UNITED STATES NUCLEAR REGULATORY COMMISSION

Notice of Availability and Opportunity for Comment on

Draft Division of Safety Systems Interim Staff Guidance DSS-ISG-2010-01:

Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools

AGENCY: Nuclear Regulatory Commission.

ACTION: Solicitation of Public Comment

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) requests public comment on a draft Division of Safety Systems Interim Staff Guidance, (DSS-ISG) DSS-ISG-2010-01, “Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools.” This draft DSS-ISG provides updated guidance to the NRC staff reviewer to address the increased complexity of recent spent fuel pool (SFP) license application analyses and operations. The guidance is intended to reiterate existing guidance, clarify ambiguity in existing guidance, and identify lessons learned based on recent submittals.

DATES: Comments may be submitted by (insert 30 days after publication in the Federal Register). Comments received after this date will be considered, if it is practical to do so, but only comments received on or before this date can be assured consideration.

ADDRESSES: You may submit comments by any one of the following methods. Please include Docket ID NRC-2010-XXXX in the subject line of your comments. Comments submitted in writing or in electronic form will be posted on the NRC website and on the Federal rulemaking website http://www.regulations.gov. Your comments will not be edited to remove any identifying or contact information, therefore, you should not include any information in your comments that you do not want publicly disclosed.

The NRC requests that any party soliciting or aggregating comments received from other persons for submission to the NRC inform those persons that the NRC will not edit their
comments to remove any identifying or contact information, and therefore, they should not include any information in their comments that they do not want publicly disclosed.

**Federal Rulemaking Website:** Go to [http://www.regulations.gov](http://www.regulations.gov) and search for documents filed under Docket ID NRC-2010-XXXX. Address questions about NRC dockets to Carol Gallagher 301-492-3668; e-mail Carol.Gallagher@nrc.gov.

**Mail comments to:** Cindy K. Bladey, Chief, Rules, Announcements and Directives Branch, Office of Administration, Mail Stop: TWB-05-B01M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by fax to RADB at (301) 492-3667.

You can access publicly available documents related to this notice using the following methods:

**NRC's Public Document Room (PDR):** The public may examine and have copied for a fee publicly available documents at the NRC's PDR, Room O1 F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland.

**NRC's Agencywide Documents Access and Management System (ADAMS):**

Publicly available documents created or received at the NRC are available electronically at the NRC's Electronic Reading Room at [http://www.nrc.gov/reading-rm/adams.html](http://www.nrc.gov/reading-rm/adams.html). From this page, the public can gain entry into ADAMS, which provides text and image files of NRC's public documents. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's PDR reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to [pdr.resource@nrc.gov](mailto:pdr.resource@nrc.gov). The Staff Guidance Regarding the Nuclear Criticality Safety Analysis Accompanying Spent Fuel Pool License Amendment Requests, DSS-ISG-2010-01, is available electronically under ADAMS Accession Number ML102220567.

**Accessing non-Publicly Available Documents:** The draft DSS-ISG-2010-01 does not include any non-publicly available documents.
Federal Rulemaking Website: Public comments and supporting materials related to this notice can be found at http://www.regulations.gov by searching on Docket ID: NRC-2010-XXXX.

FOR FURTHER INFORMATION CONTACT: Kent A. L. Wood, Reactor Systems Engineer, Reactor Systems Branch, Division of Safety Systems, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Rockville, Maryland 20852. Telephone: (301) 415-4120; fax number: (301) 415-3577; e-mail: Kent.Wood@nrc.gov.

SUPPLEMENTARY INFORMATION:

The NRC is issuing this notice to solicit public comments on the draft DSS-ISG-2010-01, “Staff Guidance Regarding the Nuclear Criticality Safety Analysis Accompanying Spent Fuel Pool License Amendment Requests.” After the NRC staff considers any public comments received, it will make a determination regarding issuance of the proposed DSS-ISG.

