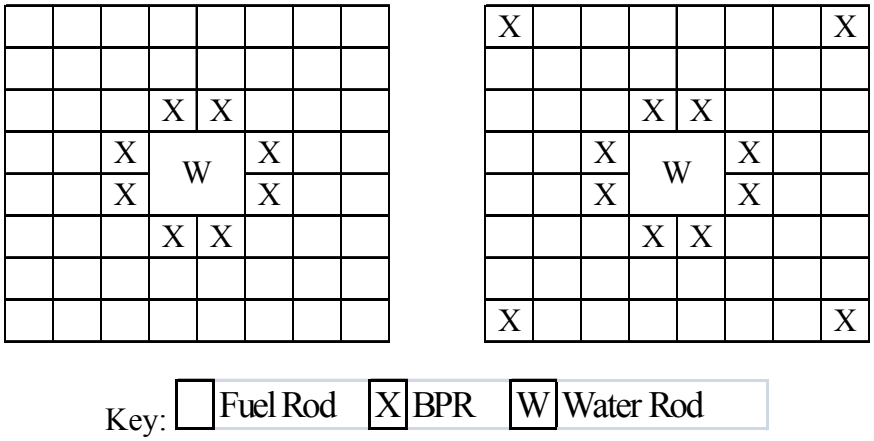

Figure 7 BPR Loading for BWR Analyses: 8 BPRs (left), 12 BPRs (right)

Cycle lengths vary with fuel design, plant design, and operational preference. The CRC data represented here is from plant operations occurring in the 1980's, thus fuels were burned for shorter cycles than current plant operations. Cycle length data, represented by effective full power days, is available for more than the operating histories

data provided. Therefore, an average of all cycles is calculated to represent the operating data. Although approximately 12 months in length, using the actual cycle length data provides a more realistic representation of the SNF, while an additional evaluation for a 18 month cycle is used to represent current cycle lengths. Note the average cycle length for all operation histories was approximately 12 months, hence, any reference to a 12 month cycle is actually modeled in the depletion calculations by the plant cycle length average.

A measure of reactor core burnup considers thermal energy output per unit mass of fuel –Megawatt-days per metric-ton of uranium or heavy metals (MWd/MTU). According to convention, fuel is considered to be the heavy metal content (total Th, U, and Pu), exclusive of alloy or compound constituents (Knief 1992). The depletion analyses were computed for a total burnup of 60,000 MWd/MTU, as to evaluate the impact of depletion parameters at higher burnup. A higher burnup is selected since ISG-8 (NRC 2002) already allows burnup credit to 40,000 MWd/MTU and to represent current regulations' limits on burnup. Current nation limits are 55 GWd/MTU for Belgium and Japan, while some nations such as Czech Republic and South Korea follow the U.S. Nuclear Regulatory Commission (NRC) guidance of 62 GWd/MTU peak rod average burnup (OECD 2006). Research continues to investigate higher burnups.

The cycle length and core burnup affects the specific power of the depletion cycles. The specific power represents the thermal power output per fissionable material in Megawatts per metric ton of heavy metal (MW/MTU). The specific power, as used in the depletion analyses is calculated by the total burnup (60,000 MWd/MTU) divided by the total days for three cycles. For example, a single 18 month cycle has 547.875 days

multiplied by 3 cycles equals 1643.625 days, then dividing the total days into the total burnup of 60,000 MWd/MTU is equivalent to the specific power of 36.505 MW/MTU per cycle. Table 8 displays the cycle lengths and corresponding calculated specific power per fuel design evaluation.

Table 8 Burnup Power and Cycle Specifications

Fuel Design	Cycle Length (days)	Specific Power (MW/MTU)	Reference
Westinghouse 17x17	354.5	56.42	(Mays 1998a)
	547.875	36.505	--
Babcock & Wilcox 15x15	321.9	62.13	(Mays 1998b)
	547.875	36.505	--
General Electric 8x8	413.425	48.376	(Henderson 1999)
	547.875	36.505	--

Decay times of zero day (i.e., fuel discharge) and five years cooling in spent fuel pool conditions are evaluated. The five year cooling assumption is based on guidance from ISG-8 and ORNL evaluations of the effect of cooling time from discharge to 100,000 years for a GBC-32 cask. The ORNL study shows the best-estimate results for k_{eff} at a 10 year cooling time; however a lower limit on cooling time, of five years, for transportation and storage analysis will continue to be dictated by thermal and shielding requirements (Parks et al. 2000b).

The depletion parameters are applied independently and collectively to evaluate the impact on the system reactivity. In conjunction with the depletion parameters, several operation and plant parameters are used in the depletion evaluations.

Calculation and Model Specifications

Two computational models are used in defining the impact on system reactivity. First, a 2-D depletion calculation uses a quarter-assembly model with boundary conditions that represent a particular in core location. The fuel is depleted for the specified operation with varying depletion parameters. This results in an isotopic inventory for the fuel material. Then using a simplified storage basket model, a 3-D criticality safety calculation is computed with the particular fuel design and generated isotopic inventory for the fuel material from the depletion calculations. The resultant eigenvalues are then used for comparison to quantify the depletion parameter(s) impact on the criticality safety calculations of SNF. Both calculations, depletion and criticality, use the same assembly model per fuel design.

Some assembly designs are known to be considerably less reactive than others, it is beneficial to evaluate depletion parameters for each unique class of assembly designs. Each assembly type represents a different fuel manufacture design. Differences in assemblies include: array size, water hole placement, dimension variations, etc. Therefore, the PWR depletion parameters are analyzed for both PWR fuel designs represented by the data. The LWR fuel design specifications are defined in Table 9.

BWR fuel assemblies are composed of the fuel bundle and a channel surrounding the fuel to promote internal flow. The channel is accurately represented in the model as thin SS box as defined in the CRC data. For a BWR, a fixed burnable absorber is integrated into the fuel material as a BPR, and a control blade is still used for fine flux changes. A control blade is a cross that is inserted between a four square of adjacent assemblies, and is composed of a stainless steel (SS) shell with 21 boron carbide (B_4C)

absorber rods lined to the tip in each "wing" of the cross blade (Gauld 2000). For PWR designs, a control rod is composed of a burnable absorber and clad for which the cluster of control rods is then inserted into the guide tube holes when present.

Note that for the BWR water rod, the actual rod outer radius and thickness are larger from the model specification. Due to the model technique of a square lattice cell, adjacent cells must have the same outer dimensions as specified by the fuel pin pitch. In the reduction of the water rod size, a ratio of areas between the tube cladding and moderator are maintained.

Table 9 Fuel Design Model Specifications

Fuel Design Parameter (cm)	Westinghouse 17x17	Babcock & Wilcox 15x15	GE 8x8
Fuel pellet OR	0.4096	0.4667	0.5219
Fuel pin clad OR	0.4750	0.5461	0.6134
Fuel pin clad thickness	0.0572	0.673	0.0813
Fuel pin pitch	1.2600	1.4427	1.6300
Active fuel length	365.76	363.728	381.0
Guide tube clad OR	0.6121	0.6731	--
Guide tube clad thickness	0.0406	0.0406	--
Instrument tube clad OR	0.6121	0.6910	--
Instrument tube clad thickness	0.0406	0.1309	--
Water rod OR	--	--	1.6300
Water rod thickness	--	--	0.148
Control blade wing span from center	--	--	12.383 [a]
Control blade width	--	--	0.7925 [a]
Control blade sheath thickness	--	--	0.1422 [a]
Control rod pellet OR	0.4331	0.4978	0.1753 [a]
Control rod clad OR	0.4839	0.5588	0.2388 [a]
Control rod thickness	0.0470	0.0533	0.0635
Channel width	--	--	13.4061
Channel thickness	--	--	0.100
Fuel assembly pitch	21.5036	21.811	15.24
Reference	(Mays 1998a)	(Mays 1998b)	(Henderson 1999)

NOTE: [a] Reference: (Gauld 2000); OR is outer radius; "--" means not applicable due to not a design feature

Material specifications are provided by either the CRC data or standard compositions from the SCALE library are used. Industry standardized material compositions are predefined in the SCALE library and referenced by the material common name (e.g., stainless steel type 304 is SS-304). PWR control rod materials specification and density are provided by the CRC data, while temperature data is specified by a reference value to simulate approximate typical conditions. Similar temperature values are applied to the cladding materials. The component material temperature effect is negligible for criticality safety evaluations, and mainly impacts the material properties of degradation. Hence parameters with low impact on criticality analyses, such as clad and control blade temperature, are modeled with nominal values. The fuel density specification was utilized as 95% theoretical. The BPR is defined by enriched UO_2 fuel plus the burnable poison Gd_2O_3, the density is conservatively combined based 95% theoretical density UO_2 and 5 wt% Gd_2O_3 at theoretical density.

Table 10 Material Model Specifications

Material	**Westinghouse 17x17**	**Babcock & Wilcox 15x15**	**General Electric 8x8**	
Fuel	UO_2	UO_2	UO_2	UO_2-Gd_2O_3
	95% theoretical density	95% theoretical density	95% theoretical density	UO_2 95% theoretical density + 5wt% Gd_2O_3 at theoretical density
Fuel pin clad	Zircalloy	Zircalloy	Zircalloy	
	6.56 g/cc	6.56 g/cc	6.56 g/cc	
	600 K	600 K	600 K	
Control rod	Ag (80%) In (15%) Cd (5%)	Ag (80%) In (15%) Cd (5%)	B_4C [a]	
	10.16 g/cc	10.17 g/cc	1.76 [a]	
	600 K	600 K	600K	

Material	Westinghouse 17x17	Babcock & Wilcox 15x15	General Electric 8x8
Control rod clad	SS-304	SS-304	SS-304
Control blade	--	--	SS-304
Moderator	H_2O + soluble Boron	H_2O + soluble Boron	H_2O

Reference: [a] (Gauld 2000)

Two assembly locations in the reactor core are analyzed, edge and center regions. The edge region is modeled with a 30cm thick moderator reflector region and vacuum boundary conditions on two sides, simulating leakage. The center region is modeled with full reflective boundaries, simulating an infinite array of bundles. To model the fuel assembly, symmetry is used to reduce the number of pins tracked, and hence reduce computing time. Therefore, a quarter-assembly is modeled, as shown in Figure 8, using reflective boundaries on the exposed side to represent an entire assembly. Figure 8, Figure 9, and Figure 10 display the quarter assembly SCALE models for the Westinghouse 17x17, Babcock & Wilcox 15x15, and General Electric 8x8 designs, respectively, while Figure 11 displays the edge location model of the Westinghouse 17x17. The figures show the rod patterns, including the fuel rods, guide tubes, and instrument tube, as referenced from the CRC data. Below is a figure color key.

