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Dry Cask Storage Characterization Project—Phase 1: CASTOR V/21 Cask Opening and Examination

Idaho National Engineering and Environmental Laboratory

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ABSTRACT

This report documents visual examination and testing conducted in 1999 and early 2000 at the Idaho National Engineering and Environmental Laboratory (INEEL) on a Gesellschaft für Nuklear Service (GNS) CASTOR V/21 pressurized water reactor (PWR) spent fuel dry storage cask. The purpose of the examination and testing is to develop a technical basis for renewal of licenses and Certificates of Compliance for dry storage systems for spent nuclear fuel and high-level waste at independent spent fuel storage installation sites. The examination and testing was conducted to assess the condition of the cask internal and external surfaces, cask contents consisting of 21 Westinghouse PWR spent fuel assemblies from Dominion's (formerly named Virginia Power) Surry Power Station and cask concrete storage pad. The assemblies have been continuously stored in the CASTOR cask since 1985. Cask exterior surface and selected fuel assembly temperatures, and cask surface gamma and neutron dose rates were measured. Cask external/internal surfaces, fuel basket components including accessible weldments, fuel assembly exteriors, and primary lid seals were visually examined. Selected fuel rods were removed from one fuel assembly, visually examined, and then shipped to Argonne National Laboratory for nondestructive, destructive, and mechanical examination. Cask interior crud samples and helium cover gas samples were collected and analyzed. The results of the examination and testing indicate the concrete storage pad, CASTOR V/21 cask, and cask contents exhibited sound structural and seal integrity and that long-term storage has not caused detectable degradation of the spent fuel cladding or the release of gaseous fission products between 1985 and 1999.

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EXECUTIVE SUMMARY

To augment the technical basis for renewals of licenses and Certificates of Compliance for dry storage systems for spent nuclear fuel and high-level waste at independent spent fuel storage installation sites, a dry cask storage system, the CASTOR-V/21, was examined and tested after it had undergone an extended period of service. This report presents the results of visual examination and testing conducted in 1999 and early 2000 on a Gesellschaft für Nuklear Service (GNS) CASTOR V/21 pressurized water reactor (PWR) spent fuel storage cask located at the Idaho National Engineering and Environmental Laboratory (INEEL). The examination and testing evaluated the condition and integrity of the exterior and interior surfaces of the CASTOR V/21 cask and its contents consisting of 21 Westinghouse 15 × 15, PWR spent fuel assemblies from Dominion's (formerly named Virginia Power) Surry Power Station) and the concrete spent fuel storage pad after a 14-year storage duration. Performance of activities was conducted under a standard order agreement between the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (NRC-RES) and the U.S. Department of Energy (DOE), Idaho Field Office, the INEEL operated by Bechtel BWXT Idaho. The project (Dry Cask Storage Characterization) was jointly funded by Nuclear Regulatory Commission-RES, Electric Power Research Institute (EPRI), DOE-Office of Civilian Radioactive Waste Management (RW), and DOE-Office of Environmental Management (EM) to examine the spent fuel and cask in dry storage at the INEEL. Examination and testing, including video and photographic scans, were performed at the INEEL Test Area North (TAN) Hot Shop facility and consisted of pretest preparations, performance testing, and post-test activities.

Pretest preparations included conducting cask-external examination, measurement of cask exterior surface temperature, extraction of helium cover gas samples and analysis, and handling and transport of the cask from the concrete storage pad to the TAN Hot Shop. Performance examination and testing included cask remote primary lid removal, remote indirect visual examination of cask internals, primary lid seals, 21 PWR spent fuel assemblies, and measurement of selected fuel assembly temperatures. Interior crud samples were taken, and cask surface gamma and neutron dose rates were measured. Selected fuel rods were removed from one fuel assembly, visually examined, and shipped to Argonne National Laboratory West for nondestructive, destructive, and mechanical examination. Post examination and test activities included examination and removal of one fuel assembly from the TAN Hot Cell and reinstallation in the CASTOR V/21, examination of the primary lid and cask sealing surfaces, installation of new primary lid seals, primary lid installation, refill of cask with fresh helium, leak testing of primary lid seals, and extraction of helium cover gas samples and analysis.

The CASTOR-V/21 PWR spent fuel storage cask consists of a nodular cast-iron body. The cast-body material was found to be in very good condition and continues to provide effective gamma shielding. The cask is 4.9 m (16 ft) high and 2.4 m (8 ft) in diameter and weighs approximately 112 tons when loaded with PWR spent fuel. Two concentric rows of polyethylene rods incorporated in the cask wall continues to provide the required level of neutron shielding. The external surface consists of heat transfer fins oriented

circumferentially around the cask surface. The fuel basket within the cask is configured to hold 21 PWR spent fuel assemblies and is constructed of stainless steel and borated stainless steel for criticality control. The Surry spent fuel assemblies are of a standard Westinghouse 15 × 15 rod design. The cask is closed with a primary lid having both elastomer and metallic O-rings to seal the cask cavity from the environment. A secondary lid, used in commercial applications, was not used in this testing program (Section 1.3.4).