Dated at Rockville, Maryland this 25th day of August 2010

For the Nuclear Regulatory Commission

/RA/

William H. Ruland, Director
Division of Safety Systems
Office of Nuclear Reactor Regulation
I  INTRODUCTION

This U.S. Nuclear Regulatory Commission (NRC) Division of Safety Systems (DSS) interim staff guidance (ISG) provides updated guidance to the NRC staff reviewer to address the increased complexity of recent spent fuel pool (SFP) license application analyses and operations. The guidance is intended to reiterate existing guidance, clarify ambiguity in existing guidance, and identify lessons learned based on recent submittals. The current guidance appears in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” particularly Section 9.1.1, “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” Revision 3, issued March 2007 (Reference 1). Section 9.1.1 provides the existing recommendations for performing the review of the nuclear criticality safety (NCS) analysis of SFP. Additional guidance appears in the NRC memorandum from L. Kopp to T. Collins, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” dated August 19, 1998 (Reference 2), and an NRC letter from B.K. Grimes to all power reactor licensees, “OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,” dated April 14, 1978 (Reference 3). The guidance in these documents remains applicable, with the exception of the guidance set forth concerning the determination of the criticality code methodology uncertainty in Reference 2; see Section 4.c for more details.
The guidance in this draft DSS-ISG-2010-01 is to be used by NRC staff to review: (i) future applications; and (ii) future licensee applications for license amendments and requests for exemptions from compliance with applicable requirements.

II DISCUSSION

The applicable regulatory requirements for criticality safety analysis for spent fuel pools are contained in 10 CFR50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants (Reference 4) Criterion 62, Prevention of Criticality in Fuel Storage and Handling, 10 CFR 50.68, Criticality Accident Requirements (Reference 5), and 10 CFR 70.24 Criticality Accident Requirements (Reference 6). GDC 62 provides that “criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.” 10 CFR 50.68 (b)(4) provides that “if no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.” SFPs are subject to the requirements in 10 CFR 50.68 or 10 CFR 70.24. Rather than specifying a limit on the estimated ratio of neutron production to neutron absorption and leakage (k-effective, $k_{\text{eff}}$), 10 CFR 70.24 requires controls to be in place to detect and mitigate the consequences of an inadvertent criticality event. However, licensees licensed under 10 CFR 70.24 typically have an exemption that requires that the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.
Commercial reactor licensees use SFPs to store unirradiated fresh fuel and irradiated spent nuclear fuel (SNF). The SFPs were initially intended to hold a limited number of fuel assemblies to facilitate refueling operations while allowing the decay heat from SNF to dissipate before shipping the fuel assemblies off site for reprocessing or storage by the U.S. Department of Energy.

Since there is currently no means of reprocessing SNF and the U.S. Department of Energy is not accepting the SNF, licensees have increased their onsite storage capacity. Increasing the storage capacity in the existing SFP was the first step in increasing onsite storage capacity. Licensees transitioned from low-density storage, relying on flux traps caused by the large center-to-center spacing of the fuel assemblies, to high-density storage relying on installed neutron absorbers to accommodate the reduced center-to-center spacing of the fuel assemblies. However, virtually every permanently installed neutron absorber for which a history can be established has degraded in the SFP environment. If that degradation results in a reduction in the neutron absorption capability, reactivity will increase.

Other factors affecting reactivity in the SFP have not been static. The fuel assemblies have become more reactive. Increased uranium-235 enrichment is an example. Other changes include increased fuel pellet diameter, increased fuel pellet density, increased use of fixed and integral burnable absorbers, and changes to core operating parameters because of power uprates that result in more reactive fuel assemblies to be stored in the SFP.

To accommodate these effects, the SFP NCS analyses and operation have become more complex. SFP NCS analyses are taking credit for items that previously were not part of such an analysis. For example, recent license amendment requests (LARs) have credited various combinations of the following: plutonium-241 decay, americium-241 buildup, axial blankets, integral burnable poisons on fresh fuel assemblies, increased burnup (as high as 78 gigawatt day/metric ton uranium (GWD/MTU)). The proposed storage configurations are becoming more complicated. Previously, each rack design in the SFP would have one storage
configuration. Now, it is not uncommon for a rack design to have multiple sets of storage configurations. These storage configurations and the controls necessary to maintain the approved configuration are essential parts of the SFP NCS analysis.

III APPLICABILITY

The guidance in this draft DSS-ISG-2010-01 is to be used by NRC staff to review: (i) future applications; and (ii) future licensee applications for license amendments and requests for exemptions from compliance with applicable requirements.