Figures Color Key:

	Moderator
	Fuel
	Burnable Poison Fuel
	Clad

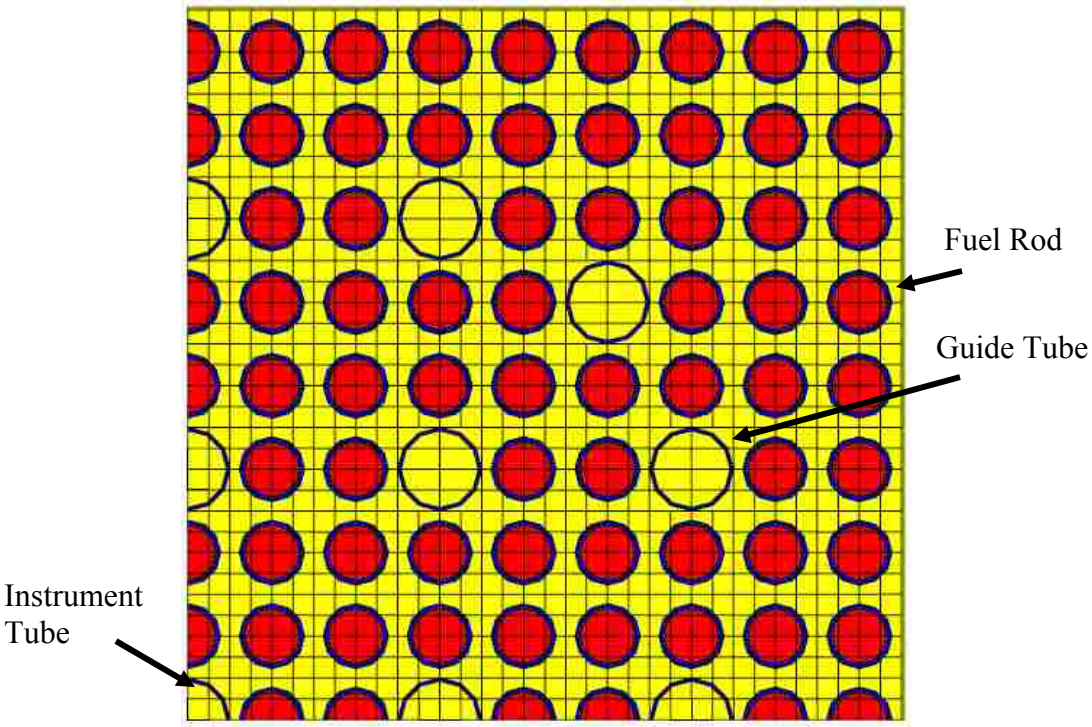

Figure 8 Quarter Assembly SCALE Model of Westinghouse 17x17 Fuel

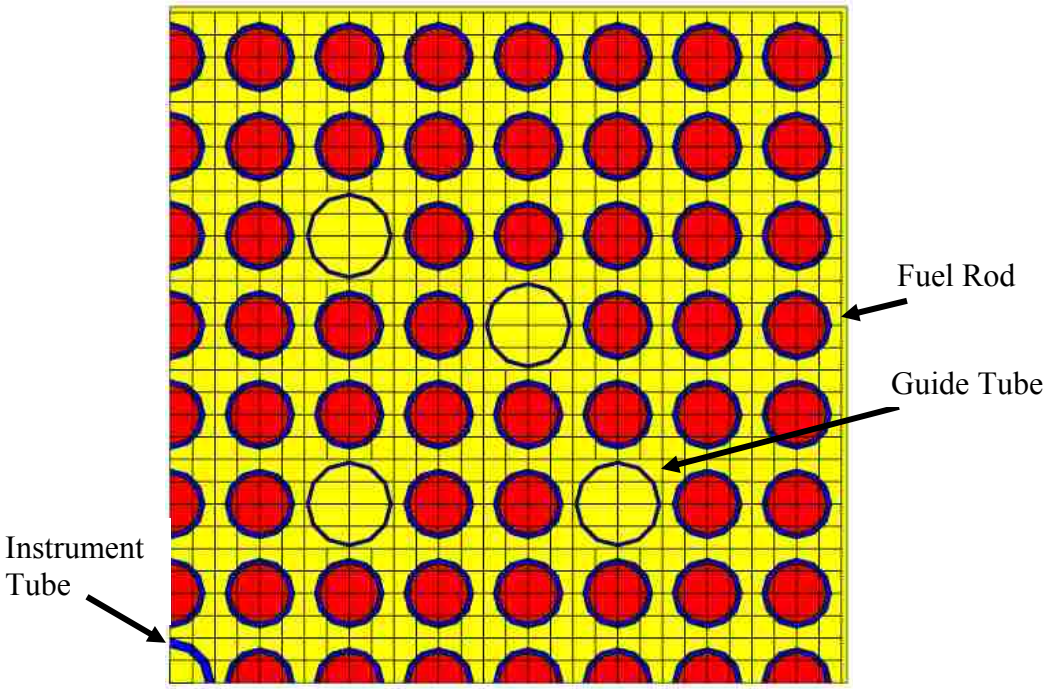

Figure 9 Quarter Assembly SCALE Model of Babcock & Wilcox 15x15 Fuel

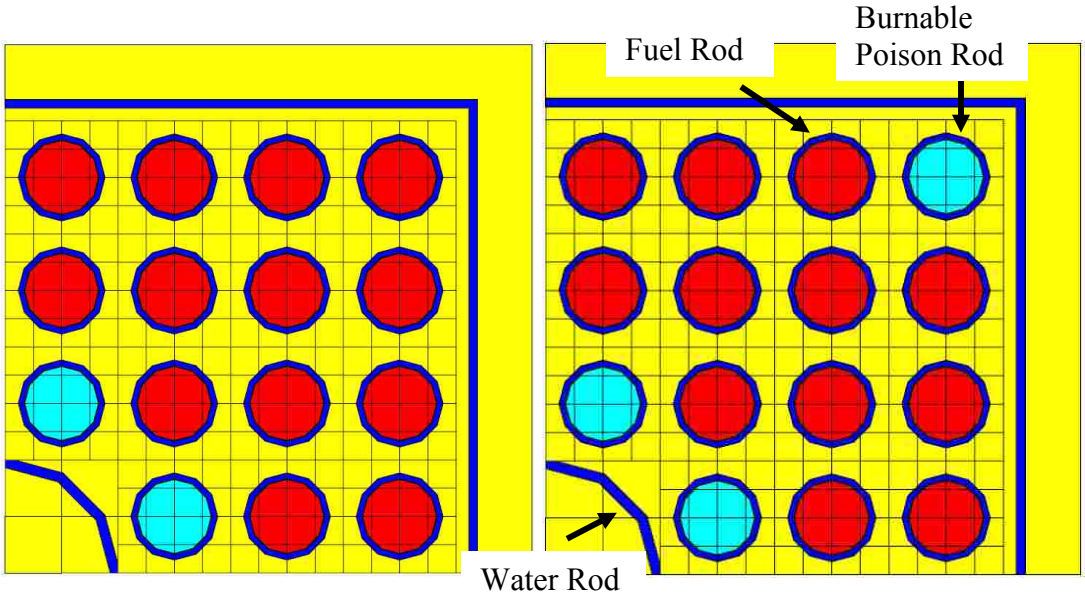

Figure 10 Quarter Assembly SCALE Model of General Electric 8x8 Fuel

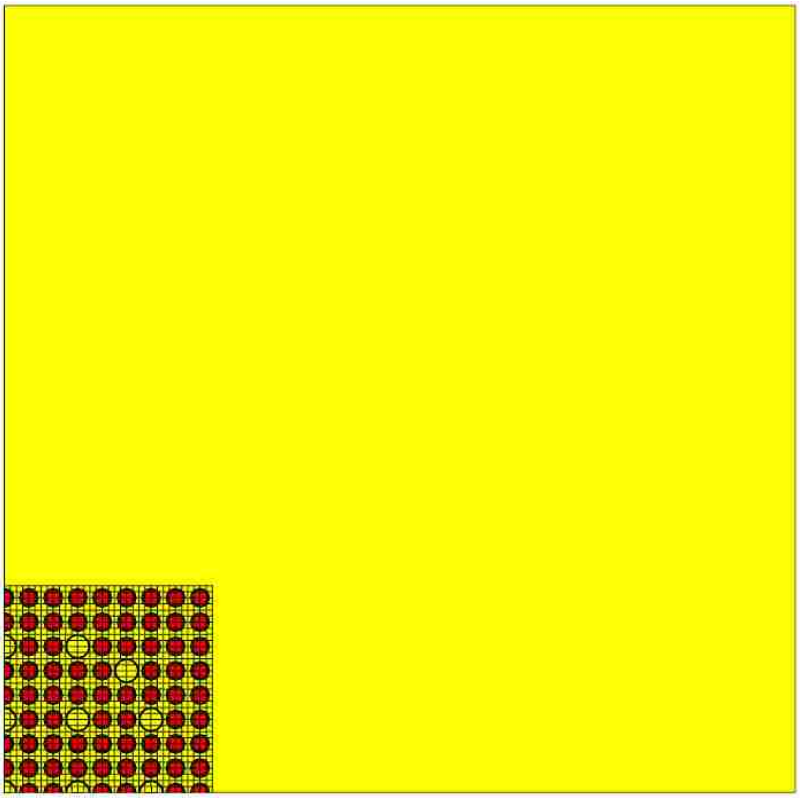

Figure 11 Quarter Assembly Model for Edge Location

For BWRs, the control blade cells are not located near the edge assemblies, as shown in Figure 12 for quarter-core symmetry where control blade cells are highlighted. Hence the edge located assembly depletion calculations are not evaluated with control blade presence.

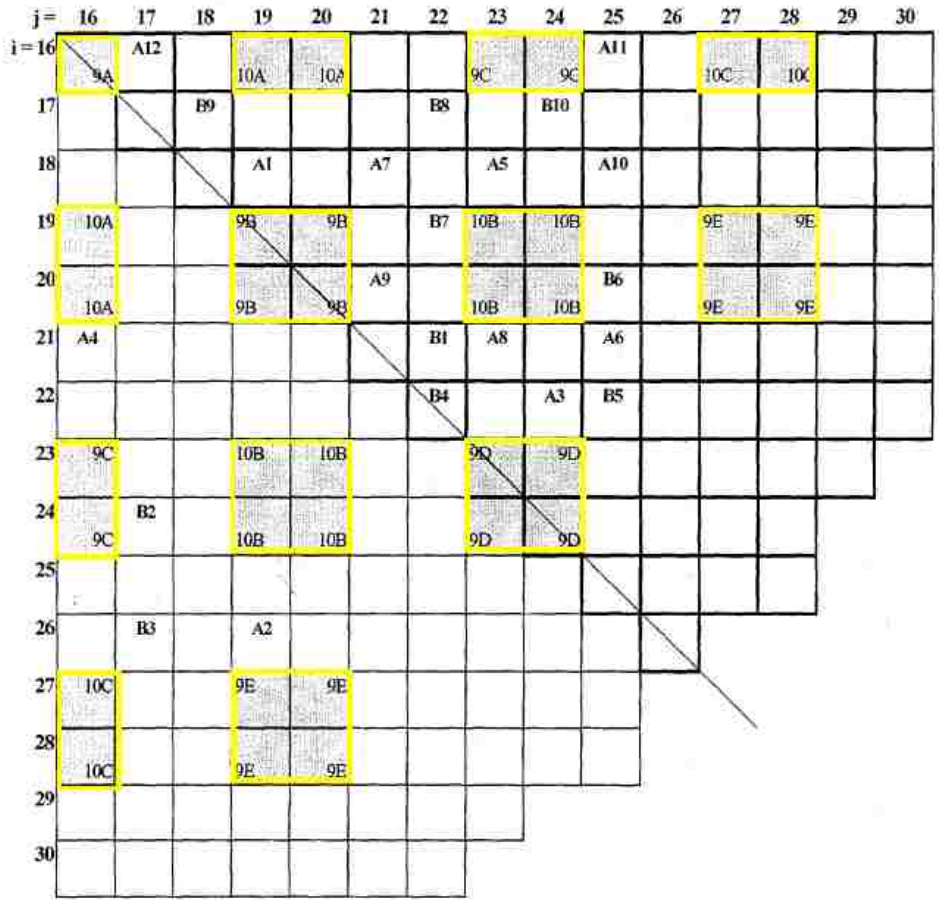

Figure 12 BWR Control Blade Cell Locations in Quarter-Core (Henderson 1999)

Additional parameters defined by assumptions and deviations from reality are modeled conservatively. For instance, the gap between the fuel pellet and the fuel clad is back filled with Helium in manufacturing; however, the gap is conservatively modeled as flooded with water at the density and temperature of the moderator.

