UNITED STATES
NUCLEAR REGULATORY COMMISSION
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**ARC CLEAN TECHNOLOGY – U.S. NUCLEAR REGULATORY COMMISSION STAFF FEEDBACK REGARDING WHITE PAPER: "SPENT FUEL STORAGE INSIDE THE REACTOR VESSEL," REVISION 0.0 (EPID NO. L-2023-LRO-0054)**

**SPONSOR INFORMATION**

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| **Sponsor:** | ARC Clean Technology (ARC) |
| **Sponsor Address:** | 901 K Street, NW<br>Washington, DC 20001 |
| **Project No.:** | 99902103 |
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**Submittal Agencywide Documents Access and Management System (ADAMS) Accession No.:** ML23178A109

**Purpose of the White Paper:** The WP describes ARC's approach for storing spent fuel inside the reactor vessel for the ARC-100 reactor design.

**Action Request:** ARC requested that the NRC staff review the WP and provide feedback and observations regarding topics for which additional discussion or consideration may be beneficial.

**FEEDBACK AND OBSERVATIONS**

The NRC staff has reviewed the WP and provided feedback and observations below. The NRC staff's feedback and observations are organized into six topical areas. The NRC staff's feedback and observations do not constitute final agency positions and are not intended to be comprehensive. Lack of feedback or observations regarding a certain aspect of the WP should not be interpreted as NRC staff agreement with ARC's position.

## I. <u>Terminology</u>

1. The term "in-vessel storage" is used in the WP to refer to the location in which irradiated fuel will be stored. Future licensing submittals addressing this topic should clarify whether the "in-vessel" term refers to a location adjacent to the reactor vessel or a compartment inside the reactor vessel.

2. Future licensing submittals addressing this topic should clarify the meaning of the phrase "ex-vessel storage" and the associated equipment.

3. In Section 1.1.1.C, “Operations,” the WP refers to, “…a transportation cask that interfaces with the on-site dry storage facility. Only the transportation cask moves between the reactor building and the on-site dry storage facility. The [fuel unloading machine (FUM)] is only used within the reactor building.” Future licensing submittals addressing this topic should clarify whether the term “transportation cask” is referring to a “transfer cask.” A transfer cask system may be used for moving spent fuel from the reactor building to a dry storage facility (i.e., an independent spent fuel storage installation).

## II. Applicability of Regulations

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 72.3, “Definitions,” defines the term *Spent nuclear fuel or Spent fuel* as:

> fuel that has been withdrawn from a nuclear reactor [emphasis added] following irradiation, has undergone at least one year's decay [emphasis added] since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.

If the irradiated fuel has not decayed for one year or more and the fuel remains in the reactor vessel, then the regulations in 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste,” may not apply. For these reasons, the storage of irradiated fuel located in the reactor vessel may need to be licensed under the NRC regulations applicable to this type of nuclear reactor or exemptions to 10 CFR Part 71, “Packaging and Transportation Radioactive Material,” need to be requested.

## III. Fuel Performance

During the review of future licensing submittals (e.g., topical reports, applications), the NRC staff would likely seek to review specific information regarding the analyses performed to support the statements made in Section 2.0, “Fuel Performance.” This information would assist the NRC staff in making its safety determination regarding the storage of the spent fuel (or the effects on the core itself) during normal operation and postulated events. Specific information may include:

1. Technical or calculational details of the methods and codes used. Details regarding the methods' verification, validation, and applicability to the analyses being performed.

2. The technical bases for the equations used, input assumptions, results obtained, uncertainties for the analyses performed, and an explanation of relevant timeframes considered in the analyses.

3. Discussion that justifies or demonstrates how the calculations and analyses performed bound (or adequately represent) the spent fuel assemblies. The NRC staff expects that this discussion would include consideration of different locations, levels of burnup, physical degradation, decay heat generation, and any other phenomena that could meaningfully influence the performance of the spent fuel and its capability to retain radionuclides within the appropriate limit(s).

