January 8, 2009

Mr. J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: WESTINGHOUSE LTR-NRC-08-44

Dear Mr. Gresham:

Thank you for your letter from September 15, 2008, describing Westinghouse efforts to develop and test an advanced light-water reactor (LWR) fuel system consisting of ceramic cladding and an alternative fuel form. Your briefings to the U.S. Nuclear Regulatory Commission (NRC) technical staff in March 2007 and August 2008 were interesting, and we note progress in your efforts to irradiate fueled test specimens in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory. We also are aware of related efforts on similar fuel designs as reported in the open literature [Reference 1].

Of particular interest to us was your estimate of schedule for further development, testing, and licensing requests. Your provision for separate tracks on Lead Test Rods and Lead Test Assemblies is prudent.

Because the goal of your program is the development of a nonconventional fuel design, it is understandable that emphasis is placed on developing operational experience and improving operational performance. However, it should be noted that regulatory emphasis is placed not only on behavior under operating conditions but also on behavior under transient and accident conditions. In addition to defining specified acceptable fuel design limits (in accordance with Appendix A of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50), General Design Criterion 10), testing under design basis conditions may be necessary to support a licensing application. At a minimum, these conditions should consider limiting design basis accidents or events (e.g., rod ejection) that result in limiting conditions for operation.

Your intent to introduce this new fuel design into current LWRs raises an issue over changes to existing plant safety analysis (provided under 10 CFR Part 50.34) and current safety margins. Changes could be expected in:

- Fuel Failure Criteria
- Onset of Gap Release of Fission Products
- Behavior of Fission Product Release (e.g., chemical forms and release kinetics)
- Thermal-Hydraulic Response of the Reactor Coolant Systems Under Design-Basis and Severe Accident Scenarios
- Core Melt Progression and Relocation
- Impact of Core Melt on Containment Integrity (e.g., core-concrete interaction)

These changes must be understood and their effect on existing safety analyses, if any, must be appropriately evaluated. For example, the proposed fuel design may behave differently than conventional fuel under design basis and severe accident conditions, affecting hydrogen loading and subsequent containment performance. In addition, changes in the source term expected from this fuel type could affect equipment qualification, and dose calculations for control room habitability, siting, and accident evaluation.

Some of our regulations (e.g., the General Design Criteria in 10 CFR Part 50, Appendix A) are imminently applicable to your new fuel design. Unfortunately, others apply only to more traditional LWR fuel designs.

An additional issue is the following item in your letter:

> *Provide irradiated clad test specimens to be used in the ANL LOCA clad testing program to determine the operational limits for ceramic clad.*

The NRC loss-of-coolant accident (LOCA) testing program at Argonne National Laboratory (ANL) is focused on the embrittlement criteria found in 10 CFR 50.46(b). In its present form, this regulation applies specifically to zirconium cladding (in fact, two specific zirconium-based alloys) and does not apply to ceramic cladding. Similarly, 10 CFR 50.46(b) applies specifically to uranium oxide fuel and no other fuel forms. As a consequence, one should not expect that participation in the ANL LOCA testing program can provide "the operational limits for ceramic cladding." Demonstrating that ceramic cladding is highly stable in high-temperature steam does not provide an operational limit. To satisfy the requirements of General Design Criterion 35, other criteria must be identified to ensure coolability of the fuel during the design basis LOCA.

Other parallels exist in the regulatory process. For example, General Design Criterion 10 assures that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). Guidelines for SAFDLs can be found in Section 4.2 (Fuel System Design) of the Standard Review Plan (NUREG-0800) and some Regulatory Guides. However, these guidelines generally apply to metallic cladding.

NRC has recognized that its current regulations primarily apply to LWRs with zirconium-based cladding and oxide fuel, and alternative criteria would need to be developed for new fuel designs. NRC considered the phenomena of interest for carbon-based fuel designs in the Next Generation Nuclear Plant Phenomena Identification and Ranking Tables [Reference 2]. The staff believes the discussion in Volumes 3-5 of NUREG/CR-6944, which deals with accident scenarios and carbon-based fuel designs, may assist you in proposing criteria for the Westinghouse advanced fuel design.

The NRC staff suggests replacing Element #4 in your schedule with the following:

> *Consider the issues raised in NUREG/CR-6944 for relevance to your proposed fuel design.*

Propose alternative acceptance criteria for current regulatory requirements or guidelines that apply specifically to metallic cladding or oxide fuel.

In some cases, Westinghouse may wish to adopt current limits (e.g., 2,200 ^{o}F in 10 CFR 50.46) and then show rod-like geometry is always maintained. In other cases, higher (or lower) limits may be proposed, but the General Design Criteria must continue to be met.

Special consideration should be given to:

- Integrity of ceramic cladding during normal operation (and AOOs).
- Thermal-hydraulic behavior of ceramic cladding.
- Mechanical behavior of ceramic cladding/alternative fuel form composite.
- Swelling and fission gas release from alternative fuel forms at high burnup.
- Water solubility of alternative fuel forms.
- Loss of integrity before loss of geometry of ceramic cladding.
- Verified and validated computer models, along with an adequate supporting experimental database, capable of accurately predicting the fuel pellet and cladding response to both normal and upset conditions.

We hope these considerations provide some positive feedback in support of your efforts, and we look forward to further briefings on your progress.

Sincerely,

/RA/

Brian W. Sheron, Director
Office of Nuclear Regulatory Research

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References

1. D. Carpenter, G. Kohse, and M. Kazimi, “Modeling of Silicon Carbide Duplex Cladding Designs for High Burnup Light Water Reactor Fuel,” Proceedings of ICAPP 2007, Nice, France, May 2007.

2. S. J. Ball and S. E. Fischer (ORNL), “Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs),” NUREG/CR-6944, Volumes 1-6, March 2008.