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SPENT FUEL INTEGRITY DURING DRY STORAGE

M. A. McKinnon

July 1995

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SPENT FUEL INTEGRITY DURING DRY STORAGE

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ABSTRACT

Information on spent fuel integrity is of interest in evaluating the impact of long-term dry storage on the behavior of spent fuel rods. Spent fuel used during cask performance tests at the Idaho National Engineering Laboratory (INEL) offers significant opportunities for confirmation of the benign nature of long-term dry storage. The cask performance tests conducted at INEL included visual observation and ultrasonic examination of the condition of the cladding, fuel rods, and fuel assembly hardware before dry storage and consolidation of the fuel; and a qualitative determination of the effect of dry storage and fuel consolidation on fission gas release from the spent fuel rods. A variety of cover gases and cask orientations were used during the cask performance tests. Cover gases included vacuum, nitrogen, and helium. The nitrogen and helium backfills were sampled and analyzed to detect leaking spent fuel rods. At the conclusion of each performance test, periodic gas sampling was conducted on each cask as part of a surveillance and monitoring activity. Continued surveillance and monitoring activities are being conducted for intact fuel in a CASTOR V/21 cask and for consolidated fuel in a VSC-17 cask. The results of the gas sampling activities are reported in this paper.

INTRODUCTION

In response to the Nuclear Waste Policy Act of 1982 (NWPA), a Solicitation for Cooperative Agreement Proposal (SCAP) was issued in May 1983 by the DOE-Richland Operations Office (DOE-RL) to help the private sector with their spent fuel storage problems, and proposals were received in August 1983. Virginia Power (VP) proposed that pressurized water reactor (PWR) spent fuel storage cask performance testing be conducted at a federal site in support of its at-reactor license demonstration. VP and DOE signed a Cooperative Agreement in March 1984, and VP signed a separate agreement with the Electric Power Research Institute (EPRI), essentially establishing a three-party cooperative agreement.

The scope of the Cooperative Agreement included performance tests of three different metal storage casks loaded with unconsolidated spent nuclear fuel. The tests were conducted at INEL. After cask performance testing with unconsolidated fuel was completed in the VP/DOE cooperative program, a decision was made by DOE and EPRI to extend the performance testing to include consolidated fuel in the Transnuclear, Inc., TN-24P cask. Dry rod consolidation was conducted at INEL as a separate DOE-only funded program.

Prior to testing, the Surry PWR spent fuel assemblies used in the cask performance tests were characterized using in-basin ultrasonic examinations and video scans. Cask internal cover gas samples were taken during testing. After testing, selected fuel assemblies were videotaped and photographed. Then fuel assemblies used in the TN-24P and MC-10 cask performance tests, along with a few Turkey Point reactor spent fuel assemblies, were consolidated and loaded into the TN-24P cask for a DOE-funded performance test. Later, a cooperative agreement was established with Sierra Nuclear Corporation (SNC), and 17 of the consolidated fuel canisters from the TN-24P cask were used in a performance test of SNC's ventilated concrete cask.

Performance test runs involved a combination of cover gases and cask orientations. The backfill environments used were vacuum, nitrogen, and helium; nitrogen and helium were sampled and analyzed to detect leaking spent fuel rods. The integrity of the fuel assemblies was determined from cover gas sampling (Creer 1986; McKinnon 1986, 1987a, 1987b, 1989, 1992, 1993). At the conclusion of each performance test, periodic gas sampling was conducted on each cask as part of a cask surveillance and monitoring activity.

This report combines the gas sampling information from the cask performance tests and cask monitoring activities. It documents the condition of the fuel from the Surry reactor prior to testing and the effect of testing on fuel integrity as ascertained through gas sampling during cask performance tests at INEL using

both intact and consolidated PWR spent fuel. It also includes results of a prior test using boiling water reactor (BWR) spent fuel and the REA-2023 cask at Morris, Illinois. Recent gas sampling data associated with cask surveillance and monitoring at INEL is also included. The pretest condition of the fuel and the significant results obtained from gas sampling during and after performance testing are described as well.

Three types of spent fuel and five casks have been used during the cask performance testing and demonstration program. BWR 7x7 spent fuel assemblies were used for the performance test of the REA-2023 cask. Westinghouse 15x15 PWR was used in the CASTOR V/21, TN-24P, MC-10, and NUHOMS performance tests. A portion of this fuel was consolidated at INEL and used in performance tests of the TN-24P and VSC-17 casks (also at INEL). Table 1 gives a summary of the fuel used in each of the performance tests.