## IV. <u>Criticality Safety and Shielding Evaluations</u>

1. An applicant should ensure that sub-critical fission rates in stored fuel during subsequent operation of a fresh core can be adequately predicted, measured, or otherwise monitored to ensure that the fuel's lifetime burnup limits are not exceeded during the period of in-vessel storage.

2. Regarding WP Section 2.2, "Criticality and Shielding Considerations," the NRC staff acknowledges and generally supports the recommendation for performing a more detailed multigroup criticality analysis. However, the NRC staff also notes that, in a potential future licensing submittal, a simplified bounding calculation with significant margin could be found acceptable provided that an applicant adequately demonstrates that:

   a. the calculation (or methodology) is bounding and conservative in nature, and

   b. considerable margin to criticality (such as the prediction provided in WP Section 2.2 and Annex 1, "Approximate Criticality Calculation for Proposed Spent Fuel in Vessel Storage Configuration") exists after appropriately accounting for uncertainties.

## V. Fuel Handling Accidents

1. Regarding WP Section 2.7, "Fuel Handling Accidents," future licensing submittals addressing this topic should provide more details about the proposed evaluation methodologies and assessment criteria. For example, whether evaluations were performed for cask or vessel storage drops.

## VI. <u>Dry Storage Facility and Transportation of Spent Fuel</u>

1. General Observations

   a. <u>Section 1.1.1.C</u>: The terms "transfer" and "transportation" are not synonyms or interchangeable. These terms have specific meanings in different regulations in 10 CFR Part 71 and 10 CFR Part 72.

   b. <u>Appendix B, "ARC-100 Core Reloading Evaluation," Second Table (Untitled) - Fuel Handling Step Table Item 18:</u> Additional information regarding this activity would likely be needed to support a future licensing submittal for the NRC staff to assess whether the applicable requirements of 10 CFR Part 72 (i.e., storage) are met. A determination of calculated (or measured) decay heat on a per-assembly basis and subsequent component temperatures and limits will need to be provided as part of a future licensing submittal related to this topic.

2. Criticality and Shielding Safety

   a. <u>Annex 1:</u> The effective multiplication factor ($k_{eff}$) for the maximum number of assemblies loaded in the most reactive configuration will be necessary to provide reasonable assurance of subcriticality for a dry storage system and a transportation package. The configuration for the criticality calculations with multiple assemblies separated by sodium makes sense for in-vessel storage. However, an applicant for a storage cask or transportation package design should be prepared to conduct additional tests to provide assurance of moderator exclusion or demonstrate subcriticality in the event of water in-leakage.

b. Section 2.0, "Fuel Performance": Irradiation parameters impact source term and fuel reactivity. Typically, bounding in-core conditions are chosen and an applicant may have to account for the potential of additional exposure from in-vessel storage or justify that it has a negligible impact on storage and transportation activities.

3. Aging Management

The NRC staff observed the following statement from Section 1.1.1.A, "Safety" of the WP:

> Appendix B provides a time and motion study describing in more detail the operations that take place when the spent fuel is stored in-vessel. In the 2nd refueling, the FUM operation takes place at a time when the very low decay heat of the fuel (20-year decay) minimizes the likelihood of an incident resulting in fuel damage and release of radioactivity in case the fuel assembly becomes stuck during the transfer operation that moves the cask to the on-site dry storage facility. The fuel has decayed long enough to allow immediate placement in the onsite dry cask storage facility. In fact, disposal of a spent core in the onsite dry cask storage facility is possible as early as 4 years after shutdown.

This statement may be accurate when considering only thermal effects issues, but it does not address long-term aging mechanisms present in used fuel which alter the initial conditions during transfer and long-term storage. Section 2.4, "Effects on Fuel Life Limiting Phenomena," of the WP addresses this to a degree, but the cursory evaluation likely will have to be substantially expanded to address aging management issues.