IV TECHNICAL GUIDANCE

1. Fuel Assembly Selection: Licensees typically have used more than one fuel assembly design. Whether an applicant has one or many fuel assembly designs, the staff should review the submittal to verify that it demonstrates that the NCS adequately bounds all designs, including variations within a design. Some of the potential variations within a design include axial blankets, cutback regions, axial enrichment zoning, radial enrichment zoning, and integral burnable neutron absorber loading. Therefore, the staff should verify each application includes a portion of the analysis that demonstrates that the fuel assembly used in the analysis is appropriate for the specific conditions.

   a. Use of a single “limiting” fuel assembly design should be assessed, as recent applications have shown that the limiting fuel assembly design can change based on the effects of other parameters in the analysis.

2. Depletion Analysis: NCS analysis for SNF for both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) typically includes a portion that simulates the use of fuel in a reactor. These depletion simulations are used to create the isotopic number densities used in the criticality analysis.

   a. Depletion Uncertainty: The Kopp memorandum (Reference 2) states the following:
A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

The staff should use the Kopp memorandum as follows:

i. "Depletion uncertainty" as cited in the Kopp memorandum should only be construed as covering the uncertainty in the isotopic number densities generated during the depletion simulations.

ii. The "reactivity decrement" should be the decrement associated with the $k_{\text{eff}}$ of a fresh unburned fuel assembly that has no integral burnable neutron absorbers, to the $k_{\text{eff}}$ of the fuel assembly with the burnup of interest either with or without residual integral burnable neutron absorbers, whichever results in the larger reactivity decrement.

b. \textit{Reactor Parameters}: Consistent with the guidance in the Kopp memorandum for "the spent fuel storage racks loaded with fuel of the maximum permissible reactivity," the depletion simulations should be performed with parameters that maximize the reactivity of the depleted fuel assembly. Several reactor parameters, when modeled in the depletion simulations, affect the reactivity of the discharged fuel assemblies. NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," issued February 2000 (Reference 7), provides some discussion on the treatment of depletion analysis parameters for PWRs. While NUREG/CR-6665 is focused on criticality analysis in storage and transportation casks, the basic principles with respect to the
depletion analysis apply generically to SFPs, since the phenomena occur in the reactor as the fuel is being used. Although a useful reference on the subject, NUREG/CR-6665 is not an exhaustive study of all of the fuel designs, core operating parameters, storage conditions, and possible synergistic effects. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates that the reactor parameters used in the depletion analysis are appropriate for the specific conditions. The staff reviewer should consider the following:

i. It may not be acceptable to use nominal or typical values, because this would not ensure that the calculated $k_{eff}$ meets the 10 CFR 50.68 requirements for a 95 percent probability at a 95 percent confidence level. Bounding values should be used, and they should be traceable to other licensee documents.

ii. 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). In those cases, the application should maximize the dominate parameter and use the nominal value for the subordinate parameter. Where this is done, the application should describe and justify the parameters used.

iii. When sensitivity studies are performed to determine the limiting parameter, they should include the synergistic effects of other variables.

c. **Burnable Absorbers:** Fixed burnable absorbers are those that are inserted into or attached to a fuel assembly for a complete reactor operating cycle, but they can be readily removed. Integral burnable absorbers refer to burnable poisons
that are physically part of the as-manufactured fuel assembly. NUREG/CR-6665 provides a brief discussion on fixed and integral burnable absorbers. NUREG/CR-6760, “Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit,” issued March 2002 (Reference 8), and NUREG/CR-6761, “Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit,” issued March 2002 (Reference 9), provide a more detailed discussion. Although these documents are useful references on the subject, they are not exhaustive studies of all of the fuel designs, core operating parameters, storage conditions, and possible synergistic effects. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates that the treatment of burnable absorbers in the depletion analysis is appropriate for the specific conditions. For example, the reviewer should consider the following:

i. Use of the limiting fixed burnable absorber applicable to their specific conditions. The reviewer should also recognize that while fixed burnable absorbers are typically used to control power shaping or peaking in the reactor, they have also been used for other purposes (e.g., flux suppressors to reduce the neutron fluence on reactor belt welds). Applications should consider all fixed burnable absorbers that have been used or are predicted to be used at their facilities.

