ATTACHMENT 7B FUEL SELECTION

In accordance with 10 CFR 71.33(b)(3), an application for a transportation package must include a description of the chemical and physical forms of the spent nuclear fuel (SNF) contents. Further, as required by 10 CFR 71.55(d)(2) and 10 CFR 71.87(a), the geometric form of the package contents must not be substantially altered during normal conditions of transport and the package is to be proper for the contents to be shipped, respectively. Therefore, for undamaged and intact assemblies, the fuel cladding serves a design function in transportation packages for ensuring that the SNF configuration remains within the bounds of the safety analyses in the application. This assurance is used when developing instructions for safely opening the transportation package (as stated in 10 CFR 71.89, "Operating Instructions"), as any potential fuel reconfiguration during transport should be accounted for in these procedures. If the fuel is classified as damaged, a separate canister (e.g., a can for damaged fuel) that confines the assembly contents to a known volume may be used to ensure the safety analyses in the application remain bounding.

The certificate of compliance (CoC) of the transportation package generally defines the allowable cladding condition for the SNF contents, and the nomenclature has historically varied from design to design. For example, the terms "intact" and "undamaged" have both been used to describe cladding without any known gross cladding breaches. New applications should adhere to the nomenclature of this standard review plan whenever practicable. Users of transportation packages are required to comply with the CoC by selecting and loading the appropriate fuel and must maintain records that reasonably demonstrate that loaded fuel was adequately selected in accordance with their approved procedures and quality assurance (QA) program.

Users may consider several methods, either singularly or in combination, to demonstrate that the fuel cladding does not contain gross breaches.

7B.1 <u>Reactor Operating Records</u>

The staff considers that adequate reactor operating records that identify only gaseous or volatile decay products (no heavy metals) in the reactor coolant system are acceptable evidence that cladding breaches are no larger than a pinhole leak or hairline crack. If heavy-metal isotopes were detected in the coolant system during reactor operation, additional fuel qualification testing is generally needed to identify grossly breached assemblies in the core.

Users should assess whether any missing records from early reactor operation, such as those lost from changes in plant ownership, may impact conclusions made about fuel discharged from a given cycle. The users should determine whether additional fuel qualification is necessary to provide reasonable assurance that the fuel to be loaded in the transportation package was properly classified.

7B.2 <u>Visual Inspection</u>

Visual examination of selected fuel has a two-fold purpose: (i) to identify any mechanical damage to the assembly that may preclude its ability of being retrieved, and (ii) to assess the extent and size of any cladding failures. The extent of visual inspection is generally limited in assessing flaws behind the spacer grids (e.g., pellet-clad interaction flaws, debris fret) and in rods in the inner matrix. Therefore, most users utilize a tape-recorded visual inspection of the exterior of the fuel assembly only as a supplement to other fuel qualification test data [e.g., sipping, ultrasonic testing (UT)]. In addition, accessibility in boiling-water reactor (BWR) assemblies may also be limited by the flow channel. Because of these limitations, unless a user can reasonably demonstrate sufficient resolution and inspection coverage, visual inspection may not provide, on its own, reasonable assurance that the fuel cladding does not contain gross cladding breaches.

7B.3 Fuel Qualification Testing

7B.3.1 Sipping

Sipping techniques are widely used to identify failed fuel assemblies by detecting radioactive fission gases (e.g., krypton-85, xenon-133) released through cladding breaches. The techniques are not considered adequate for breach sizing; therefore, users generally conservatively classify fuel with detected fission gases as damaged.

Mast sipping is generally performed during refueling operations, as the first lift from the core generally yields the highest release of fission gases (from the decreasing water head pressure). Three primary techniques are used for sipping, depending on the reactor type: (i) in-mast sipping for pressurized-water reactors (PWRs), (ii) telescope sipping (for PWRs or BWRs), and (iii) mast sipping (for PWRs). The operations vary. For example, in-mast sipping generally employs air injection at the bottom of the mast to help entrain released fission gases; telescope sipping generally includes processing a gas sample from a liquid extraction; and mast-sipping allows for sampling at different locations. The staff considers mast sipping records to be adequate for fuel selection if testing is performed at the time of discharge under conditions not known to result in nonconservative measurements. For example, inner core assemblies from cycles with significant grid-to-rod fretting may increase the background counts and mask small-release leakers, particularly for sipping methods that do not use gas entrainment. Therefore, when determining whether the fuel is intact or undamaged, the user should review mast sipping data considering the limitations of the respective technique.

The staff does not expect any operable degradation mechanisms to result in gross cladding breaches during wet storage. Therefore, telescope sipping has historically been used for fuel qualification of wet stored fuel (e.g. during spent fuel pool transfers). However, the use of telescope sipping for SNF that has been in wet storage for a significant period should consider the sensitivity of the technique relative to the fuel’s decreasing fission gas inventory.

International Atomic Energy Agency Nuclear Energy Series No. NF-T-3.6, “Management of Damaged Spent Nuclear Fuel,” issued June 2009, recommends that xenon-133 measurements be taken up to 2 months after discharge and krypton-85 measurements be taken up to 10 years after discharge.

The industry generally regards vacuum can sipping as one of the most sensitive fuel qualification techniques currently available, particularly for low-power and low-fission-yield assemblies. This technique involves individually placing each assembly inside an isolation chamber (sealed can) and drawing a negative pressure to drive noble fission gas releases (if the cladding is breached), which are collected at the top of the can. The staff considers this technique acceptable for all fuel.

7B.3.2 Ultrasonic Testing

In-bundle UT is generally performed by placing multiple UT wands at a preestablished axial elevation on the probed assembly. PWR assemblies do not require dismantling for accessibility;

however, BWR assemblies generally require de-channeling. UT relies on the measurement of the reflected amplitude of a shear wave signal as it transverses the cladding tube. Water ingress to the rod leads to UT signal attenuation (amplitude reduction) and identification of a cladding breach.