The depletion calculations represent 60 GWd/MTU burnup for three equivalent cycles at lengths of approximately 12 months, represented by CRC data, and 18 months. The isotopic inventory is then summarized at two decay heat times, discharge at end of cycle and five year cooling. For the five year cooling time evaluation, the assembly is simulated by SNF pool conditions, where the moderator is water at a nominal density of 1 g/cc and room temperature of 70°F.

Upon completion of the depletion evaluations, the resultant isotopic inventory for the fuel is used to define the fuel material for the criticality calculation in a basic storage configuration. For each of the two calculations, depletion and criticality, the same SCALE model of the assembly specifications is used, except where noted for the BWR water rod. While the depletion calculations are in 2-D, the criticality calculation is in 3-D by merely extruded to the length of the fuel. Using the fuel isotopic concentrations as specified by the depletion calculations, the fuel assembly is modeled in a basic storage configuration as shown in Figure 13.

The storage basket is represented by a SS box surrounded by neutron absorber materials. The most common neutron absorber is a metal matrix composite known as Boral™ composed of 70% aluminum and 30% boron carbide at its manufacturing limits; because of its extensive use and testing it has demonstrated suitability for SNF storage and transportation (EPRI 2005). The typical cross-section of a PWR storage cell,

specified in EPRI 2005, is used for the criticality calculation. The assembly is centered within the storage cell of an 8.95 in. square SS box with an outer neutron absorber plate sandwiched between SS sheathing at a total thickness of 0.14 in. and 7.5 in. width. The storage cells are offset by 11 in. spacing in a flooded environment.

Full representation of the BWR assembly limits the modeling capability of a full water rod, as it occupies four pin cells, hence, the single water rod is modeled as four small rods, maintaining a ratio of areas between the tube cladding and moderator.

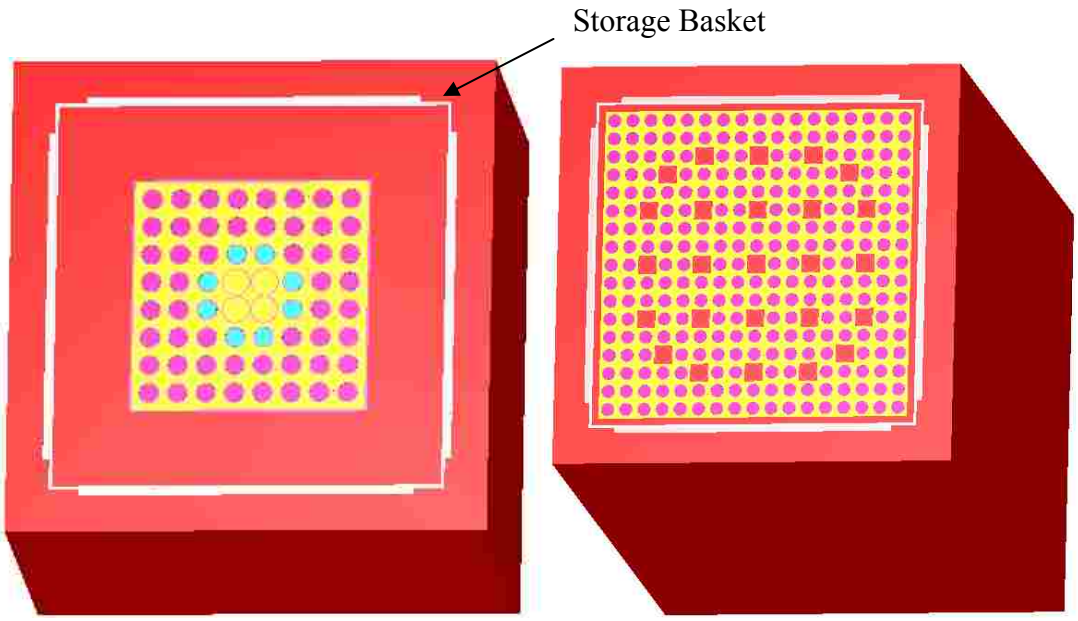

Figure 13 Storage Configuration: BWR (left), PWR (right)

Utilizing a k_{eff} basis, the depletion parameter impact on the system is quantified by a Δk comparison. Hence a base case is defined as the nominal represented by the

average value of each depletion parameter. Table 11 and Table 12 display PWR and BWR, respectively, summary of the values used to evaluate each depletion parameter, as defined in the previous section, per fuel design. In addition, the fuel designs are evaluated with 4% and 5% U-235 enrichment with a burnup of 60 GWd/MTU at discharge and 5 year cooling time. Due to quarter-assembly symmetry modeling, the average number of BPRs in the BWR lattice is increased from seven to eight.

Table 11 PWR Depletion Parameter Value Summary

		T_{fuel} (K)		ρ_{mod} (g/cc)		T_{mod} (K)		S_B (ppm)	
Fuel Design	**Cycle length (days)**	**Ave**	**Max**	**Ave**	**Min**	**Ave**	**Max**	**Ave**	**Max**
Westinghouse 17x17	354.5	880.42	1128.87	0.711	0.6433	559.3	600	545.05	750
	547.875	880.42	1128.87	0.711	0.6433	559.3	600	817.57	1125
Babcock & Wilcox 15x15	321.9	880.42	1128.87	0.711	0.6433	559.1	600	592.1	750
	547.875	880.42	1128.87	0.711	0.6433	551.1	600	888.2	1125

Table 12 BWR Depletion Parameter Value Summary

		T_{fuel} (K)		ρ_{mod} (g/cc)			T_{mod} (K)	BPRs (No. @ wt%)	
Fuel Design	**Cycle length (days)**	**Ave**	**Max**	**Min**	**Ave**	**Max**	**Ave**		
General Electric 8x8	413.425	910.38	1358.4	0.1741	0.399	0.7398	354.3	8 @ 5wt%	12 @ 5wt%
	547.875	910.38	1358.4	0.1741	0.399	0.7398	354.3	8 @ 5wt%	12 @ 5wt%

Selected values are considered reasonable as based on statistics and utilized comparisons. The comparisons should not be sensitive to minor variations in the parameter values. However, the results will be sensitive to major variations in the parameter values (e.g., significant increases or decreases in the difference between bounding and nominal values). The quantifiable effect of individual modeling parameters are evaluated independently and in a cumulative manner, concluding a conservative design that includes primary bounding depletion parameters necessary for burnup-credit criticality safety evaluations. To determine the depletion parameter reactivity impact, the k_{eff} of each system is evaluated and compared. Table 13 and Table 14 display a PWR and BWR, respectively, summary of cases evaluated per fuel design at 4% and 5% U-235 enrichment for 12 month and 18 month cycles with a burnup of 60 GWd/MTU at discharge and 5 year cooling time. For BWRs, BPRs are always present except in naturally enriched U lattices (Henderson 1999). Therefore depletion parameter impacts are quantified against the average, base case for each evaluated number of BPRs, 8 and 12, respectively.

Table 13 Summary of PWR Depletion Parameter Variations by Case

	T_{fuel}		ρ_{mod}		T_{mod}		S_B		CR	
Case	**ave**	**max**	**ave**	**min**	**ave**	**max**	**ave**	**max**	**in**	**out**
base-ave	x		x		x		x			x
1	x		x		x			x		x
2	x			x	x		x			x
3	x			x	x			x		x
4	x			x		x	x			x
5	x			x		x		x		x
6		x	x		x		x			x
7		x	x		x			x		x
8		x		x	x		x			x

	T_{fuel}		ρ_{mod}		T_{mod}		S_B		CR	
Case	**ave**	**max**	**ave**	**min**	**ave**	**max**	**ave**	**max**	**in**	**out**
9		x		x	x			x		x
10		x		x		x	x			x
11		x		x		x		x		x
12	x		x		x		x		x	
13	x		x		x			x	x	
14	x			x	x		x		x	
15	x			x	x			x	x	
16	x			x		x	x		x	
17	x			x		x		x	x	
18		x	x		x		x		x	
19		x	x		x			x	x	
20		x		x	x		x		x	
21		x		x	x			x	x	
22		x		x		x	x		x	
23		x		x		x		x	x	

Note: T_{fuel} is fuel temperature; ρ_{mod} is moderator density; T_{mod} is moderator temperature; S_B is soluble boron concentration; CR is control rod presence; ave is average; max is maximum; min is minimum

Table 14 Summary of BWR Depletion Parameter Variations by Case

	T_{fuel}		ρ_{mod}			BPR		CB	
Case	**ave**	**max**	**min**	**ave**	**max**	**8**	**12**	**in**	**out**
base-ave	x			x		x	x		x
1	x			x		x	x		x
2	x				x	x	x		x
3	x		x			x	x		x
4		x		x		x	x		x
5		x			x	x	x		x
6		x	x			x	x		x
7	x			x		x	x	x	
8	x				x	x	x	x	
9	x		x			x	x	x	
10		x		x		x	x	x	
11		x			x	x	x	x	
12		x	x			x	x	x	

Note: T_{fuel} is fuel temperature; ρ_{mod} is moderator density; BPR is burnable poison rod quantity; CB is control blade presence; ave is average; max is maximum; min is minimum

SCALE 6 Code

The SCALE code system is being developed and supported at ORNL under a co-sponsorship of the NRC and DOE (ORNL 2009). It is a multi-purpose computer code system for the analysis of nuclear facilities and packages including but not limited to reactor physics, fuel cycle, criticality safety, shielding, lattice physics, radiation source terms, SNF and HLW characterization (ORNL 2009).

The combination, execution and communication between various SCALE functional modules are maintained by control modules. Control modules operate as sequence controllers, preparing input for functional modules, transferring data, and executing functional modules in the appropriate sequence for a particular analysis type (ORNL 2009). This research is based on the capabilities of the TRITON and CSAS control modules. TRITON is utilized for 2-D depletion calculations through coupling of NEWT geometry processor and the ORIGEN-S depletion code (ORNL 2009). While the CSAS control module is utilized for 3-D criticality calculations through automated, problem-dependent, cross-section processing followed by calculation of the neutron multiplication factor for the system being modeled by the functional module KENO-Va. (ORNL 2009).

