UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
OFFICE OF NEW REACTORS
WASHINGTON, DC 20555-0001

October 8, 2009

NRC INFORMATION NOTICE 2009-23: NUCLEAR FUEL THERMAL CONDUCTIVITY DEGRADATION

ADDRESSEES

All holders of operating licenses and construction permits for nuclear power reactors under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel. All current and potential applicants for an early site permit, combined license, or standard design certification for a nuclear power plant under the provisions of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." All holders of, and applicants for, a certificate of compliance for a spent nuclear fuel transportation package under the provisions of 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." All holders of a certificate of compliance for a spent fuel storage cask and all holders of a license for an independent spent fuel storage installation under the provisions of 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."

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to notify addressees of information related to the impact of irradiation on fuel thermal conductivity. The NRC expects the recipients to review the information within this IN for applicability to their facilities and consider actions, as appropriate, for their facility. However, suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

It is well understood that irradiation damage and the progressive buildup of fission products in fuel pellets result in reduced thermal conductivity of the pellets. However, thermal performance codes approved by NRC before 1999 did not include this reduction in thermal conductivity with increasing irradiation because earlier test data were inconclusive as to the significance of the effect.

Measurements collected from an instrumented assembly at the Halden ultra-high-burnup experiment during the 1990s have indicated steady degradation in the thermal conductivity of uranium fuel pellets with increasing exposure. These data indicate a degradation of approximately 5 to 7 percent for every 10 gigawatt-days per metric tonne of exposure.

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On the basis of these experimental data, the NRC updated its confirmatory fuel thermal-mechanical performance tool, FRAPCON, to include a new model for predicting fuel thermal conductivity as a function of exposure. NUREG/CR-6534, Volume 1, "FRAPCON- 3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Application," issued October 1997, discusses these bases and model updates.

Beginning in 1999, several reactor fuel vendors submitted improved fuel thermal models to the NRC for review and approval. These new models incorporate updates to the fuel thermal conductivity models that account for degradation caused by irradiation. The improved vendor models generally considered experimental qualification data that were substantially similar to the data considered in NUREG/CR-6534. However, the staff is aware that models that do not account for the effect of degradation are still used to perform safety analyses.

BACKGROUND

Licensees employ a series of computer codes to analyze plant behavior in the safety analyses they perform to demonstrate compliance with the Commission's regulations. The computational approach models various physical processes to predict transient and accident events. These models simulate reactor conditions for postulated events and compare predicted plant performance to applicable regulatory criteria.

The simulation of the fuel element is an integral part of the safety analysis. Within the analysis, the fuel pellet thermal conductivity model determines the rate at which heat is transferred from the fuel pellet, first to the gas gap, then to the fuel cladding, and subsequently to the coolant. A lower fuel pellet conductivity results in higher fuel temperatures at a given linear heat-generation rate. Therefore, the analytical prediction of the fuel thermal conductivity will affect the results of several types of safety analyses. Any codes used for safety analyses that incorporate data starting at the fuel rod level and generated by the pre-1999 models may mischaracterize the expected plant performance.

DISCUSSION

General Design Criterion (GDC) 10, "Reactor Design," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, establishes that licensees should not exceed specified acceptable fuel design limits (SAFDLs) during any condition of normal operation, including the effects of anticipated operational occurrences, to ensure that the fuel is not damaged. Also, the general requirements to maintain control rod insertability and core coolability appear in the GDC (e.g., GDC 27, "Combined Reactivity Control Systems Capability," and 35, "Emergency Core Cooling"). In particular, 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," provides the specific coolability requirements for a loss-of-coolant accident. In addition, 10 CFR 50.46(a)(3) specifies requirements for evaluating and reporting each change to, or error discovered in, an acceptable evaluation model.

Technical specifications require licensees to submit a report on core operating limits that incorporates the revised cycle-specific parameters resulting from the new core configuration implemented during the refueling outage. Technical specifications require that the analytical methods used to determine the core operating limits be those previously reviewed and approved by the NRC. Licensees rely on computer codes for fuel performance calculations and to perform safety analyses. Within the scope of reload licensing evaluations, they use these computer codes to establish cycle operating limits to ensure that all applicable requirements (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, emergency core cooling system limits, and nuclear design limits) are met.

If the pre-1999 methods misrepresent fuel thermal conductivity, calculated margins to SAFDLs and other limits may be less conservative than previously understood.

GENERIC APPLICABILITY

Safety analyses performed for reactors using pre-1999 methods may be less conservative than previously understood.

Lower fuel pellet conductivity does not appear to significantly influence spent nuclear fuel cladding temperatures that are typically estimated for aged spent nuclear fuel during dry cask storage and transportation operations. However, an increase in estimated cladding temperatures could challenge small thermal margins in the design bases for certified or licensed spent nuclear fuel storage casks and certified spent nuclear fuel transportation packages.

CONTACT

This IN requires no specific action or written response. Licensees should refer any questions about this notice to the technical contacts listed below or to the appropriate project manager in the Office of Nuclear Reactor Regulation. Combined license applicants should refer any questions about this notice to the technical contact listed below or to the appropriate project manager in the Office of New Reactors.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

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