ii. Use of the limiting integral burnable absorber applicable to their specific conditions.

iii. Burnable absorbers are modeled appropriately. For example, modeling burnable absorbers as full length when they are actually part length may lead to nonconservative conclusions about their effect on SFP reactivity.
iv. Competing effects are considered, such as the depletion of the burnable absorber and the increased rate of plutonium production from increased fast neutron capture in uranium-238.

d. *Rodded Operation:* Rodded operation would affect reactivity in a manner similar to fixed burnable absorbers. Since rodded operation has the potential to affect the discharge reactivity of the fuel assemblies, it should be considered. NUREG/CR-6759, “Parametric Study of the Effect of Control Rods for PWR Burnup Credit,” issued February 2002 (Reference 10), provides a more detailed discussion. Although this document is a useful reference on the subject, it is not an exhaustive study of all of the fuel designs, core operating parameters, storage conditions, and possible synergistic effects. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates its treatment of rodded operation is appropriate for its specific conditions.

3. **Criticality Analysis**

a. *Axial Burnup Profile:* One of the most important aspects of fuel characterization is the selection of the axial burnup profile. NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” issued March 2003 (Reference 11), provides an insightful discussion of the "end effect" and recommendations for selecting an appropriate axial burnup profile. Although NUREG/CR-6801 is a useful reference on axial burnup profiles, it is not an exhaustive study of all of the fuel designs, core operating parameters, storage conditions, and possible synergistic effects. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates its treatment of axial burnup profile is appropriate for its specific conditions. For example, the reviewer should consider the following:
i. Use of the limiting axial burnup distributions from NUREG/CR-6801 are acceptable for existing PWRs, provided they are used in a manner consistent with NUREG/CR-6801. The NRC staff reviewer should verify the applications for plant designs that set the limiting profiles in NUREG/CR-6801 provide a site specific justification for the axial burnup distributions.

ii. Applications using site-specific profiles should consider all past and present profiles, and include licensee controls to ensure that future profiles are not more reactive.

iii. Use of uniform profiles is conservative at low burnup levels. At some amount of burnup, the use of a uniform profile will become nonconservative. The burnup point where that occurs is dependent on the specifics of the situation. Applications that use uniform axial burnup profiles should clearly demonstrate where that occurs.

b. **Rack Model:** The rack model consists of the dimensions and materials of construction, including any installed neutron absorber. Given all the combinations that are in existence, it is impossible to predict all of the combinations that could be proposed. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates that the rack model analysis used in its submittal is appropriate for its specific conditions. For example,

i. The dimensions and materials of construction should be traceable to licensee design documents.
ii. The efficiency of the neutron absorber should be established, especially considering the potential for self-shielding and streaming.

iii. Any degradation should be modeled conservatively, consistent with the certainty with which the material condition can be established.

c. **Interfaces:** For applications that contain more than a single storage configuration, in order to ensure that the regulatory requirement for $k_{eff}$ to be known with a 95 percent probability at a 95 percent confidence level is met the NCS should consider the interface between storage configurations. Given all the combinations that are in existence, it is impossible to predict all of the combinations that could be proposed. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates that the interface analysis used is appropriate for its specific conditions.

i. Absent a determination of a set of biases and uncertainties specifically for the combined interface model, use of the maximum biases and uncertainties from the individual storage configurations could be acceptable in determining whether the $k_{eff}$ of the combined interface model meets the regulatory requirements.

d. **Normal Conditions:** The static condition where all fuel assemblies are in approved storage locations is not the only “normal” condition. Movement of fuel in and around the SFP is a normal operation, as are other activities such as fuel inspections and reconstitution, and should also be treated as normal conditions in the NCS analysis. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates that the NCS considers all appropriate normal conditions for its specific conditions.
e. **Accident Conditions:** The Kopp memorandum states, “The criticality safety analysis should consider all credible incidents and postulated accidents.” Typically analyzed accident conditions include misplacement or drop of a fuel assembly alongside the storage rack, misloading of a fuel assembly into an unapproved location, loss of SFP cooling, and boron dilution. The reviewer should verify all credible accident conditions are addressed. If an application determines that based on site specific rationale an accident condition is not credible, the submittal should include an analysis that quantitatively evaluates the probability of occurrence for that event.

i. Accidents should be considered with respect to all normal conditions.

