Shielding Topics

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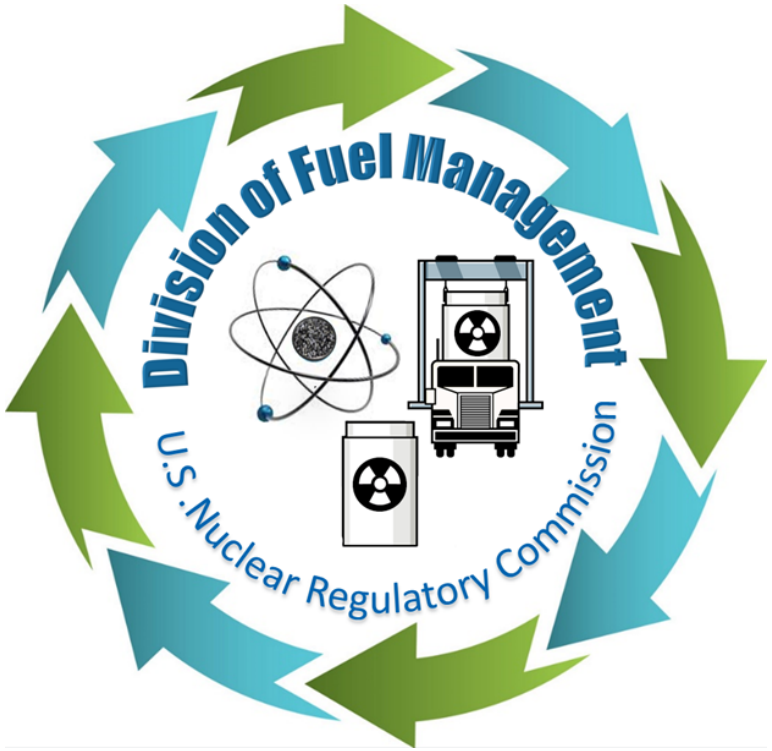
NRC, Division of Fuel Management

June 23 Public Meeting with NEI



History

- Previous meetings:
  - **March 26:** NEI presented its concepts for using more realistic rather than conservative depletion parameters in source term calculations (NEI Recommendation III-1, III-2)
  - **April 15:** NEI presented its proposal with rationales to revise the NRC's standard review plan to use a typical cask with representative content when performing demonstration calculations for the design given the fact that site specific shielding evaluations are required as part of 72.212 evaluations (NEI Recommendation V-1 and V-2)

RIRP I-20-01

- Staff developed a Regulatory Issue Resolution Plan to address NEI Recommendation III-3
- Milestones:

ACTION	RESPONSIBLE PARTY	DUE DATE
1 . Develop alternative licensing approaches, with pros and cons of each	NRC	June 2020
2. Conduct a public workshop to discuss alternatives and receive feedback from stakeholders	All	July 2020
3. Select optimal alternative	NRC	September 2020
4. Develop draft review criteria considering flexibility within shielding modeling (including the source terms and shielding design) and conservatisms within analysis	NRC	December 2020

NEI Recommendations

- Rather than address recommendations individually, staff has been working on a proposal that may address multiple recommendations
  - III-3 – NRC should perform a less detailed review when conservative modeling is used by GL
  - V-1 – NRC should revise guidance to use representative rather than bounding dose rates
  - V-2 – NRC should revise shielding review guidance to use industry experience showing conservatism within analyses
  - VI-2 – NRC and industry should align fuel qualification with practice in current operating reactors
  - VI-1 – CoC holders should amend CoCs to use the graded approach method to reduce TS volume

Regulations

- Spent Fuel Dry Storage is regulated under 10 CFR Part 72
  - A two phase regulatory framework: (1) Design certification and (2) Use of the design
  - Subpart K establishes the general license
  - Subpart L has regulations for the approval of *cask design* that can be used under the general license granted in Subpart K
- Although there are other applicable regulations pertaining to shielding and radiation protection, three key ones are related to shielding design discussed today:
  - **10 CFR 72.236(a):** Specifications must be provided for spent fuel to be stored …
  - **10 CFR 72.236(b):** Design bases and design criteria must be provided for structures, systems, and components important to safety
  - **10 CFR 72.236(d):** Radiation shielding and confinement features must be provided sufficient to meet the requirements in §§ 72.104 and 72.106.

