IMPARTING REALISM TO THE CRITICALITY EVALUATION OF A BWR FUEL ASSEMBLY PACKAGE

Peter Vescovi
Westinghouse Electric Company

Tanya Sloma
Westinghouse Electric Company

ABSTRACT

The criticality evaluation is a demonstration of the most reactive configuration of the individual package in isolation, arrays of undamaged packages, and arrays of damaged packages. The most reactive configuration for the fuel assembly contents in a BWR package must take into consideration a number of parameters that include partial length fuel rods, neutron absorbing burnable absorber rods in the fuel bundle, rearrangement of the fuel bundle during accident transport conditions in the form of lattice expansion, and partial loadings of fuel rods. Packaging material composition and arrangement of packaging materials are also important to consider in the demonstration of maximum reactivity. Values must be assigned for these parameters that may not be known with a high degree of certainty, such as burnable absorber rod distribution, lattice expansion, and packaging material composition during a fire. Imparting realism to the criticality evaluation requires a thorough understanding of the effect that impact, fire and water immersion may have on the package configuration and material properties. Evaluating the sensitivity of neutron multiplication to intrinsic material property uncertainties can be accomplished by applying perturbation methods. However, evaluation of sensitivity to other package configuration uncertainties is a more heuristic process that requires a detailed understanding of the fuel assembly design and package performance during accident transport conditions. There is no guarantee for a particular sequence of impacts or complete progression of a fire during a transport accident, yet intermediate conditions that result in the maximum neutron multiplication are often overlooked. A criticality evaluation of a BWR package has been done to demonstrate a realistic maximum neutron multiplication using values for parameters that takes into consideration credible intermediate transport conditions. Values for parameters used in the criticality evaluation are assigned in a manner consistent with constraints imposed by the fuel assembly design and performance of the contents and packaging materials during the sequence of mechanical, thermal and water immersion tests.

INTRODUCTION

A summary view of nuclear criticality safety is characterized by known and acceptable risk associated with transport conditions and identification of potential criticality for credible package configurations (Figure 1). Transport regulations define normal and accident transport conditions that enable the package evaluation to be limited to identification of potential criticality [1]. The identification of potential criticality is based on the results of transport condition tests that are performed on a package or simulated with computational modeling software. Nuclear analysis considers the package configurations, adequate optimization, and other estimations that are consistent with the consequences of transport conditions known from testing or simulation.

Transportation safety for fissile material packages is best served when identification of potential criticality is based upon realistic assumptions for criticality parameters and credible nuclear analysis.

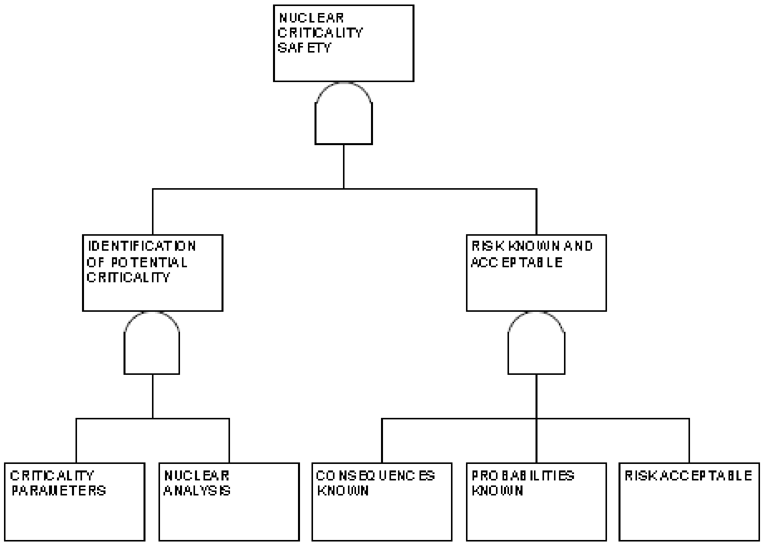

Figure 1. Summary view of nuclear criticality safety
(Source: The Radioactive Materials Packaging Handbook, ORNL/M5003)

IDENTIFICATION OF POTENTIAL CRITICALITY-NUCLEAR ANALYSIS AND CRITICALITY PARAMETERS

A criticality evaluation of a fissile material package should demonstrate a maximum neutron multiplication by using realistic values for parameters and taking into consideration credible transport conditions including any credible intermediate conditions. The packaging and contents are controlled by the design specifications, and this known configuration should be represented in the nuclear analysis. Values for parameters used in the criticality evaluation should be assigned in a manner consistent with constraints imposed by the fuel assembly design and performance of the package during the prescribed sequence of mechanical, thermal and water immersion tests.