4. **Criticality Code Validation:** The Kopp memorandum states the following:

   The proposed analysis methods and neutron cross-section data should be benchmarked, by the analyst or organization performing the analysis, by comparison with critical experiments. This qualifies both the ability of the analyst and the computer environment. The critical experiments used for benchmarking should include, to the extent possible, configurations having neutronic and geometric characteristics as nearly comparable to those of the proposed storage facility as possible.

NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” issued January 2001 (Reference 12), provides a more detailed discussion. Although a useful reference on the subject, NUREG/CR-6698 focuses on nuclear fuel cycle facilities and may not be all-inclusive with respect to a validation intended for fuel stored in an SFP.

a. **Area of Applicability:** The area of applicability is where the application demonstrates that the experiments cover the range of the analyzed system's
parameters. Experiments should fully cover the range of the analyzed system. If the experiments do not fully cover the analyzed system, then the results should be extrapolated. Therefore, the staff should verify that applications demonstrate that the validation fully covers the area of applicability for their specific SFP;

i. The reviewer should verify that any validation that used for SNF appropriately considers actinides and fission products. NUREG/CR-6979, “Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data,” issued September 2008 (Reference 13) provides experiments that model the actinide content of PWR fuel. Not all experiments may be appropriate for use by every application; the NRC staff reviewer should assess the appropriateness of the experiments used.

ii. Experiments should be appropriate to the system being analyzed. For example, an SFP without soluble boron should not have experiments with soluble neutron absorbers, and fresh-fuel-only NCS analyses should not have mixed oxide and HTC experiments. Parameters in the experiments should bound those of the system being analyzed. Experiments with parameters significantly in excess of those of the system being analyzed should be scrutinized for possible deleterious effects on the validation.

iii. The reviewer should recognize that too few experiments may not be statistically significant to cover the parameters and may lead to invalid trend analysis conclusions.

b. Trend Analysis: Part of the validation is to identify whether the bias or bias uncertainty or both have a dependency on any of the parameters in the area of applicability. Linear regression is typically used in the trend analysis. However,
it is not the only method for investigating trends, and in some cases it may not be the best method. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates that the trend analysis used in its validation is appropriate for its specific conditions: For example, the staff should consider whether

i. A trend analysis was performed on each parameter used to define the area of applicability.

ii. The submittal states and justifies its criteria for accepting or rejecting hypothesized trends.

iii. Identified trends are fully evaluated and appropriately applied.

c. **Statistical Treatment:** The products of the validation are a methodology bias and bias uncertainty. The Kopp memorandum states the following:

   The benchmarking analyses should establish both a bias (defined as the mean difference between experiment and calculation) and an uncertainty of the mean with a one-sided tolerance factor for 95-percent probability at the 95-percent confidence level (Ref. 8).

However, this use of the “uncertainty of the mean” does not ensure that $k_{\text{eff}}$ is known with a 95 percent probability at a 95 percent confidence level such that any single calculation that calculates as subcritical is indeed subcritical. Use of the “uncertainty of the mean” may not be consistent with other statements in the Kopp memo and does not guarantee compliance with the requirements of 10 CFR 50.68. Use of the uncertainty of the mean would be inconsistent with NUREG/CR-6698. Recent applications related to spent fuel pool criticality have used a methodology consistent with NUREG/CR-6698 regarding the
development of the code bias and bias uncertainty that has been accepted by the staff. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates that the statistical treatment used in its validation is appropriate for its specific conditions. The staff should consider whether

i. Applications use the variance of the population about the mean, instead of the variance of the mean.

ii. Appropriate confidence factors are used when determining the 95 percent probability and 95 percent confidence level.

iii. Nonnormal distributions are treated using appropriate statistical methods.

d. Lumped Fission Products: Vintage depletion codes use lumped fission products to collectively model isotopes of lesser importance in the reactor environment. It is not clear how the lumped fission products will behave in the environmental conditions of the SFP. Therefore, the staff should verify that each application that includes lumped fission products includes a portion of the analysis that demonstrates that the lumped fission products used in its validation are appropriate for its specific conditions. For example,

i. There are no critical experiments or cross-section libraries with lumped fission products, so transferring their number densities and cross-sections into the criticality code will require an extrapolation in the validation.