TRITON Module

TRITON has the capability to perform precise burnup-dependent physics calculations with few implicit approximations, limited primarily by the accuracy of nuclide cross-sectional data (DeHart 2006). TRITON uses a predictor-corrector method, iteratively calling NEWT and ORIGEN-S to track changing flux and power distributions

with burnup, matching time-dependent power to user-specified operating histories (ORNL 2009). The depletion approach performs cross-section updates with burnup in order to capture the effect of changing nuclide inventories as the initial fuel is depleted and the buildup of higher actinides and fission products occurs. The T-DEPL sequence allows for refinement of the cycle depletion time-step size by allowing more intermediate steps of cross-section approximation within each cycle resulting in an increased number of transport/depletion steps in each cycle (ORNL 2009). As fuel depletes, the isotopic inventory changes over time, which affects the neutron spectrum observed in the fuel. Frequent updates reflect change in spectrum over time. The T-DEPL sequence of the TRITON module consists of two components during this iterative phase: (1) transport calculations (cross section processing and the neutron transport solution) and (2) depletion calculations (COUPLE and ORIGEN-S). Transport calculations are used to calculate fluxes and prepare weighted cross sections and other lattice physics parameters based on a given set of nuclide concentrations; depletion calculations are used to update nuclide concentrations, which can be used in the following transport calculation (ORNL 2009).

This predictor-corrector process, as explain in the manual (ORNL 2009), is shown in Figure 14, where in the figure, transport and depletion calculations are represented by the labels T and D, respectively. The T0/D0 step is a predictor calculation to estimate cross sections for using in the subsequent step. Hence, the D1 depletion cycle restarts at time zero, but performs a depletion calculation using updated cross sections provided by the T1 transport solution, corrector step. Both D1 and D2 depletion steps are executed in two stages: the first stage performs the current cycle (1 and 2, respectively) depletion and

decay calculations, and provides the isotopic concentrations for the beginning of the next cycle; the second stage is a predictor step to obtain concentrations for the successive transport solution.

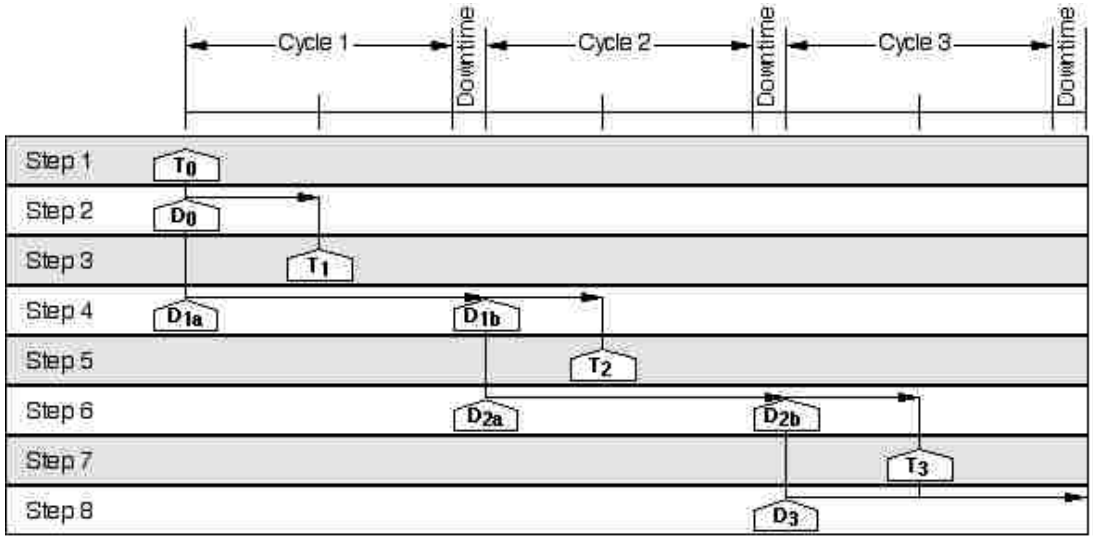

Figure 14 TRITON Predictor-Corrector Depletion Process (ORNL 2009)

This analysis uses the NEWT functional module at the 2-D quarter-assembly modeling level. NEWT, as utilized in TRITON, calculates spatial flux distributions, collapses nuclide cross sections, and generates a library of cross sections as a function of burnup. The functional module, NEWT, utilizes a discrete-ordinates approximation to the transport equation on an arbitrary grid, and provides a deterministic solution for non-orthogonal configurations (DeHart 2006). In depletion mode, NEWT creates a three-group weighted cross-section library based on calculated and volume-averaged fluxes for each mixture.

Cross section processing is carried out by NITAWL based on the ENDF/B-V 44-group broad group library, which was collapsed from the fine group library using a spectrum representative of a UO_2 LWR spectrum (ORNL 2009). NITAWL applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. This is important because as the self-shielding increases, the average resonance cross section decreases, and if the absorption cross section decreases, k_{eff} will increase. After performing the resonance analysis, NITAWL combines the shielded cross sections with the fast and thermal data to produce a working library organized by reaction type and scattering expansion order (ORNL 2009). Doppler broadening treatment in NITAWL has been shown to demonstrate maximum deviation of less than 1% from integrations over numerical quadrature evaluations (ORNL 2009).

As explained in the manual (ORNL 2009), NITAWL is basically a two-region integral transport theory method for a fuel lump surrounded by a moderator region. The Material Information Processor utilizes a unit cell description to provide information for the resonance self-shielding corrections and the Dancoff corrections that are applied to the cross sections to create a problem-dependent cross-section library. A unit cell description is specified defining the materials, dimensions, and boundary conditions of the geometry that will be used in the Dancoff factor calculations for NITAWL, the resonance self-shielding calculations, and the flux-weighting cell calculations used in cross-section processing.

The NITAWL method was designed to treat a single fuel lump in an infinite moderator. To account for the heterogeneous effects of a lattice of fuel lumps, a correction known as the Dancoff factor is applied to the leakage probability from the

lump. The interlump Dancoff factor is the probability that a neutron emitted isotropically from the surface of one absorber lump will pass through the external media and enter a nearby absorber lump. The overshadowing of one neighboring lump by another is accounted for analytically and includes all nearest and second-nearest neighbors, as shown in Figure 15 for a square lattice of cylinders. For a particular fuel lump, the total Dancoff factor is calculated for the appropriate lattice based on the summation of all fuel regions visible to the lump, including an added a correction factor to treat the interaction to the third and subsequent nearest neighbors.

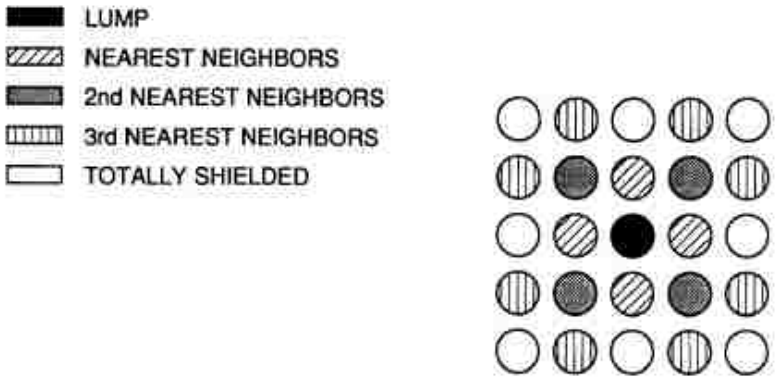

Figure 15 Dancoff Factor Neighbor Specification (ORNL 2009)

CSAS Module

CSAS5 provides automated, problem-dependent, cross-section processing followed by calculation of the neutron multiplication factor, k_{eff}, for the system being modeled using KENO-Va (ORNL 2009). The modules utilized in CSAS start with an

AMPX master format cross-section library and generate a self shielded, group-averaged library applicable to the specific problem configuration. These cross sections are then used in the KENO-Va Monte Carlo code to determine the effective neutron multiplication factor. LATTICECELL is used to describe the fuel assembly as a lattice, and is used for cross section correction for resonance self-shielding, including geometry effects. For consistency, CSAS criticality calculations have also been specified with NITAWL cross-section processing based on the ENDF/B-V 44-group library. With NITAWL, the same resonance self-shielding corrections and Dancoff corrections are made as described in the previous section.

KENO-Va is a multigroup Monte Carlo criticality code that uses shapes to create geometry units and arrays to create repeated structures, such as pins in an assembly. In criticality safety, the Monte Carlo procedure is applied by developing a model that is capable of "tracking" individual neutrons through a material medium containing fissile and other materials. The Monte Carlo method relates physical events (scattering, absorption, fission, etc.) to random numbers by using probability density functions. The probability of occurrence of any given event is identical to the probability that the corresponding random number will be selected.

Code Validations

Validation methods include isotopic bias and uncertainty via radiochemistry assay experiments or sensitivity studies with confirmatory data and cross section bias via critical experiments or sensitivity studies with confirmatory data. Often simpler to verify actinide burnup as these isotopes are well represented by experiments and measured data,

fission products and minor actinides have less experimental data for validation of the isotopic and cross section data. An EPRI 2010 report seeking to quantify validation uncertainties, particularly for fission products, shows uncertainties associated with fission products and minor actinides are significantly larger than that of major actinides. However, the Δk values are on the same order of magnitude for actinide-only burnup, for PWR 0.014, and total burnup, for PWR 0.037; this is due to the reactivity worth of fission product being lower than major actinides (Wells 2010).

Using a common methodology of the propagation of errors, uncertainty and bias sensitivity studies of isotopic and cross-section data is 2.5% Δk more conservative than utilizing CRC data, which has been shown to have a bias and uncertainty of -0.0143 Δk (Wells 2010).

TRITON tends to over-predict reactivity, by less than 5% Δk, at high burnup over other depletion codes for k-infinity analysis of BWR fuel assemblies; this may be due to the inclusion of insufficient numbers of fission products in the transport model (DeHart 2006). Utilizing radiochemical assay data to benchmark the TRITON depletion sequences, ORNL showed agreement within approximately 10% of the 2-D deterministic transport method T-DEPL and the measured spent fuel data for majority of nuclides (DeHart et al. 2005). Nuclides of higher error have relatively low concentrations and importance (DeHart et al. 2005). The benchmarks cover a wide variety of fuel designs and modeling techniques, defining an adequate use of the 2-D TRITON depletion module for LWR system analyses.

For CSAS criticality calculations the effective neutron multiplication factor is determined by the Monte Carlo code. This implicates a statistical uncertainty associated

with each calculation. All criticality calculation performed for this research use the CSAS-KENO-Va Monte Carlo method and have an associated uncertainty within $3.0 \cdot 10^{-3} \pm 9.0 \cdot 10^{-4}$.

Low uncertainties and bias in isotopic and cross section data are important to validating use of computational methods. Utilizing the same cross section data for depletion calculation with TRITON and criticality calculations with CSAS maintain consistency in calculation data. Since depletion parameter reactivity impacts are defined by a Δk, no bias or uncertainty is added to the quantified values. However, further research is needed in the validation of SNF isotopic and cross section uncertainty and bias per specific cases for licensing purposes as required by ISG-8 (NRC 2002).