The BWR packaging used in this review consists of inner and outer containers that retain the contents within a fixed geometry relative to other such packages in an array. The radioactive contents consists of a fuel assembly with structure that retains the fuel rods within a fixed geometry. Individual fuel rods retain the fuel pellets within a fixed geometry of a fuel rod tube. Therefore, the confinement system is known to consist of the inner and outer containers, fuel assembly structure, and the fuel rod tube.

Neutron absorption is provided by packaging materials and burnable neutron absorbers present in the fissile fuel mixture. The packaging may not have specific design features that provide neutron moderation and absorption for criticality control. However, neutron absorbers in the structural components and contents provide significant neutron absorption that is considered in the criticality safety evaluation.

Internal moderation is provided by packaging materials such as paper honeycomb, wood, and polyethylene, but none of these materials are present in a configuration to provide the sufficient neutron moderation required for effective neutron absorption or multiplication, with exception of accident conditions for air transport. Hence, neutron moderation from external sources is required to have significant neutron multiplication. Because the water immersion test is not performed, assumptions are made about leakage of water into and out of the package void spaces that are subject to engineering judgment. Adequate assumptions are made for optimizing neutron moderation from internal and external sources that are consistent with the known transport conditions and laws of nature.

Possible configurations of the radioactive contents and packaging (single package, arrays of undamaged packages, and arrays of damaged packages) that are consistent with each condition of transport are evaluated. The most reactive contents are evaluated with the packaging to identify the optimum combination of packaging materials, internal moderation and interspersed moderation. The most reactive configuration for each type of fuel assembly contents takes into consideration partial length fuel rods in fuel bundle, neutron absorbing BA rods in the fuel bundle, and rearrangement of the fuel bundle in the form of lattice expansion during accident transport conditions. Fuel rearrangement is limited by the fuel bundle structure, fuel assembly structure, or inner wall of the inner container. First, the fuel bundle structure (tie plates, spacer grids) confines fuel rods to a nominal pitch during normal transport conditions. Second, rearrangement of the bundle lattice resulting from an impact consistent with accident transport conditions is confined by the fuel channel for fuel assembly contents. Third, the inner wall of the inner container provides confinement for fuel bundle contents or fuel rods without the rod container.

Arrangement of radioactive contents and packaging material composition are important to consider in the optimization of reactivity. Since there is no guarantee of a particular sequence of impacts or the complete progression of a fire during a transport accident, intermediate conditions that may result in the maximum neutron multiplication are considered. Examples of the rational for parameters that are not known with a high degree of certainty, such as burnable absorber rod distribution, packaging material composition during a fire, fuel bundle lattice expansion, criticality parameter uncertainty, and margin of subcriticality are provided in the discussion that follows.

Burnable absorber rod distribution

Burnable absorber (BA) rods that are used to extend the life of the fuel bundle during the power generation cycle also provide neutron absorption for transport conditions where moderation of the fuel occurs. Internal sources of moderation from polyethylene packaging materials such as foam, protective spacers, cluster separators, and sheathing, or water from external sources are credible sources of moderation for the fuel bundle. The effectiveness of the BA rods as a neutron absorber is significant in a moderated fuel bundle, but the relative efficacy as a neutron absorber varies sensitively with the location of the BA rod within the fuel bundle lattice. In order to evaluate the relative efficacy of BA rods, neutron absorption in the gadolinium is assessed at each location within a fuel bundle lattice.

A sensitivity analysis based on analytical perturbation methods is used to select the BA rod locations. Constraints that are consistent with the design objectives for a BWR fuel assembly are as follows:

1) *Rule of symmetry* -BA rods shall be in positions that are symmetric across the geometric major diagonal
2) No BA rod shall be located in the outermost edge or corner location of the fuel rod lattice
3) Partial length fuel rods shall not be BA rods.
4) At least one BA rod shall be located in three of the four fuel lattice quadrants.
5) There shall be at least 8 BA rods in the fuel bundle.