ii. Replacing the lumped fission products with a quantity of a known isotope, such as boron-10, that results in an equivalent reactivity at some statepoint is an assumption that the substitute isotope is an adequate representation of the actual isotopes represented by the lumped fission products. This assumption will require an extrapolation in the validation.
e.  **Code-to-Code Comparisons:** The Kopp memorandum states that “The proposed analysis methods and neutron cross-section data should be benchmarked, by the analyst or organization performing the analysis, by comparison with critical experiments.” NUREG/CR-6698 reinforces this statement. As with any guidance, applicants can use alternate methods, provided those methods are technically sound. There is not an accepted standard by which a code-to-code comparison for validating a criticality code may be performed and judged. Therefore, should an application include a validation via code-to-code comparisons, the application should clearly demonstrate the approach and provide sufficient detail for the staff review. The staff should verify that the application includes:

i. A complete validation for the code being used as the standard.

ii. A complete validation for the code being validated using the code-to-code comparison. The application of the statistical analysis of the standard should be fully justified.

iii. Used a sufficient number of comparisons such that they are a statistically viable sample size and able to identify any trends.

5. **Miscellaneous**

a.  **Precedents:** The NRC staff uses precedent to make reviews more efficient, but the agency is not necessarily controlled by precedent when a licensee proposes to change their licensing basis by amendment. In order for a precedent to be applicable to a LAR, they should be substantially similar. Applications have cited precedent, yet when the NRC staff compared the cited precedent to the current LAR, there was insufficient commonality to warrant its use as a precedent.
Therefore, the staff should verify that for cited precedents, the application includes a portion of the analysis that demonstrates the commonality of the precedent to the submittal, with any differences identified and justified with respect to the use of the precedent.

b. References: References can make the NRC reviews more efficient, but they are not without limitations. For example, although this ISG cites numerous references, not one fully addresses every possible aspect of the area for which it is cited. Conclusions in references may only apply to a limited extent. For example, NUREG/CR-6801 observed, “Because the axial blankets have significantly lower enrichment than the central region, the end effect for assemblies with axial blankets is typically very small or negative.” Since “typically” implies “not always,” and “very small” is relative, the NRC reviewer should verify that references cited in the application are used in context and within the bounds and limitations of the references. Any extrapolation outside the context or bounds of the reference should be demonstrated as appropriate.

c. Assumptions: Applications contain numerous assumptions, both explicit and implicit. All assumptions should be justified. Failure to justify assumptions can lead to their inappropriate use. The applicability of an assumption may change with different scenarios in the NCS analysis. Therefore, applications should explicitly identify and justify all assumptions used in their applications.

V CONCLUSION

This draft DSS-ISG provides updated guidance to the NRC staff reviewer that is responsive to the increased complexity of recent spent fuel pool (SFP) license application analyses and operations. The guidance is intended to reiterate existing guidance, clarify ambiguity in existing guidance, and identify lessons learned based on recent submittals.
VI COMPLIANCE WITH THE BACKFIT RULE

Issuance of this DSS ISG does not constitute a backfit as defined in Title 10 of the Code of Federal Regulations (10 CFR) 50.109(a)(1) (Reference 14), and the NRC staff did not prepare a backfit analysis for issuing this DSS ISG. Consistent with the description in Reference 15 this draft DSS-ISG-2010-01 is a “forward fit,” that is, the guidance will be applied only to: (i) future applications; and (ii) future licensee applications for license amendments and requests for exemptions from compliance with applicable requirements. In these circumstances, the NRC does not consider the issuance of “forward fit” interpretive guidance to constitute “backfitting.”

VII REFERENCES


5. Title 10 of the Code of Federal Regulations (10 CFR) 50.68, Criticality Accident Requirements,

6. Title 10 of the Code of Federal Regulations (10 CFR) 70.24 Criticality Accident Requirements,


15. U.S. Nuclear Regulatory Commission letter from Stephen G. Burns, General Counsel, to Ellen C. Ginsberg, Vice President, General Counsel and Secretary, Nuclear Energy Institute, July 14, 2010. (ADAMS Accession No. ML101960180)