CHAPTER 4

DATA ANALYSIS

Results

Calculations performed were not an attempt to define a limiting profile or an appropriate safety margin; rather, they were performed to demonstrate relative effects in depletion modeling and evaluate effects of depletion parameters on spent fuel reactivity. Calculations evaluated the effects of varying ranges of depletion parameters on the calculated neutron multiplication factor of the system. Each of the depletion parameters was evaluated independently and then collectively to determine the reactivity impact of the system.

Depletion parameters examined in this research include: fuel temperature, moderator density (converted from moderator specific volume), moderator temperature, burnable absorber presence defined by soluble boron concentration for PWRs and BPRs for BWRs, and control rod/blade presence. The reactivity impacts of each depletion parameter individually and collectively are quantified in the following tables. The reactivity impact comparison ($\Delta k = k_{eff\text{-}case\ X} - k_{eff\text{-}ave}$) is evaluated as the difference between the parameter variation case *X* ($k_{eff\text{-}case\ X}$) and the established base case ($k_{eff\text{-}ave}$), which represents the average of each depletion parameter. Table 15 and Table 16 represent the Westinghouse 17x17 PWR depletion parameter impact results for 12 month and 18 month cycles, respectively. Tables 17 and 18 represent the Babcock & Wilcox 15x15 PWR depletion parameter impact results for 12 month and 18 month cycles, respectively. Tables 19 and 20 represent the General Electric 8x8 BWR depletion parameter impact results for 12 month cycles at 4% and 5% U-235 enrichments,

respectively. Eighteen month cycle depletion parameter impact results are represented in Tables 21 and 22 for 4% and 5% U-235 enrichment, respectively. Individual depletion parameter cases are highlighted in yellow in each table.

Trends observed are provided for PWR and BWR separately. Note that only positive reactivity impacts are discussed, as this represents an increase in the reactivity of the SNF due to that depletion parameter variation. For PWRs, decay time has minimal impact on reactivity of SNF, commonly ±1%, and often resulting in no change at all from discharge values. Trends at 5% U-235 enrichment are less than 4% U-235 enrichment. Additionally, for all cases, Babcock & Wilcox 15x15 fuel has lower impacts on reactivity than Westinghouse 17x17 fuel.

Individual depletion parameter reactivity impacts are varying degrees of increased, positive reactivity. For both evaluated PWR designs, similar depletion parameter reactivity impact trends are observed. PWR depletion parameters are listed in decreasing worth of impact on reactivity: (1) highest impact is the presence of fixed burnable absorbers (i.e., control rods), (2) is minimum moderator density at average moderator temperature, (3) maximum fuel temperature, and (4) maximum burnable poison concentration as soluble boron. The realistic depletion parameter combination of minimum moderator density and maximum moderator temperature result in a lower impact on reactivity of 0.2% - 0.5%, than simulating a minimum density and average temperature of the moderator for depletion calculations. While increasing any of these parameters in combination results in a larger impact on reactivity of SNF, the presence of control rods is largest contributor.

For Westinghouse 17x17 design, the individual parameters have the following reactivity impacts, in order of decreasing impact:

1. Control rod presence: 6.0% - 2.7%
    - Upper range is represented by 4% U-235 enrichement and center core locations
    - Lower range is represented by 5% U-235 enrichment and edge core locations
    - 18 month cycle lengths have a 0 - 0.5% less reactivity impact than 12 month cycle lengths
2. Minimum moderator density at average moderator temperature: 3.1% - 1.2%
    - Upper range is represented by 4% U-235 enrichement and center core locations
    - Lower range is represented by 5% U-235 enrichment and edge core locations
    - 18 month cycle lengths have a 0 - 0.5% less reactivity impact than 12 month cycle lengths
3. Maximum fuel temeprature: 1.7% - 0.4%
    - Upper range is represented by 4% U-235 enrichement and center core locations
    - Lower range is represented by 5% U-235 enrichment and edge core locations
4. Maximum soluble boron concentration: 1.7% - 0.3%
    - Upper range is represented by 4% U-235 enrichement and edge core locations
    - Lower range is represented by 5% U-235 enrichment and center core locations

Combination of the individual highest worth parameters, control rod presence, minimum moderator density with average temperature, maximum fuel temperature, and maximum soluble boron concentration, represent the largest Westinghouse 17x17 PWR

positive reactivity impact of 8.7 - 6.1% for all center location cases and 6.5 - 4.4% for edge locations. While the difference between 5% and 4% U-235 enrichment is reacitivity impact increase of 1% - 2% for a decrease in enrichment.

For Babcock & Wilcox 15x15 design, the individual parameters have the following reactivity impacts, in order of decreasing impact:

1. Control rod presence: 4.9% - 2.9%
    - Upper range is represented by 4% U-235 enrichement and center core locations
    - Lower range is represented by 5% U-235 enrichment and edge core locations
2. Minimum moderator density at average moderator temperature: 2.3% - 1.3%
    - Upper range is represented by 4% U-235 enrichement and center core locations
    - Lower range is represented by 5% U-235 enrichment and edge core locations
3. Maximum fuel temeprature: 0.8% - 0.4%
    - Upper range is represented by 4% U-235 enrichement and center core locations
    - Lower range is represented by 5% U-235 enrichment and edge core locations
4. Maximum soluble boron concentration: 0.5% - 0.2%
    - Upper range is represented by 4% U-235 enrichement and edge core locations
    - Lower range is represented by 5% U-235 enrichment and center core locations

Combination of the individual highest worth parameters, control rod presence, minimum moderator density with average temperature, maximum fuel temperature, and maximum soluble boron concentration, represent the largest Babcock & Wilcox 15x15

PWR positive reactivity impact of 7.6 - 6.0% for all center location cases and 6.1 - 4.7% for edge locations. While 4% U-235 enrichment represents the upper range and 5% U-235 enrichment represents the lower range.

For BWR the dominating feature is integral BPRs in the assemblies. Inclusion of BPRs reduces reactivity of SNF, however they are always present except in naturally enriched U lattices (Henderson 1999). Therefore the following depletion parameter impacts are quantified against the average, base case for the evaluated number of BPRs. It is important to note that trends show an increased number of BPRs decreases the depletion parameter reactivity impact.

General trends shows for BWRs nearly similar reactivity impacts for each case at 12 month and 18 month cycle lengths for 5% and 4% U-235 enrichments, respectively. Edge locations produce a lower reactivity impact than center locations; however, the SNF reactivity is higher for edge locations. SNF at discharge tends to show a lower impact trend for depletion parameters than SNF after a 5 year cooling period, except impacts at less than 1% show little difference (less than a few tenths of a percent). This is due in part to the buildup of longer lived fission products.

Individual depletion parameter reactivity impacts vary from positive and negative. The largest positive parameter impact on BWR SNF reactivity is the minimum moderator density at approximately 10.2-6.2% for center and 1.1-0.5% for edge locations. The upper end of the range represents 8 BPRs and 4% U-235 enrichment, and the lower range is more commonly held by 12 BPRs and 5% U-235 enrichment. The second positive impacts are control blade insertion, at 1.7% - 1.5% for 8 BPRs, while with 12 BPRs the reactivity impact is negative. The third positive reactivity impact is maximum fuel

temperature at 0.8% - 0.2% for all cases. Other evaluated depletion parameters, maximum moderator density and increased number of BPRs, cause a negative impact on reactivity.

Combination of the individual highest worth parameters, minimum moderator temperature, control blade presence, and fuel temperature, represent the second largest positive reactivity impact, 10.2% - 8.4% for 8 BPRs and 7.4% - 6.0% for 12 BPRs, often equivalent to the individual impact of minimum moderator density. However, the largest impact, 10.6% - 6.5% for center locations, is the combination of minimum moderator density and maximum fuel temperature, which is representative of realistic operation conditions. The lower end of the range is represented by 12 BPRs, while the upper end represents 8 BPRs. This combination of parameters, minimum moderator density and maximum fuel temperature, also represents the largest reactivity impact for edge location at a range of 1.3% - 0.7%.

Table 15 PWR Westinghouse 17x17, 12 month cycle Parameter Reactivity Impact Results

PWR 12 month cycle Westinghouse 17x17											4% enrichment, Δk reactivity impacts				5% enrichment, Δk reactivity impacts			
											center		edge		center		edge	
	T_{fuel}		ρ_{mod}		T_{mod}		S_B		CR		BU 60		BU 60		BU 60		BU 60	
Case	ave	max	ave	min	ave	max	ave	max	in	out	discharge	5 yr cool	discharge	5 yr cool	discharge	5 yr cool	discharge	5 yr cool
base-ave	x		x		x		x			x	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	x		x		x			x		x	0.005	0.005	0.007	0.006	0.003	0.004	0.006	0.005
2	x			x	x		x			x	0.022	0.022	0.016	0.017	0.019	0.019	0.014	0.013
3	x			x	x			x		x	0.026	0.026	0.022	0.022	0.021	0.021	0.019	0.018
4	x			x		x	x			x	0.018	0.019	0.013	0.013	0.016	0.016	0.011	0.011
5	x			x		x		x		x	0.023	0.023	0.019	0.020	0.019	0.019	0.017	0.016
6		x	x		x		x			x	0.007	0.008	0.006	0.006	0.006	0.007	0.005	0.004
7		x	x		x			x		x	0.012	0.012	0.012	0.012	0.010	0.011	0.011	0.010
8		x		x	x		x			x	0.030	0.030	0.022	0.022	0.024	0.025	0.019	0.018
9		x		x	x			x		x	0.033	0.033	0.028	0.028	0.028	0.028	0.023	0.023
10		x		x		x	x			x	0.028	0.026	0.019	0.019	0.022	0.023	0.016	0.015
11		x		x		x		x		x	0.031	0.031	0.025	0.025	0.025	0.026	0.021	0.021
12	x		x		x		x		x		0.057	0.057	0.043	0.044	0.043	0.045	0.034	0.035
13	x		x		x			x	x		0.059	0.060	0.048	0.050	0.045	0.047	0.039	0.039
14	x			x	x		x		x		0.074	0.076	0.054	0.056	0.058	0.059	0.042	0.045
15	x			x	x			x	x		0.076	0.078	0.059	0.061	0.060	0.061	0.047	0.049
16	x			x		x	x		x		0.073	0.074	0.051	0.053	0.056	0.058	0.041	0.042
17	x			x		x		x	x		0.075	0.076	0.056	0.059	0.058	0.060	0.045	0.047
18		x	x		x		x		x		0.063	0.064	0.047	0.049	0.048	0.050	0.039	0.039
19		x	x		x			x	x		0.066	0.067	0.053	0.054	0.051	0.052	0.043	0.043
20		x		x	x		x		x		0.080	0.081	0.059	0.061	0.063	0.065	0.046	0.049
21		x		x	x			x	x		0.082	0.084	0.063	0.065	0.065	0.066	0.051	0.053
22		x		x		x	x		x		0.079	0.079	0.055	0.058	0.061	0.064	0.044	0.046
23		x		x		x		x	x		0.081	0.082	0.061	0.063	0.063	0.066	0.049	0.050