Users historically have relied on UT data for fuel classification and selection. However, users should consider potential technique limitations during their review of UT data. More specifically, the user's review should consider (i) whether the lack of water inside the fuel rod at the elevation of the UT inspection can reasonably ensure no water ingress at other axial elevations (particularly for high-burnup fuel, where the interspace between the cladding and the fuel pellet may be closed); (ii) the effects of pellet-to-clad interactions, which may produce multiple echo signals that are difficult to assess; and (iii) any potential misalignment of the transducers from the presence of CRUD or oxide flaking, or any fuel rod bowing or geometry changes from irradiation (e.g., bowing caused by larger-diameter guide tubes). These limitations may result in a user not adequately classifying an assembly, potentially resulting in fission gas releases during drying operations.

In the past, 10 CFR Part 72 licensees have revised operating procedures to limit or avoid the use of UT inspections for fuel classification. For example, a secondary review of UT data from assemblies loaded during a late 2004 campaign at Arkansas Nuclear One resulted in the conservative reclassification of five assemblies loaded in four MPCs as damaged fuel (NRC 2005). The licensee concluded that UT data could not reasonably be used to size the identified failures. Therefore, the licensee submitted an exemption request from the requirements of 10 CFR 72.212(a)(2) and 10 CFR 72.214, which included revised safety analyses assuming up to two damaged fuel pins, each in a separate fuel assembly. In a separate event in 2014, Arkansas Nuclear One conservatively reclassified an assembly as damaged following a noble fission gas release (krypton-85) during forced helium dehydration of a loaded multipurpose cask (NRC 2016; Entergy 2014). The licensee cited the prevalence of grid-to-rod fretting in the operating cycles for the subject assemblies and the lower reliability of UT relative to other fuel qualification test methods as the most likely cause of the event. As a corrective action, the licensee revised operating procedures to avoid the use of UT for future fuel classification. The licensee for the Calvert Cliffs Nuclear Power Plant has also chosen to rely on vacuum can sipping for fuel classification activities in the interest of potentially identifying any legacy fuel that may be vulnerable to releases.

7B.4 Noble Gas Releases During Loading Operations of Transportation Packages

Noble fission gas releases may occur during SNF loading operations of transportation packages. The staff expects users to document the occurrence of these releases and take actions consistent with their approved procedures and QA program. These actions may include a review of fuel-selection records, the performance of a root-cause or apparent-cause analysis, and a review of industrywide operating experience pertaining to these releases to determine additional followup actions. Users should ensure the contents loaded into the transportation package meet the applicable CoC conditions pertaining to the fuel condition.

If drying activities are suspended after a release, acceptable practice would be to place the transportation package in a safe condition. Examples of followup actions the staff finds acceptable include ensuring that the fuel design-basis temperature limit is not exceeded, and preventing any inadvertent ingress of oxidizing species to the containment (or canister) cavity that may compromise cladding integrity. The staff has reasonable assurance that the fuel is unlikely to degrade if the fuel atmosphere is inert and the temperature is controlled. Therefore, backfilling with helium consistent with the CoC is expected to prevent degradation of the fuel until drying operations resume.

The staff recognizes that no fuel qualification test method is 100 percent accurate, and quantifying the reliability is difficult because of the low failure rate of modern fuel (about 0.001 percent). Nevertheless, a user's evaluation of operating experience may identify limitations of a given technique, and the staff recommends that the user take appropriate actions consistent with the approved site procedures and QA program. Such actions may include revising operating procedures to limit the use of certain techniques, depending on the type of fuel or sensitivity limits of the instrumentation, as well as assessing the need for secondary characterization.

The staff considers that the release of noble fission gases during SNF loading operations is possible through existing pinholes or hairline cracks in undamaged cladding. Therefore, if the fuel being loaded was adequately classified and protected against inadvertent degradation, the staff considers that the release of noble fission gases during loading operations is not indicative of the presence or development of a cladding gross breach.

7B.5 References

10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

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."

Entergy Operations Inc., 2014, "Special Report—Dry Fuel Cask MPC-24-060, Arkansas Nuclear One–Units 1 and 2 Docket Nos. 50-313 and 50-368, and 72-13, License Nos. DPR-51 and NPF-6," letter and attachment from Stephanie L. Pyle, Entergy Operations, Inc., to "Document Control Desk," U.S. Nuclear Regulatory Commission, October 13, 2014, Agencywide Documents Access and Management System (ADAMS) Accession No. ML14286A037.

International Atomic Energy Agency, "Management of Damaged Spent Nuclear Fuel," Nuclear Energy Series No. NF-T-3.6, June 2009, https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1395_web.pdf.

U.S. Nuclear Regulatory Commission, 2005, "Exemption from 10 CFR 72.212 and 72.214 for Dry Spent Fuel Storage Activities–Arkansas Nuclear One (TAC NO. L23826)," letter and attachment from William Ruland, NRC Office of Nuclear Material Safety and Safeguards, to Dale E. James, Acting Director, Arkansas Nuclear One, Entergy Operations, Inc., April 8, 2015, ADAMS Accession No. ML052510724.

NRC, 2016, "Arkansas Nuclear One, Units 1, 2, and Independent Spent Fuel Storage Installation (ISFSI)–NRC Inspection Report 05000313/2015011, 05000368/2015011, and 07200013/2015001," letter and attachment from Ray L. Keller, P.E., Chief, Division of Nuclear Materials Safety, to Jeremy Browning, Site Vice President, Arkansas Nuclear One, Entergy Operations, Inc., January 21, 2016, ADAMS Accession No. ML16021A485.