4. Materials Evaluation

a. The following sections discuss how the FUM transfers spent fuel from the reactor head location to the wash area and ultimately to the dry cask (for the in-vessel storage approach) or from the spent fuel storage vessel to the dry cask (for the ex-vessel storage approach):

- Section 1.1.1.C,
- Appendix B (first table, untitled), Fuel Handling Steps 2G and 2H, and
- Appendix B (second table, untitled), Fuel Handling Comment 18.

(1) The process of cleaning all sodium from the fuel should be discussed as part of a future licensing submittal that addresses this topic, as well as how the fuel is adequately dried and inerted throughout the movement to dry storage. The purpose of this evolution is to prevent degradation of either the fuel cladding or a challenge to the performance of the spent fuel dry storage system that would be caused by a reaction between residual sodium and water or water vapor, which could form hydrogen and sodium hydroxide. Preventing the formation of

hydrogen from this reaction is also necessary to prevent other flammable or explosive reactions from occurring.

(2) Guidance available regarding related safety considerations for dry storage includes the following:

i. NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities – Final Report," (ML20121A190)

A. Section 8.5.13.1, "Flammable and Explosive Reaction"

B. Section 8.5.13.2, "Corrosion," provides guidance on corrosive reactions

C. Section 8.5.15.2.3, "Effective Cladding Thickness," provides guidance on drying adequacy

ii. NRC paper "Storage Experience with Spent (Irradiated) Advanced Reactor Fuel Types," (ML20211L885) contains information on sodium reactions with water in regard to dry storage of spent fuel.

The NRC staff notes this information is relevant to the regulatory requirements in 10 CFR 72.120, "General considerations," paragraph (d), applicable to specific licensees; 10 CFR 72.122, "Overall requirements," paragraph (h)(1), applicable to both specific and general licensees; and 10 CFR 72.236, "Specific requirements for spent fuel storage cask approval and fabrication," paragraph (h) applicable to storage system certificate of compliance holders.

b. Fuel cladding feedback and observations:

(1) While the data within WP reference [3] provides potential insight into the thickness of cladding wastage following irradiation, it is incumbent upon the applicant to verify that statistically significant experimental data for the characteristics of irradiated HT9 material was gathered under a quality assurance program that meets the requirements of 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." An applicant's use of data directly from published peer-reviewed journals, academic texts, and internal memos does not ensure that the data was gathered in accordance with an acceptable quality assurance program. Although not specific to spent fuel, NUREG-2246, "Fuel Qualification for Advanced Reactors," provides general guidance to applicants on fuel qualification for advanced reactors (ML22063A131).

(2) WP Section 2.4 describes that "…significant additional damage…would occur only if the condition were to last for a period of 90 days, during which time the condition can be rectified." Additional information would need to be provided as part of a future licensing submittal to explain whether the cladding will maintain its integrity following the 90-day unprotected reactivity excursion.

(3) The NRC staff encourages applicants to supply unambiguously legible material for review. The NRC staff found parts of the following materials illegible:

- Table 1, “Summary of [Fuel-Cladding Chemical Interaction] data from X447 and MFF3, MFF5,” is generally (but not entirely) legible (lotti, 2021)

- Inagaki, K. et al., "Investigation on Fuel-Cladding Chemical Interaction in Metal Fuel for FBR," Transactions of the Atomic Energy Society of Japan, Vol. 12, No. 2, p. 149-157 (2013)

- Lotti, R. C., "Calculation to compare Lanthanides generated layer thickness in experiments carried out at the EBR II and FFTF, with predictions of reference 1 and consequences to the AR 100 long term possible weakening of the cladding," July 2-5, 2021

## VII. References

The NRC staff would expect complete bibliographic information for all the references described in a licensing submittal related to this topic. Applicants should also perform a thorough quality check on material submitted to the NRC staff and ensure all figures and tables have titles and are legible. Broken cross-references were found throughout the WP, as indicated by “Error! Reference source not found.”

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