Table 16 PWR Westinghouse 17x17, 18 month cycle Parameter Reactivity Impact Results

PWR 18 month cycle Westinghouse 17x17											4% enrichment, Δk reactivity impacts				5% enrichment, Δk reactivity impacts			
											center		edge		center		edge	
	T_{fuel}		ρ_{mod}		T_{mod}		S_B		CR		BU 60		BU 60		BU 60		BU 60	
Case	ave	max	ave	min	ave	max	ave	max	in	out	discharge	5 yr cool	discharge	5 yr cool	discharge	5 yr cool	discharge	5 yr cool
base-ave	x		x		x		x			x	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	x		x		x			x		x	0.017	0.006	0.008	0.008	0.005	0.004	0.006	0.006
2	x			x	x		x			x	0.031	0.020	0.016	0.015	0.019	0.017	0.012	0.013
3	x			x	x			x		x	0.037	0.026	0.023	0.024	0.023	0.021	0.019	0.019
4	x			x		x	x			x	0.028	0.018	0.013	0.013	0.016	0.015	0.011	0.011
5	x			x		x		x		x	0.033	0.023	0.020	0.021	0.020	0.020	0.016	0.017
6		x	x		x		x			x	0.017	0.007	0.005	0.006	0.007	0.006	0.004	0.006
7		x	x		x			x		x	0.024	0.014	0.014	0.014	0.012	0.010	0.011	0.011
8		x		x	x		x			x	0.038	0.028	0.021	0.021	0.025	0.023	0.017	0.018
9		x		x	x			x		x	0.044	0.034	0.028	0.029	0.029	0.027	0.024	0.024
10		x		x		x	x			x	0.036	0.025	0.018	0.019	0.023	0.021	0.015	0.016
11		x		x		x		x		x	0.041	0.031	0.026	0.027	0.026	0.025	0.021	0.022
12	x		x		x		x		x		0.060	0.051	0.035	0.037	0.040	0.040	0.027	0.029
13	x		x		x			x	x		0.064	0.054	0.040	0.041	0.042	0.041	0.031	0.033
14	x			x	x		x		x		0.078	0.070	0.046	0.048	0.055	0.054	0.036	0.038
15	x			x	x			x	x		0.081	0.071	0.050	0.053	0.056	0.056	0.040	0.042
16	x			x		x	x		x		0.076	0.068	0.044	0.046	0.054	0.053	0.034	0.036
17	x			x		x		x	x		0.079	0.071	0.048	0.051	0.055	0.054	0.038	0.040
18		x	x		x		x		x		0.067	0.058	0.041	0.041	0.046	0.045	0.031	0.033
19		x	x		x			x	x		0.070	0.061	0.045	0.047	0.048	0.047	0.036	0.037
20		x		x	x		x		x		0.084	0.076	0.051	0.053	0.059	0.059	0.040	0.043
21		x		x	x			x	x		0.087	0.077	0.055	0.058	0.061	0.061	0.044	0.046
22		x		x		x	x		x		0.083	0.074	0.048	0.050	0.059	0.058	0.038	0.040
23		x		x		x		x	x		0.086	0.077	0.053	0.055	0.060	0.060	0.041	0.045

Table 17 PWR Babcock & Wilcox 15x15, 12 month cycle Parameter Reactivity Impact Results

PWR 12 month cycle Babcock & Wilcox 15x15											4% enrichment, Δk reactivity impacts				5% enrichment, Δk reactivity impacts			
											center		edge		center		edge	
	T_{fuel}		ρ_{mod}		T_{mod}		S_B		CR		BU 60		BU 60		BU 60		BU 60	
Case	ave	max	ave	min	ave	max	ave	max	in	out	discharge	5 yr cool	discharge	5 yr cool	discharge	5 yr cool	discharge	5 yr cool
base-ave	x		x		x		x			x	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	x		x		x			x		x	0.004	0.004	0.005	0.005	0.002	0.003	0.004	0.004
2	x			x	x		x			x	0.022	0.022	0.017	0.017	0.018	0.019	0.013	0.014
3	x			x	x			x		x	0.026	0.025	0.021	0.022	0.020	0.021	0.018	0.018
4	x			x		x	x			x	0.019	0.019	0.013	0.013	0.016	0.017	0.010	0.011
5	x			x		x		x		x	0.023	0.022	0.018	0.018	0.018	0.019	0.014	0.016
6		x	x		x		x			x	0.007	0.008	0.006	0.006	0.006	0.007	0.004	0.004
7		x	x		x			x		x	0.012	0.012	0.012	0.010	0.009	0.009	0.009	0.009
8		x		x	x		x			x	0.030	0.029	0.022	0.022	0.025	0.025	0.018	0.019
9		x		x	x			x		x	0.033	0.033	0.027	0.027	0.027	0.028	0.022	0.022
10		x		x		x	x			x	0.026	0.026	0.019	0.019	0.022	0.023	0.016	0.015
11		x		x		x		x		x	0.030	0.030	0.024	0.023	0.024	0.025	0.019	0.020
12	x		x		x		x		x		0.047	0.047	0.038	0.039	0.036	0.038	0.029	0.031
13	x		x		x			x	x		0.050	0.048	0.042	0.041	0.038	0.037	0.032	0.033
14	x			x	x		x		x		0.066	0.067	0.050	0.052	0.051	0.054	0.039	0.042
15	x			x	x			x	x		0.069	0.069	0.054	0.055	0.054	0.055	0.042	0.045
16	x			x		x	x		x		0.064	0.065	0.048	0.049	0.050	0.052	0.037	0.040
17	x			x		x		x	x		0.066	0.067	0.051	0.053	0.052	0.054	0.041	0.043
18		x	x		x		x		x		0.054	0.054	0.043	0.044	0.042	0.043	0.033	0.036
19		x	x		x			x	x		0.056	0.057	0.047	0.047	0.043	0.046	0.036	0.039
20		x		x	x		x		x		0.073	0.073	0.054	0.056	0.058	0.059	0.044	0.046
21		x		x	x			x	x		0.075	0.075	0.058	0.060	0.060	0.061	0.047	0.049
22		x		x		x	x		x		0.070	0.071	0.052	0.054	0.056	0.057	0.041	0.043
23		x		x		x		x	x		0.073	0.074	0.055	0.058	0.058	0.059	0.044	0.046

Table 18 PWR Babcock & Wilcox 15x15, 18 month cycle Parameter Reactivity Impact Results

PWR 18 month cycle Babcock & Wilcox 15x15											4% enrichment, Δk reactivity impacts				5% enrichment, Δk reactivity impacts			
											center		edge		center		edge	
	T_{fuel}		ρ_{mod}		T_{mod}		S_B		CR		BU 60		BU 60		BU 60		BU 60	
Case	ave	max	ave	min	ave	max	ave	max	in	out	discharge	5 yr cool	discharge	5 yr cool	discharge	5 yr cool	discharge	5 yr cool
base-ave	x		x		x		x			x	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	x		x		x			x		x	0.005	0.004	0.005	0.005	0.003	0.003	0.004	0.005
2	x			x	x		x			x	0.023	0.022	0.016	0.017	0.019	0.019	0.013	0.015
3	x			x	x			x		x	0.025	0.025	0.021	0.022	0.021	0.021	0.017	0.019
4	x			x		x	x			x	0.019	0.018	0.013	0.014	0.016	0.016	0.011	0.011
5	x			x		x		x		x	0.022	0.022	0.018	0.019	0.018	0.019	0.014	0.016
6		x	x		x		x			x	0.007	0.008	0.006	0.006	0.007	0.006	0.005	0.005
7		x	x		x			x		x	0.012	0.012	0.010	0.011	0.009	0.010	0.009	0.009
8		x		x	x		x			x	0.030	0.029	0.022	0.022	0.025	0.025	0.018	0.019
9		x		x	x			x		x	0.033	0.033	0.027	0.028	0.027	0.028	0.022	0.024
10		x		x		x	x			x	0.027	0.026	0.019	0.019	0.023	0.022	0.016	0.017
11		x		x		x		x		x	0.030	0.030	0.024	0.024	0.025	0.026	0.019	0.020
12	x		x		x		x		x		0.047	0.049	0.038	0.041	0.037	0.038	0.029	0.032
13	x		x		x			x	x		0.051	0.049	0.042	0.041	0.039	0.038	0.033	0.033
14	x			x	x		x		x		0.067	0.068	0.050	0.053	0.054	0.054	0.039	0.042
15	x			x	x			x	x		0.070	0.070	0.054	0.056	0.055	0.056	0.043	0.046
16	x			x		x	x		x		0.065	0.065	0.047	0.049	0.052	0.053	0.037	0.039
17	x			x		x		x	x		0.067	0.068	0.051	0.053	0.053	0.054	0.040	0.043
18		x	x		x		x		x		0.054	0.055	0.043	0.045	0.043	0.044	0.035	0.035
19		x	x		x			x	x		0.057	0.057	0.047	0.049	0.045	0.046	0.038	0.039
20		x		x	x		x		x		0.074	0.074	0.055	0.058	0.059	0.060	0.045	0.046
21		x		x	x			x	x		0.076	0.076	0.059	0.061	0.060	0.062	0.047	0.049
22		x		x		x	x		x		0.071	0.072	0.052	0.054	0.057	0.058	0.042	0.043
23		x		x		x		x	x		0.074	0.074	0.056	0.058	0.059	0.059	0.045	0.048

Table 19 BWR General Electric 8x8, 12 month cycle and 4% U-235 Enrichment Parameter Reactivity Impact Results

BWR 12 month cycle General Electric 8x8								4% enrichment, Δk reactivity impacts							
								center				edge			
	T_{fuel}		ρ_{mod}			CB		discharge		5 yr cool		discharge		5 yr cool	
Case	ave	max	min	ave	max	in	out	8 BPRs	12 BPRs	8 BPRs	12 BPRs	8 BPRs	12 BPRs	8 BPRs	12 BPRs
base-ave	x			x			x	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	x				x		x	-0.094	-0.087	-0.091	-0.085	-0.005	-0.009	-0.006	-0.011
2	x		x				x	0.098	0.071	0.102	0.075	0.008	0.005	0.011	0.007
3		x		x			x	0.008	0.006	0.008	0.007	0.003	0.003	0.004	0.003
4		x			x		x	-0.088	-0.081	-0.085	-0.080	0.001	-0.004	-0.002	-0.006
5		x	x				x	0.102	0.074	0.105	0.079	0.010	0.006	0.013	0.008
6	x			x		x		0.014	-0.004	0.017	-0.002	--		--	
7	x				x	x		-0.056	-0.074	-0.053	-0.071	--		--	
8	x		x			x		0.092	0.064	0.099	0.071	--		--	
9		x		x		x		0.020	0.000	0.023	0.003	--		--	
10		x			x	x		-0.050	-0.069	-0.048	-0.066	--		--	
11		x	x			x		0.095	0.067	0.102	0.074	--		--	

Table 20 BWR General Electric 8x8, 12 month cycle and 5% U-235 Enrichment Parameter Reactivity Impact Results