Applying these rules in the selection process results in a pattern of burnable absorber rods that is not the most reactive conceivable arrangement nor an actual rod pattern expected in the fuel design, but rather represents a pattern that provides adequate neutron absorption and acknowledges realistic constraints imposed by the fuel bundle design.

Packaging material composition during a fire

Intermediate conditions result from the transition of packaging materials compositions or phases changes that occur during a fire. The combustion or redistribution of packaging material during the fire are considered in the evalutaon, because the nuetron multiplication may be larger for the intermeditate condition as compared to the final state.

Water or void is commonly assumed to fill the void space left by the complete combustion of impact absorber material. However, thermal testing and analysis demonstrate that impact absorber material (paper honeycomb, balsa wood) may undergo only partial combustion during a fire. The chemical composition of impact absorber material is carbon (C), hydrogen (H), and oxygen (O). Char is produced in the absence of oxygen by the slow pyrolysis of the impact absorber material. By the action of heat, charring removes hydrogen and oxygen from the solid so that the remaining char is composed primarily of carbon. Carbon at the original density is assumed to evaluate the effect that incomplete combustion has on neutron multiplication.

Moderating ratio increases when char displaces water in the package or char instead of void is assumed to remain in the package. Moderating ratio is a measure of the effectivensess of neutron scattering in the packaging materials and contents to slow down neutrons to thermal energies. Chromium in the stainless steel package stucture and fissile uranium in the contents compete for absorbtion of neutrons during the slowing down process. An increase in moderating ratio for the individual package configuration results in preferential absorption in the fissile uranium contents due to the limited quantity of stainless steel, and the neutron multiplication increases as compared to a reference configuration with water instead of char. An increase in moderating ratio for the package array results in preferential absorption in the stainless steel due to the large amount of neutron interaction between packages as neutrons slow down. The multiplication factor for the package array decreases as compared to a reference configuration with void instead of char. An intermdiate material condition due the incomplete combustion of impact absorber material can result in a maximum neutron multiplication that may otherwise have been overlooked if complete combustion is assumed.

During a fire, redistribution of moderating materials such as polyethylene packing materials may also provide moderation of the contents that results in an increase, decrease or no significant change in neutron multiplication. Packaging materials normally present and contiguous with the contents such as polyethylene cluster separators, spacers, and wrap are considered for all transport conditions. The effect on moderation by these packing materials is evaluated by assuming that these materials are uniformly distributed on the fuel rod outer surface regardless of the condition of transport. The effect of polyethylene foam cushion that may melt during accident conditions and provide additional moderation within the fuel bundle is also considered in the evaluation. An intermediate condition such as the accident transport condition prior a fire or the absence of a fire, results in a credible configuration where foam material remains rather than becoming a space filled with water during immersion. This configuration with the foam intact results in more neutron interaction for the accident package array than if the foam were assumed to melt and be replaced by water. Therefore, this intermediate accident condition for foam can not be ignored as it is a credible accident condition that results in the maximum neutron multiplication.

<u>Fuel bundle lattice expansion</u>

Tests demonstrate that virtually all fuel rod deformations induced from an axial impact are due to interactions between the end of the fuel rod and the deformed nozzles. BWR fuels are designed to be under moderated, hence an impact event which increases the pin pitch results in an increase in reactivity.

It has been observed that for BWR fuel subject to end impacts, the lattice may contract near the impacted end but expand slightly in the adjacent intra-grid length as shown in Figure 2. Relying only on the fuel bundle structure for confinement, a mean lattice pitch change of less than 5 mm is predicted by static analysis methods between the second and third spacer grids from the bottom of the fuel assembly [2]. Nominal dimension between the second and third grid is less than 50 cm for BWR fuel assemblies. Analyzed performance of the lower tie plate and cladding during an end impact predicts responses similar to that observed in mechanical tests. The analysis concludes that the lower tie plate will not fail during an end drop and the cladding will not rupture due to the rod bowing. The testing and analytical results justify the assumptions that the individual fuel pellets are contained in the cladding and no water can leak into the void space between fuel pellet and cladding during accident transport conditions.