Based on 1985 pretest ORIGEN2 predictions, fuel assembly decay heat generation rates totaled approximately 28 kW at the start of testing and 27 kW at the end of testing in 1985. In 1985, thirteen of the twenty-one fuel assemblies had decay heat rates near 1 kW; the remaining eight assemblies had decay heat rates of approximately 1.8 kW. The 1999 measured temperature of the exterior cask surface ranged from 103°F (39°C) to 141°F (61°C) (ambient temperature 91.2°F [32.9°C]). The hottest of selected fuel assemblies measured 309°F (154°C). This compares to 109°F (43°C) to 221°F (105°C) on the exterior cask surface, and a peak internal cask guide tube temperature of 676°F (358°C) measured in 1985 with a nitrogen backfill, with lid in place and cask in the vertical orientation (EPRI NP-4887).

Examination and Testing Findings

Based on the 1999 examination and testing results, there was no evidence of cask, shielding, or fuel rod degradation during long-term (14 years) storage that would affect cask performance or fuel integrity. There was no evidence of exterior or interior cask deterioration during storage period, nor were there any signs of seal or shielding failure. The fuel was intact, and there were no indications of any significant cladding creep or rod bow. There was some crud adherent to the rods, but no crud appears to have fallen into either the spacers or cask bottom. The crud is not a result of the fuel or cladding oxidation during dry storage, but rather the results of oxidation of steel components by the coolant in reactor operations. However, fifteen of sixteen stitch welds (Appendix B) in the triangular air channels at the perimeter of the basket were found to be cracked. These welds attached the basket borated stainless partition plates to the basket barrel. Based on the cask testing performed in 1985 (EPRI NP-4887 report), it was concluded that applicable triangular air channels were not examined back in 1985, and the defective stitch welds cracked during high temperature testing cycles and test configurations that the cask experienced during the 1985 testing campaign rather than during the long-term storage period. These welds in their current condition do not affect required cask performance parameters.

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ACRONYMS

ANL-E	Argonne National Laboratory-East
ANL-W	Argonne National Laboratory-West
CASTOR	Cast Iron Cask for Storage and Transport of Radioactive Material
DOE	U.S. Department of Energy
EDS	Energy Dispersive Spectrometry
EM	Office of Environmental Management
EPRI	Electric Power Research Institute
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LLNL	Lawrence Livermore National Laboratory
LTSM	Long-Term Storage Monitoring (Program)
NRC	U.S. Nuclear Regulatory Commission
PNNL	Pacific Northwest National Laboratory
RES	Office of Nuclear Regulatory Research
RW	Office of Civilian Radioactive Waste Management
SEM	Scanning Electron Microscope
SFB	Spent Fuel Behavior (Study)
TAN	Test Area North
TED	track etch dosimeter
TLD	thermoluminescent dosimeter

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1. INTRODUCTION

1.1 Regulations

Most nuclear power plants in the United States were not originally designed with a storage capacity for the spent fuel generated over the operating life by their reactors. Utilities originally planned for spent fuel to remain in the spent fuel pool for a few years after discharge and then to be sent to a reprocessing facility. Because reprocessing has been eliminated, and no other option for spent fuel disposition currently exists, utilities expanded the storage capacity of their spent fuel pools by using high-density storage racks. This has been a short-term solution with many utilities having reached, or soon will reach, their spent fuel pool storage capacity (Fisher and Howe 1998). Utilities have developed independent spent fuel storage installations as a means of expanding their spent fuel storage capacity on an interim basis until the geologic repository is available to accept spent fuel for permanent storage.

The U.S. Nuclear Regulatory Commission (NRC) promulgated 10 CFR Part 72 (Title 10 1999) for the independent storage of spent nuclear fuel and high-level radioactive waste outside reactor spent fuel pools. Part 72 currently limits the license term for an independent spent fuel storage installation to 20 years from the date of issuance. In preparation for possible license renewal, the NRC, Office of Nuclear Material and Safeguards, Spent Fuel Project Office is developing the technical basis for renewals of licenses and Certificates of Compliance for dry storage systems for spent nuclear fuel and high-level radioactive waste at independent spent fuel storage installation sites. These renewals would cover periods from 20 to 100 years and would require development of a technical basis for ensuring continued safe performance under the extended service conditions. An analysis of past performance of selected components of these systems is required as part of that technical basis. The components include the spent fuel and all structures, systems, and other components with functions important to safety. The safety functions, which apply for normal, off-normal, and accident conditions, are as follows: maintain subcriticality, maintain confinement, ensure that radiation rates and doses to workers and the public do not exceed acceptable levels and remain as low as reasonably achievable, maintain retrievability, and ensure heat removal as needed to meet the safety requirements.