BWR 12 month cycle General Electric 8x8								5% enrichment, Δk reactivity impacts							
								center				edge			
	T_{fuel}		ρ_{mod}			CB		discharge		5 yr cool		discharge		5 yr cool	
Case	ave	max	min	ave	max	in	out	8 BPRs	12 BPRs	8 BPRs	12 BPRs	8 BPRs	12 BPRs	8 BPRs	12 BPRs
base-ave	x			x			x	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	x				x		x	-0.093	-0.084	-0.093	-0.082	-0.004	-0.008	-0.007	-0.011
2	x		x				x	0.084	0.062	0.087	0.067	0.009	0.005	0.011	0.007
3		x		x			x	0.007	0.005	0.006	0.006	0.003	0.002	0.003	0.003
4		x			x		x	-0.087	-0.078	-0.086	-0.077	0.000	-0.005	-0.003	-0.007
5		x	x				x	0.088	0.065	0.090	0.069	0.011	0.006	0.012	0.009
6	x			x		x		0.014	-0.002	0.016	0.000	--		--	
7	x				x	x		-0.054	-0.069	-0.052	-0.066	--		--	
8	x		x			x		0.080	0.058	0.087	0.064	--		--	
9		x		x		x		0.019	0.002	0.021	0.005	--		--	
10		x			x	x		-0.049	-0.064	-0.047	-0.061	--		--	
11		x	x			x		0.084	0.060	0.089	0.067	--		--	

Table 21 BWR General Electric 8x8, 18 month cycle and 4% U-235 Enrichment Parameter Reactivity Impact Results

BWR 18 month cycle General Electric 8x8								4% enrichment, Δk reactivity impacts							
								center				edge			
	T_{fuel}		ρ_{mod}			CB		discharge		5 yr cool		discharge		5 yr cool	
Case	ave	max	min	ave	max	in	out	8 BPRs	12 BPRs	8 BPRs	12 BPRs	8 BPRs	12 BPRs	8 BPRs	12 BPRs
base-ave	x			x			x	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	x				x		x	-0.094	-0.088	-0.092	-0.087	-0.004	-0.008	-0.007	-0.011
2	x		x				x	0.098	0.071	0.102	0.076	0.009	0.005	0.011	0.007
3		x		x			x	0.007	0.006	0.008	0.006	0.003	0.003	0.003	0.002
4		x			x		x	-0.088	-0.082	-0.086	-0.081	0.001	-0.004	-0.002	-0.008
5		x	x				x	0.102	0.075	0.106	0.079	0.011	0.007	0.013	0.008
6	x			x		x		0.013	-0.006	0.017	-0.003	--		--	
7	x				x	x		-0.056	-0.075	-0.053	-0.072	--		--	
8	x		x			x		0.092	0.063	0.099	0.070	--		--	
9		x		x		x		0.020	0.000	0.022	0.002	--		--	
10		x			x	x		-0.050	-0.069	-0.048	-0.067	--		--	
11		x	x			x		0.095	0.067	0.102	0.061	--		--	

Table 22 BWR General Electric 8x8, 18 month cycle and 5% U-235 Enrichment Parameter Reactivity Impact Results

BWR 18 month cycle General Electric 8x8								5% enrichment, Δk reactivity impacts							
								center				edge			
	T_{fuel}		ρ_{mod}			CB		discharge		5 yr cool		discharge		5 yr cool	
Case	ave	max	min	ave	max	in	out	8 BPRs	12 BPRs	8 BPRs	12 BPRs	8 BPRs	12 BPRs	8 BPRs	12 BPRs
base-ave	x			x			x	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	x				x		x	-0.094	-0.084	-0.092	-0.084	-0.004	-0.009	-0.007	-0.011
2	x		x				x	0.085	0.064	0.088	0.067	0.009	0.005	0.011	0.007
3		x		x			x	0.007	0.005	0.007	0.005	0.003	0.003	0.002	0.003
4		x			x		x	-0.087	-0.078	-0.087	-0.078	0.000	-0.004	-0.003	-0.007
5		x	x				x	0.088	0.067	0.092	0.070	0.010	0.007	0.012	0.009
6	x			x		x		0.014	-0.002	0.017	0.000	--		--	
7	x				x	x		-0.054	-0.069	-0.052	-0.067	--		--	
8	x		x			x		0.081	0.058	0.087	0.064	--		--	
9		x		x		x		0.019	0.003	0.022	0.004	--		--	
10		x			x	x		-0.048	-0.063	-0.047	-0.355	--		--	
11		x	x			x		0.084	0.061	0.090	0.066	--		--	

CHAPTER 5

CONCLUSIONS AND RECOMMENDATIONS

Utilizing the statistically determined bounding values, each depletion parameter was evaluated independently and collectively to determine the reactivity impact as the difference of k_{eff} of individual and combined depletion parameters. The recommended representative values for each depletion parameter that will produce bounding, positive reactivity impacts on SNF for burnup credit criticality safety evaluations are shown in Table 23 for PWRs and Table 24 for BWRs. These recommendations are based on 60 GWd/MTU at discharge and 5 year cooling for 4% and 5% U-235 enrichment.

For PWRs, the results demonstrated that the increase in reactivity impact associated with the use of bounding depletion parameters is as follows:

(1) dominated by control rod presence,

(2) notably impacted by the bounding minimum moderator density, and

(3) less impacted (< 1%) by bounding maximum fuel temperature and soluble boron concentrations.

It is recommended, when representing PWR depletion parameters, reactivity is most impacted by the combination of presence of fixed burnable absorbers (i.e., control rods), minimum moderator density at average moderator temperature, maximum fuel temperature, and maximum soluble boron concentration.

Discussed in the new ISG-8 revision review, it may be physically impossible for the fuel assembly to simultaneously experience two bounding values (i.e., the moderator temperature associated with the "hot channel" fuel assembly and the minimum specific power) (NRC 2010). In those cases, it is recommended by NRC, the evaluation should

maximize the dominate parameter and use the nominal value for the subordinate parameter (NRC 2010). Hence as the increased moderator temperature is more realistic with increased fuel temperature, the impact on reactivity increased. Therefore, it is more conservative to neglect the bounding moderator temperature as the impact is lower.

For BWRs, the results demonstrated that the increase in reactivity impact associated with the use of bounding parameters is as follows:

(1) dominated by the bounding value for minimum moderator density,

(2) notably impacted by inclusion of additional burnable absorbers (i.e., control rods),

(3) less impacted (< 1%) by bounding maximum fuel temperature.

It is recommended, when representing BWR depletion parameters, reactivity is most impacted by the combination of minimum moderator density and maximum fuel temperature. While the inclusion of control blades increases reactivity, the simultaneous inclusion with other bounding depletion parameters reduces the reactivity impact.

Table 23 PWR Depletion Parameters Recommended

Cycle length (months)	**T_{fuel} (K)**	**ρ_{mod} (g/cc)**	**T_{mod} (K)**	**S_B (ppm)**
	Max	**Min**	**Ave**	**Max**
12	1129	0.6433	560	750
18	1129	0.6433	560	1125

Table 24 BWR Depletion Parameters Recommended

Cycle length (months)	T_{fuel} (K)	ρ_{mod} (g/cc)	T_{mod} (K)
	Max	Min	Ave
12 and 18	1359	0.1741	354.3

Future Examinations

This research is not exhaustive of all depletion parameters, as the variability in design and operation is complex. Advancements of this research may include large-scale 3-D depletion calculations. A 3-D analysis could incorporate complex features of a fuel design including axially varying rod enrichments, partial length rods, and variations of operations including partial control rod insertion. These axial variations affect the axial burnup profile of the fuel, and hence the SNF reactivity.

In establishing a bounding axial burnup profile from the axial-profile database of discharged fuel, ORNL quantified the use of bounding profiles results in end effects that are generally between 1% and 4% Δk (Wagner et al. 2003b). Although the bounding axial burnup profile adds margin, as compared to the average profile approximation, additional studies should be evaluated to relate the depletion parameters collective effects with axial profile 3-D modeling.

Additionally, a database update including current higher burnup and enrichment discharged fuel assemblies would strengthen the justified conservatism of utilizing statically outlying profiles. Additionally, effects of burnable absorbers on reactivity, evaluated by ORNL for the GBC-32 cask, revealed the reactivity increase associated with burnable absorbers decreases with inclusion of the axial-burnup distribution (Parks et al. 2000b). Hence, the depletion parameter impacts will vary when evaluated in a 3-D

system.

Evaluation of parameters, such as BPR configurations and layouts, may provide a more representative impact on reactivity. Locations of BPRs in a lattice can affect the neighboring neutronics, hence, rod patterns representative of actual assembly designs would be more beneficial as the loading of BPRs is the dominant impact on reactivity.

Supplementary comparison of current operation histories to analyzed data would ensure a bounding selection of depletion parameter values. A prudent approach to burnup credit validation should involve assay data validation, followed by cross-section validation for actinides and fission products, since code calculations are limited by validation through comparison to measurement data and there remains a lack of measurement data applicable to most modern PWR and BWR designs.

APENDIX 1

NOMENCLATURE FOR UNCOMMON TERMS

ANS	American Nuclear Society
ANSI	American National Standard Institute
BPR	burnable poison rod
BWR	boiling water reactor
CRC	commercial reactor criticals
DOE	Department of Energy
EPRI	Electric Power Research Institute
g/cc	grams per cubic centimeter
GBC	generic burnup credit cask
Gd_2O_3	Gadolinium Oxide
GWd/MTU	Gigawatt days per metric ton of heavy metal
HLW	high level waste
ISG-8	Interim Staff Guidance – Eight
k_{eff}	effective neutron multiplication factor
LWR	light water reactor
MTHM	metric ton of heavy metal
MW/MTHM	Megawatt per metric ton of heavy metal
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
Pu	plutonium
PWR	pressurized water reactor
SNF	spent nuclear fuel
U	uranium
wt%	weight percent

APENDIX 2

SCALE MODEL INPUT SAMPLE

The following SCALE inputs are shown here as representative cases for the depletion calculation with the TRITON module and the criticality calculation with CSAS. The first input text describes average, base case of the Westinghouse 17x17 depletion calculation with the TRITON module. While the second input text describes the storage cell with the Westinghouse 17x17 design enclosed for the criticality calculation with CSAS. The text boxes included are intended to help familiarize an individual with the input model, but not teach the SCALE code or model scheme.