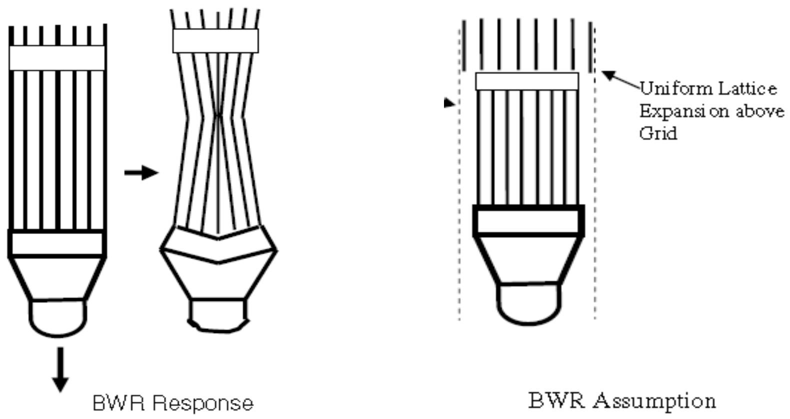

Figure 2. Effect of end impact of BWR fuel bundle

The criticality analysis ignores lattice contraction near the end but does consider the uniform lattice expansion above the first grid. The BWR fuel assembly is evaluated to determine the maximum reactivity due to an increase in lattice pitch that is confined to a length of 50 cm near the end of the fuel bundle. This assessment is done for a range of fuel rod pitch that includes the dimensions that are associated with each confinement boundary (nominal fuel bundle, fuel channel, inner container). Lattice expansion for a fuel bundle shipped with the fuel channel installed, referred to as the fuel assembly, is confined to the fuel channel. A fuel bundle not confined by the fuel channel can expand to the inside dimension of the inner container. The nuclear analysis demonstrates that a allowed package array size is dependent on the extent of the lattice expansion. By recognizing this realistic difference in the confinement boundary for a fuel bundle as compared to the fuel assembly, a smaller criticality safety index (CSI) is possible when the fuel channel is present.

<u>Criticality parameter uncertainties</u>

TS-G-1.1, Appendix VI, suggests that "the criticality section of the SAR should address dimensional tolerances of the packaging, including components containing neutron absorbers," and "the range of material specifications and associated uncertainties should be used to select parameters that produce the highest reactivity."[3] The sensitivity of neutron multiplication to material property uncertainties was accomplished for the BWR package by applying perturbation methods to evaluate the effect of small variations in dimensions or material specifications. An allowance (Δk_u) is determined that covers the change in neutron multiplication due to uncertainty in a criticality parameter.

The value for Δk_u was calculated for each package configuration and transport condition. The allowance for package material uncertainties can be significant, and the value for Δk_u depends on both the package configuration and transport condition. Approximately half of the total allowance is attributed to material and fabrication tolerances (i.e. fuel rod dimension, fuel rod pitch, stainless steel sheet thickness) with the remainder contributed by uncertainty in geometric or material representation (i.e. container spacing, polyethylene redistribution). The greatest contributors to the total uncertainty for package configurations and conditions of transport are the uncertainty in polyethylene thickness on fuel rods, thickness of packaging stainless steel sheet, and fuel rod pitch. The values for Δk_u for package array configurations are larger due to the greater sensitivity of neutron multiplication to uncertainties that affect interaction between packages.

The total allowance, Δk_u is the sum of allowances for uncertainties in material and fabrication tolerances and uncertainties due to limitation in the geometric or material representations used in the computational method. A maximum or minimum value for a parameter may produce a positive Δk_u. For example, the positive tolerance on diameter of a fuel rod results in a positive Δk_u where as the same positive tolerance applied to a BA fuel rod results in a negative Δk_u. Therefore, parameters for contents such as fuel rods should be specified as a tolerance applied to a nominal value instead of just a maximum or minimum dimension. The uncertainties were combined additively with the assumption that they are correlated. Realistically, there would be some uncertainties that are independent and could be combined statistically.