1.2 Objectives and Scope

In the mid-1980s, the U.S. Department of Energy (DOE) procured three prototype dry storage casks for testing at the Idaho National Engineering and Environmental Laboratory (INEEL): MC-10, TN-24P, and CASTOR V/21. The primary purpose of the test was to benchmark thermal and radiological codes and to determine the thermal and radiological characteristics of the three casks. The CASTOR V/21 cask was loaded with irradiated assemblies from the Surry Nuclear Power Plant and then tested in a series of configurations using a variety of fill gases. Because the tests were not intended to be fundamental fuel behavior tests, the fuel prior to the tests had undergone only minimal characterization consisting of visual examination of the outside of the assemblies and ultrasonic examination to ensure no breached rods would be included. During the tests, the temperature at various locations was monitored, and the cover gas was periodically analyzed to determine if any leaking rods had developed. None were found. The details of these tests have been reported in a number of documents.

In 1999, a project was jointly funded by NRC-Office of Nuclear Regulatory Research (RES), Electric Power Research Institute (EPRI), DOE-Office of Civilian Radioactive Waste Management (RW), and DOE-Office of Environmental Management (EM) to examine the Surry spent fuel in dry storage at the INEEL. This project is intended to provided confirmatory data to be used by licensees submitting an

application (no later than 2004 for the first licensee) for continuing dry storage beyond 20 years and by the NRC staff in their technical licensing reviews. The objectives of the Dry Cask Storage Characterization Project are to:

1. Obtain data to confirm the predicted long-term integrity of dry storage cask systems and spent nuclear fuel under dry storage conditions

2. Provide data to augment the technical bases and criteria for evaluating the safety of spent fuel storage and transportation systems, and for extending dry cask storage licenses.

Phase 1 of the overall program involved moving the CASTORV/21 dry storage cask from the INEEL Test Area North (TAN) storage area to the INEEL TAN Hot Shop facility; extracting helium cover gas samples and analysis; obtaining temperature readings of the cask exteriors; performing a radiation survey; performing video and photographic inspection of the cask exterior and interior surfaces; examining cask primary lid seals, fuel assembly exteriors and selected fuel rods; and returning the cask to storage. The specific tasks executed in the project are as follows:

Task 1.1—Equipment: Design, fabricate, and purchase of material, equipment, and fixtures necessary to (1) move the cask; (2) remove and replace fuel assemblies; (3) remove fuel rods from one selected assembly; and (4) videotape and photograph the external and internal surfaces of the cask, fuel assemblies, and fuel rods.

Task 1.2—Procedures and Training: Develop procedures, obtain required reviews and approvals, perform needed training, and other pertinent activities associated with (1) movement of the cask, (2) removal of the designated fuel assemblies or canisters, (3) removal of fuel rods, (4) returning the fuel assemblies or canisters to the cask, and (5) returning the cask to storage.

Task 1.3—Hot Shop/Cell Rental: Use the TAN Hot Shop and Hot Cell and TAN operations oversight.

Task 1.4—Inspection of Cask and Internals: Activities include (1) temperature readings of the cask exterior; (2) radiation survey; (3) video and photographic inspections of the cask exterior and interior, seal, and storage pad; (4) video and photographic inspections of sixteen fuel assemblies in general and five assemblies in detail and twelve fuel rods selected from one fuel assembly; (5) obtain crud and smear samples; and (6) temporary storage of the fuel rods until transported to ANL for Phase 2 evaluations.

Task 1.5—Transportation: Transport the fuel rods to Argonne National Laboratory-West (ANL-W) and crud and smear samples to the Idaho Nuclear Technology and Engineering Center (INTEC) for analysis. Return the fuel rod pieces and other material (e.g., crud samples) to the TAN facilities for permanent storage.

Phase 2 of the program, to be reported by others in the future, involves nondestructive, destructive, and mechanical examinations of dry-stored spent nuclear fuel elements. This will provide quantitative and qualitative information concerning the integrity of the fuel. Examples of the type of information that will be obtained include: in situ creep; percentage of fission gas release, internal rod pressure; oxide thickness, hydride morphology and orientation, residual cladding thickness, cladding microstructure; hydrogen content; creep rates, breakaway temperature; tensile strengths; and ductility. These tasks will take place at either ANL-W or ANL-East (ANL-E). The Phase 2 task interface between TAN and ANL-W at the INEEL are activities at ANL-W associated with the transportation of the fuel rods from the TAN facilities at the INEEL to ANL-W, the fuel rod segments from ANL-W to ANL-E, and the return of the fuel rod pieces and other material from ANL-W and ANL-E to the TAN facilities for storage.