Input 1: Depletion Calculation

```
=t-depl parm=nitawl
Large scale 2-D depletion model
' PWR: Sequoyah W17x17
' performs an assembly-averaged depletion for a 1/4 assembly, with
reflected bounds
' All fuel rods are modeled with a single mixture.
'
' Tf=ave rhomod=ave cr=out
44groupndf

read comp
'Fuel
uo2 1 den=10.96 0.95 880.42
 92235 4.0 92238 96.0 end
zirc4 25 den=6.56 1 600 end
h2o 2 den=0.711 1 559.3 end
h2o 26 den=0.711 1 559.3 end
wtptbor 26 0.711 1 5000 100 545.05e-6 559.3 end
end comp

read celldata
latticecell squarepitch pitch=1.2600 26 fuelr=0.4096 1 gapr=0.4178 2
cladr=0.4750 25 end
end celldata

read depletion 1 end depletion
```

Comp defines the material specifications

Celldata defines lattice specification for cross section processing

Depletion specifies which materials to track changing isotopics

```
'60 GWd/MTU for average cycle of 354.5 days = 56.417 MW/MTU
'at discharge
read burndata
power=56.417 burn=354.5  nlib=5 end
power=56.417 burn=354.5  nlib=5 end
power=56.417 burn=354.5  nlib=5 end
power=0.0    burn=1 nlib=1 end end burndata

read model
Sequoyah W17 fuel assembly (one-fourth)

read parm
prtflux=no drawit=no  echo=yes
collapse=yes prtmxsec=no prtbroad=no epsilon=1e-3
end parm

read materials
1 1 ! 5 wt % enriched fuel ! end
2 1 !gap water! end
25 1 !cladding! end
26 1 !water! end
end materials

read geom
'unit 15 is a water hole
unit 15
cylinder 10 .5715
cylinder 20 .6121
cuboid 30 0.63 -0.63 0.63 -0.63
media 26 1 10
media 25 1 20 -10
media 26 1 30 -20
boundary 30 2 4
'unit 25 is a right-half water hole
unit 25
cylinder 10 .5715 chord +x=0.0
cylinder 20 .6121 chord +x=0.0
cuboid 30 0.63 0.0 0.63 -0.63
media 26 1 10
media 25 1 20 -10
media 26 1 30 -20
boundary 30 2 4
'unit 45 is top-half water hole
unit 45
cylinder 10 .5715 chord +y=0.0
cylinder 20 .6121 chord +y=0.0
cuboid 30 0.63 -0.63 0.63 0.0
media 26 1 10
media 25 1 20 -10
media 26 1 30 -20
boundary 30 4 2
'unit 46 is a 1/4 water hole
unit 46
cylinder 10 .5715 chord +x=0 chord +y=0
cylinder 20 .6121 chord +x=0 chord +y=0
cuboid 30 0.63 0. 0.63 0.
media 26 1 10
```

Burndata defines the cycle length, specifc power, and number of cross section updates (*nlib*)

Parm defines parameter specifications for output file prints and other code calculation bounds

Geom provides the NEWT model specification, as defined by units.

Each unit represents a portion of the assembly, then collected together by an *array* to represent the 1/4 assembly configuration.

The *global unit* specifies the outer boundary of the model, for which the boundary conditions are applied (i.e., reflective).

```
media 25 1 20 -10
media 26 1 30 -20
boundary 30 2 2
'unit 1 is a full material #1 rod
unit 1
cylinder 10 .4096
cylinder 15 .4178
cylinder 20 .4750
cuboid 30 0.63 -0.63 0.63 -0.63
media 1 1 10
media 2 1 15 -10
media 25 1 20 -15
media 26 1 30 -20
boundary 30 4 4
'unit 2 is a top-half material #1 rod
unit 2
cylinder 10 .4096 chord +y=0
cylinder 15 .4178 chord +y=0
cylinder 20 .4750 chord +y=0
cuboid 30 0.63 -0.63 0.63 0.0
media 1 1 10
media 2 1 15 -10
media 25 1 20 -15
media 26 1 30 -20
boundary 30 4 2
'unit 3 is a right-half material #1 rod
unit 3
cylinder 10 .4096 chord +x=0
cylinder 15 .4178 chord +x=0
cylinder 20 .4750 chord +x=0
cuboid 30 0.63 0.0 0.63 -0.63
media 1 1 10
media 2 1 15 -10
media 25 1 20 -15
media 26 1 30 -20
boundary 30 2 4
global unit 100
cuboid 2 10.71 0.0 10.71 0.0
array 10 1
cuboid 1 10.7518 0.0 10.7518 0.0
media 26 1 2
media 26 1 1 -2
boundary 1
end geom.

' 17x17 array
read array
ara=10 nux=9 nuy=9 pinpow=yes typ=cuboidal
fill
46 2 2 45 2  2 45 2 2
3  1 1  1 1  1  1 1 1
3  1 1  1 1  1  1 1 1
25 1 1 15 1  1 15 1 1
3  1 1  1 1  1  1 1 1
3  1 1  1 1 15  1 1 1
25 1 1 15 1  1  1 1 1
3  1 1  1 1  1  1 1 1
```

```
3  1 1  1 1  1  1 1 1
end fill
end array

read bounds all=refl end bounds


end model
end
```

Input 2: Critcality Calculation

```
=csas25   PARM=nitawl
Storage cell
44GROUPNDF5 LATTICECELL
'-----------------------------------------------------'
U-234  1 0  8.225E-08    293.15       END
U-235  1 0  9.321E-05    293.15       END
U-236  1 0  1.300E-04    293.15       END
U-238  1 0  2.123E-02    293.15       END
PU-238 1 0  9.990E-06    293.15       END
PU-239 1 0  1.402E-04    293.15       END
PU-240 1 0  7.451E-05    293.15       END
PU-241 1 0  4.367E-05    293.15       END
PU-242 1 0  2.970E-05    293.15       END
PU-243 1 0  4.558E-10    293.15       END
AM-241 1 0  1.009E-06    293.15       END
AM-242m 1 0 2.095E-08    293.15       END
AM-243 1 0  9.344E-06    293.15       END
CM-242 1 0  6.084E-07    293.15       END
CM-243 1 0  2.404E-08    293.15       END
CM-244 1 0  5.064E-06    293.15       END
CM-245 1 0  2.836E-07    293.15       END
MO-95  1 0  6.435E-05    293.15       END
TC-99  1 0  7.351E-05    293.15       END
RU-101 1 0  7.564E-05    293.15       END
RH-103 1 0  3.592E-05    293.15       END
AG-109 1 0  7.773E-06    293.15       END
CS-133 1 0  7.656E-05    293.15       END
ND-143 1 0  4.250E-05    293.15       END
ND-145 1 0  4.127E-05    293.15       END
SM-147 1 0  3.035E-06    293.15       END
SM-149 1 0  1.282E-07    293.15       END
SM-150 1 0  2.056E-05    293.15       END
SM-151 1 0  7.907E-07    293.15       END
SM-152 1 0  7.445E-06    293.15       END
EU-153 1 0  8.158E-06    293.15       END
GD-155 1 0  2.476E-09    293.15       END
O-16   1 0  4.645E-02    293.15       END
H2O      5  DEN=1.00     1         293.15       END
H2O      6  DEN=1.00     1         293.15       END
ZIRC4    7  DEN=6.56     1         293.15       END
SS304    8  DEN=7.94     1         293.15       END
B4C      9  DEN=2.644    0.291331  293.15       END
```

The specification of isotopes by atom density is the resultant isotopic inventory for the prior depletion calculation, and represents the fuel material after burnup

```
AL       9  DEN=2.644    0.708669 293.15        END
'
end comp
'
'U-238 Resonance Dancoff Correction Factor Input'
SQUAREPITCH 1.2600 0.8192 1 5 0.9500 7 0.8356 0 END
'
Storage scenario
READ PARAM  TME=500.0 NUB=YES  HTM=no
            FAR=YES GEN=2050  NPG=2000  NSK=50 RUN=YES
END PARAM

read geometry
'-------------------------------------------
' Define Pin Cells                          '
'-------------------------------------------------------'
'
unit 1
com='STD Zone 1'
cylinder   1  1 0.4096                365.76 0
cylinder   0  1 0.4178                365.76 0
cylinder   7  1 0.4750                365.76 0
cuboid     5  1 0.63 -0.63 0.63 -0.63 365.76 0
'
unit 5
com='STD Fuel Pin '
array      1 -0.63   -0.63 0
'
unit 6
com='STD G.T.'
cylinder   6 1 0.5715                365.76 0
cylinder   7 1 0.6121                365.76 0
cuboid     6 1 0.63 -0.63 0.63 -0.63 365.76 0
'
unit 7
com='STD I.T.'
cylinder   6 1 0.5715                365.76 0
cylinder   7 1 0.6121                365.76 0
cuboid     6 1 0.63 -0.63 0.63 -0.63 365.76 0
'
'-----------------------------------------------------'
' Define Lattice And Wrappers                         '
'-----------------------------------------------------'
'
unit 8
com='Assembly'
array 2     -10.71 -10.71 0
'
unit 9
com='top wrapper'
cuboid 9 1  +9.525  -9.525   +0.26924 +0.00001  365.76 0
cuboid 8 1  +9.6139 -9.6139  +0.35814 +0.00001  365.76 0
'
unit 10
com='right wrapper'
cuboid 9 1  +0.26924 +0.00001  +9.525  -9.525   365.76 0
cuboid 8 1  +0.35814 +0.00001  +9.6139 -9.6139  365.76 0
```

The lattice specification defines the appropriate use for Dancoff factor corrections and cross section processing

Parm defines parameter specifications for output file prints and other code calculation bounds

Geometry provides the KENO.Va model specification, as defined by units.

Each unit represents a portion of the assembly, then collected together by an *array* to represent the full assembly configuration.

The *global unit* specifies the outer boundary of the model, for which the boundary conditions are applied

```
'
unit 11
com='bottom wrapper'
cuboid 9 1  +9.525  -9.525   -0.00001 -0.26924  365.76 0
cuboid 8 1  +9.6139 -9.6139  -0.00001 -0.35814  365.76 0
'
unit 12
com='left wrapper'
cuboid 9 1  -0.00001 -0.26924  +9.525  -9.525   365.76 0
cuboid 8 1  -0.00001 -0.35814  +9.6139 -9.6139  365.76 0

Global unit 13
com='Fresh Fuel Assy With Can Around'
cuboid 6 1  +11.176 -11.176  +11.176 -11.176      426.72 -60.96
hole   8      0         0         0
cuboid 8 1  +11.3665 -11.3665  +11.3665 -11.3665  426.72 -60.96
cuboid 6 1  +13.843  -13.843   +13.843  -13.843   426.72 -60.96
hole   9      0       +11.3665    0
hole   10   +11.3665    0         0
hole   11     0       -11.3665    0
hole   12   -11.3665    0         0

end geometry
'
read array
'
' ---------- Pin Array ------------
'
ara=1 nux=1 nuy=1 nuz=1
com='STD Fuel Pin '
  fill
    1
  end fill
'
'-------------------------------------------------------'
'
ara=2 nux=17 nuy=17 nuz=1
com='STD Fuel Assembly'
  fill
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 6 5 5 6 5 5 6 5 5 5 5 5
    5 5 5 6 5 5 5 5 5 5 5 5 5 6 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 6 5 5 6 5 5 6 5 5 6 5 5 6 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 6 5 5 6 5 5 7 5 5 6 5 5 6 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 6 5 5 6 5 5 6 5 5 6 5 5 6 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 6 5 5 5 5 5 5 5 5 5 6 5 5 5
    5 5 5 5 5 6 5 5 6 5 5 6 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
  end fill
```

```
'
'-------------------------------------------------------'
end array
'
read bounds
  xfc=peri yfc=peri zfc=mirr
'
end bounds
'
end data
end
```

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