The BWR assemblies used during the REA 2023 cask performance test were a general Electric 7x7 design and were taken from the Nebraska Power Cooper Reactor. The intact PWR assemblies used in the performance tests conducted at INEL were taken from VP's Surry Reactors. After the performance tests using intact PWR fuel, INEL consolidated 48 PWR fuel assemblies of the Westinghouse 15x15 design into 24 canisters of consolidated fuel that were used in two subsequent cask performance tests. A description of the fuel is contained within the performance reports.

SPENT FUEL INTEGRITY

PRETEST FUEL INSPECTION Four examination methods were used to assess the integrity of the spent fuel used in the cask performance tests. Methods common to the PWR and BWR fuel included visual observations such as full-length black and white videos and color photographs; and analyses of the cover gas in the cask. In addition to these methods, the BWR spent fuel was examined by in-basin sipping and the PWR spent fuel from the VP's Surry Reactor was examined using an in-pool ultrasonic examination.

In-Basin Sipping In-basin sipping consisted of placing a hood over the selected BWR assembly and analyzing the water that was drawn off the top of the assembly. All the sipping data were compared with background readings to assess fuel integrity. Although the differences between the pretest and post-test radionuclide concentrations vary, the values were lower than if the assembly contained leaking fuel rods. The sipping results did not indicate any leaking fuel rods in any of the fuel assemblies used in the cask either before or after cask testing.

In-Basin Ultrasonic Inspections In-basin ultrasonic inspections were performed on PWR fuel at VP's Surry Reactor using the Babcock & Wilcox Failed Fuel Rod Detection System (FRDS). The FRDS system uses ultrasonic techniques to differentiate between leaking and non-leaking rods by detecting moisture in the former. Only Surry Reactor fuel assemblies with non-leaking fuel rods were used in the performance tests.

TABLE 1. Spent Fuel Assembly Characteristics

Cask	REA-2023	CASTOR V/21	TN-24P	TN-24P(a)	MC-10	VSC-17(a)
Fuel Type	BWR	PWR	PWR	PWR	PWR	PWR
Assembly Type	7 x 7	15 x 15	15 x 15	Consolidated 15 x 15	15 x 15	Consolidated 15 x 15
Burnup, GWd/MTU	24-28	24-35	29-32	24-35	24-35	26-35
Cooling Time, years	2.3-3.4	2.2-3.8	4.2	6.2-12.2	4.6-10.1	8.8-14.3
Enrichment, wt%	2.5	2.9-3.1	2.9-3.2	1.9-3.2	1.9-3.2	2.56-3.2
Assembly Decay Heat, W	235-300	1000-1800	832-919	701-1185	400-700	700-1050
Average, W	290	1350	860	970	530	877
Cask, kW	15.2	28.4	20.6	23.3	12.6	14.9

(a) Performance test using consolidated fuel in the cask.

Visual, Video, and Photographic Examinations The PWR fuel assemblies were examined visually to establish their general condition after shipment from VP, handling at the INEL hot shop, cask performance testing, and during consolidation. Similar exams were made of the Cooper BWR fuel during the REA 2023 performance tests at GE-Morris. Two kinds of visual examinations were used: black-and-white videos and color photography of selected fuel assemblies.

The black-and-white videos taken at GE-Morris, VP, and INEL did not provide sufficient detail to characterize the crud or very small features on the fuel rods. They did not reveal any indication of significant variations in the fuel rods after shipment, handling, and performance testing. The resolution of the videotapes did not provide enough information to adequately determine the integrity and condition of the fuel and fuel cladding. Examination of the video scans showed that all the fuel assemblies and fuel rods look basically the same when viewed from outside the assemblies. There was some discoloration of the fuel rod cladding in the area of the grid spacers, which was expected.

Color photographs showed that a typical orange/reddish crud (probably Fe_2O_3) was evenly deposited on all of the Zircaloy 2 cladding and fuel assembly hardware. There were no noticeable changes in the characteristics or adherence of the crud during handling operations involving the spent fuel assemblies at GE-Morris or INEL. Some scratches and worn spots were apparent on the spacer grids and some fuel rods, but these features did not change as a result of examination or handling operations. In general, the fuel rods were in excellent condition, with a very adherent crud layer.