Status Quo

- Currently NRC staff approves contents with respect to shielding design via *values* of burnup, enrichment and cooling time (BECT) to satisfy portions of 72.236(a) that normally appear in the technical specifications
  - Can be in the form of Fuel qualification table (FQT) and/or Correlations
- These BECT values are used within shielding demonstration calculations to satisfy 72.236(d) that the *shielding features* are sufficient to meet the requirements in 10 CFR 72.104 and 72.106
  - Within these requirements are an annual dose limit for controlled area boundary under normal operating conditions and anticipated occurrences and a dose limit under design basis accidents

Need for Improvement

- CoC applicants have refined their allowable contents for a dry storage system to maximize capacity while maintaining performance of the system (i.e. limiting dose).
- Some systems allow thousands of possible permutations of fuel parameters that can be used to create a single loading pattern for one cask
  - Each one of these thousands of possible combinations is designed to have roughly similar performance in terms of dose rate.
- The staff has not required specifying fuel in this way, staff reviews what is submitted. Reviewing and approving systems with complex fuel specifications is difficult and time consuming
- To alleviate this issue, the staff is seeking a more efficient way that the systems can demonstrate compliance with regulations while maintaining the same level of safety
- Staff is interested in reducing its review time so that it can focus its reviews in the most safety significant areas

NRC Proposal

- NRC is considering acceptance of a *Method of Evaluation* approach to determine specifications for burnup, enrichment, and cooling time to meet the requirements of 10 CFR 72.236(a)
- For example, instead of actual *values* of burnup, enrichment and cooling time, technical specifications could state that a specific NRC approved method will be used to derive and limit these parameters
  - NRC would review and approve the specific method that is referenced in the technical specifications
- This is similar to technical specifications for operating reactors that use a core operating limits reports (COLR)
  - limits are not set in the technical specifications, but specific NRC approved methods are used to derive these limits
- This approach builds on lessons learned during the Graded Approach (72-1004 Amd. 16) review

NRC Proposal (Continued)

- The MOE approach would be able to use a *representative* source to perform the demonstration that the shielding is sufficient per the requirements in 72.236(d)
- Applicants would need to justify that the source is adequately "representative" of what could be allowed by the approved method of evaluation for the allowable contents
- NRC staff is considering potential ways this can be established that satisfies 72.236(b) by providing design criteria and design bases as required by this regulation
- In the July workshop, NRC would like to discuss its ideas and get feedback

Additional Shielding Work

- NRC would like to update its shielding review guidance
  - Specific to support documentation such as NUREG/CR-6802
  - NRC would like to include more modern fuel and insert designs and more modern and realistic depletion parameters
  - NRC would like to prepare for the possibility of higher enrichment/higher burnup fuel
  - Generically address areas where there are frequent RAIs and applicants are performing evaluations on a case-by-case basis
    - Examples: axial burnup profile, axial blankets, use of stainless steel replacement rods, assumptions on fuel inserts, etc

Background

- During 3/26 meeting, NEI expressed interest in the NRC accepting more realistic rather than conservative values within evaluations
  - NEI stated it will provide a demonstration of the safety margins within its source term calculations using data from ORNL NUREGs
- NRC has asked ORNL to identify more modern/realistic data for updating NUREGs
- As a result of ORNL's search, they were not able to find more modern data than what has been used in previous NUREGs

Available Data

ORNL has identified the following sources of available data to perform shielding evaluations:

1. *Characteristics of Spent Fuel, High Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation*, DOE/RW-0184, U.S. Department of Energy, December 1987
2. *Characteristics of Spent Fuel, High Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation*, DOE/RW-0184, R1, U.S. Department of Energy, December 1992.
3. *Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code*, ORNL/TM-6051, ORNL, 1978
4. *Physical and Decay Characteristics of Commercial LWR Spent Fuel*, ORNL/TM-9591, 1986
5. A. Luksic, *Spent Fuel Assembly Hardware: Characterization and 10CFR 61 Classification for Waste Disposal,* PNL-6906-vol. 1, Pacific Northwest Laboratory, June 1989
6. R. J. Cacciapouti and S. Van Volkinburg, *Axial Burnup Profile Database for Pressurized Water Reactors,* YAEC-1937, Yankee Atomic Electric Company (May 1997).
7. L. B. Wimmer and C. W. Mays, *BWR Axial Burnup Profile Evaluation*, 32-5045751-00, Framatome ANP, Lynchburg, VA (2004).

Data Uses

- Physical characteristics of limiting insert, control rod assembly/control blade, neutron source assembly would be derived from 1992 data (Ref: 1, 2, 4)
- Assumed Co-59 impurity for steel/Inconel will be mainly from References 1,2, 3, and 4
- Flux scaling factors for fuel assembly hardware will be from Reference 5
- Physical characteristics of limiting fuel characteristics and non-fuel hardware will be derived from 1992 data (Ref: 1,2)
- Limiting burnup profile and operating characteristics will be based on CRCs, Reference 6 and 7 and conservative assumptions to be determined
- Reactor operating parameters will be based off of NUREG/CR-6802, conservative assumptions and other sources to be determined

Opportunities for Input

- NRC recognizes that some of this data is old and may be overly conservative
- In performing sensitivity studies and updating NUREGs, ORNL would have to use conservative assumptions where data is unknown
- NEI has an opportunity to have input to this process by proposing sources of data that contain more realistic values
- NEI also has an opportunity to provide feedback by identifying areas where there are excessive conservatisms so that NRC staff can determine if updating guidance is of value