Margin of subcritiality

TS-R-1 requires that "fissile material shall be transported so as to maintain subcriticality during normal and accident conditions of transport."[4] The criteria to establish subcriticality of the package includes calculation of a bias, an allowance for uncertainty in the bias, and an administrative margin to ensure subcriticality of the package. The regulatory requirement for subcriticality does not specify a minimum administrative margin of subcriticality for safety (Δk_m).

The value for Δk_m can have a significant effect on the package design and operation of a package in terms of the criticality safety index (CSI). There is some guidance as to what constitutes sufficient technical justification for the minimum administrative margin of subcriticality. U.S. NRC recommends a minimum administrative margin of subcriticality no less than 0.05 in the application for approval of packages and advises the applicant to consider that the value may need to be increased by an arbitrary amount if there is lack of sufficient critical data to adequately determine the calculation bias and uncertainty [5]. Hence, a Δk_m value equal to 0.05 is commonly assigned in the package application with no justification. Other guidance suggests a Δk_m value less than 0.05 is possible with technical justification [3]. The BWR package evaluation not only provides technical justification for a Δk_m value as low as 0.02, but demonstrates that the Δk_m value depends on the package configuration and conditions of transport. A statistical method was used to demonstrate that a Δk_m value as small as 0.02 is adequate for a given set of critical experiments used in the validation [6]. Applicants should be allowed to apply statistical methods to provide a technical justification for an administrative margin, instead of defaulting to the use of an arbitrary value for Δk_m.

Establishment of a range of applicability is commonly based on a characteristic criticality parameter such as hydrogen-to-fissile ratio or the average energy causing fission. For the BWR package evaluation, TSUNAMI modules provided in SCALE 6 were used to calculate sensitivity and uncertainty data for each of the critical experiments and the package[7]. TSUNAMI-IP was used to calculate global indices that assess the similarity of the package and critical experiments on a system wide basis for all nuclides and reactions. A set of integral indices, c_k values were calculated for each package configuration. The interpretation of the integral index, c_k, is the following, a value of 0.0 represents no correlation between the package configuration and critical experiment and a value of 1.0 represents full correlation between the systems.

Critical experiment cases selected from benchmarks for light-water-reactor fuel in transportation packages[6] and critical experiments performed for actual BWR fuel configurations with burnable neutron absorber rods[8]. Benchmarks with c_k greater than 0.80 were included to predict the upper subcritical limit (USL). The actual BWR fuel assembly benchmarks were expected to show a high correlation with the package configurations, but tended to have c_k values less than 0.80 indicating a low correlation. Critical experiments with materials similar to those used for the BWR packaging and contents tended to show higher correlation to the package application, instead of critical experiments with similarity to only to contents. In addition, each package configuration showed different sensitivities to the nuclear cross section data that affect the determination of the USL.

CONCLUSIONS

This review of the criticality evaluation for a BWR fuel assembly package highlights the importance of imparting realism to the identification of potential criticality for fissile packages.

Optimization of the contents and packaging accounted for constraints inherent in the fuel assembly and packaging design such as confinement for lattice expansion and burnable absorber rod distribution. The evaluation of packaging materials during the sequence of accident transport tests demonstrated that an intermediate condition can result in a higher neutron multiplication than for the final package condition. Nominal package configurations consistent with transport conditions were used for the nuclear analysis. Allowances for uncertainties in material and fabrication tolerances and uncertainties due to limitations in material and geometric representations may be significant. The effects of these uncertainties were calculated and allowances were made to determine the maximum neutron multiplication for the package. Acceptable margin to subcriticality was shown to be dependent on the package configuration and transport condition. Although the recommended administrative margin of subcriticality equal to 0.05 Δk was used to comply with regulatory guidance, a value of 0.02 is acceptable and the smaller margin should be allowed based on the statistical technical justification. Upper subcritical limits were shown to be dependent on the combination of package configuration and transport condition. The consideration of realistic criticality parameters and credible package configurations that are consistent with transport conditions had a significant effect on the calculation of the maximum neutron multiplication, and this resulted in specifying CSI values that optimize the operation of the BWR fuel assembly package. The criticality evaluation of the BWR fuel assembly package provides a view of nuclear criticality safety that includes realistic criticality parameters and credible nuclear analysis. Transportation safety is best served when limits for package operation are based upon realistic calculations.

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