Additional visual examinations of the fuel were conducted during the dry rod consolidation program. According to Vinjamuri (1988a, 1988b):

> No noticeable cladding defects in the rod surfaces were observed for any of the fuel processed. The oxide layer on the surface of the fuel rods appears to be intact and firmly attached to the cladding. The oxide layer does not appear to be loose, thick, soft, or powdery. However, the oxide layer and some of the zirconium cladding was scraped from the rod surface by the spacer grids as the rod was pulled during fuel consolidation. Very little crud buildup on the surfaces of the rods was observed. The surfaces of the rods displayed only a thin oxide layer, which had the appearance of surface discoloration rather than any rough or loose material. The rod surfaces are discolored near the spacer grids. The discoloration has an appearance of a dark mottling of the surface and is progressively more predominant from the middle of the rod length toward the rod bottom. The rods are generally clean, with limited amounts of clad discoloration and oxidation . . . The evidence of fuel rod growth since fabrication was visually obvious during the consolidation process.... Length variation between rods appears to be as much as 2 cm (0.8 inch). The rods that grew longer than others appeared to be randomly located within the fuel assembly.

CASK COVER GAS SAMPLING The cask cover gas was sampled several times during each cask performance test to evaluate the integrity of the spent fuel rods. Each sample was collected in a separate 500-cc stainless steel cylinder that had been leak-tested before sampling. Initially, during the CASTOR-V/21 cask performance test, the cylinders were equipped only with quick-disconnect fittings and no bellows-sealed valves as part of the closure. During the early sampling efforts with the CASTOR-V/21 cask, the cover gas samples in the cylinders were diluted with ambient air from the vicinity of the sampling apparatus, air that leaked into the cylinder during shipment, and argon introduced at Lawrence Livermore National Laboratory (LLNL). In many cases, this dilution was made more severe by the collection of small amounts of cask cover gas, presumably due to short equilibration times between the cask and the sample bottle during the actual cask cover gas collection procedure. The end effect of small, diluted samples on the cask cover gas analyses was to increase detection limits, increase measurement uncertainties, and introduce questions of sample validity. Once bellows-sealed valves were added to the sampling cylinders, the problem of air leakage into the sampling cylinders was eliminated.

Gas sample analysis included mass spectroscopy and radiochemical gamma analysis. Mass spectra were analyzed for all common fixed gases with masses less than 100 to verify the purity of backfill gas composition. Only N_2, 0_2, He, Ar, and CO_2 concentrations above 0.01% are detected in any of the samples. The integrity of the fuel rods was assessed from the radionuclide concentration based on gamma spectroscopy.

However, the relatively low amounts of ^{85}Kr detected indicate that no leaking fuel rods were present in the GNS CASTOR-V/21 and MC-10 casks during performance testing with unconsolidated fuel and up to about a year after testing. At this time, gas sampling in these casks was discontinued. The final gas sample from the CASTOR-V/21 cask during this period was taken in December 1986. In September 1994, the CASTOR cask was opened and backfilled with a fresh charge of helium gas. The pre- and post-test backfills were checked for purity. Gas samples taken in March 1995, after six months of gas residence in the cask, did not contain detectable amounts of krypton-85, which indicates no leaks from fuel rods during that storage period. This is particularly significant, because the first few assemblies loaded in the CASTOR-V/21 cask were exposed to air for approximately 200 hours during incremental loading of the cask and fuel assembly/basket inspections at a reduced temperature. In addition, after testing was completed and long-term surveillance started, all the fuel assemblies were in a 70% He and 30% air environment for approximately four months, because a quick disconnect fitting on the CASTOR-V/21 cask lid had not sealed properly.

Two casks loaded with intact fuel have shown krypton gas concentrations indicative of a leaking fuel rod. During the performance test of the REA-2023 cask, krypton gas was detected after the cask was rotated from a vertical to a horizontal orientation. The accumulated amount of krypton gas released to the cask was consistent with the release from a single fuel rod (Barner 1985; Guenther 1988). The cladding defect was assumed to be very small since the release rate was essentially linear during 2.5 months of testing. There was no confirmation of a leaking fuel rod either by visual inspection or sipping of the fuel assemblies after the cask test. The gas analyses provided the only indication of a leaking fuel rod. The leaking fuel rod had no impact on the basin operation or handling of the fuel assemblies subsequent to the cask test.

The other cask loaded with intact fuel that showed krypton levels indicative of a leaking fuel rod was the TN-24P cask. In this performance test, the cumulative amount of ^{85}Kr detected just after the cask was rotated from a vertical to a horizontal orientation indicated that a fuel rod had leaked during this portion of the test. The decay in the leak rate, as indicated by subsequent gas samples, indicates that the leak was small. It took several days to vent the gas from the fuel rod.

In May 1987, 36 of the 48 intact fuel assemblies in the TN-24P and MC-10 casks, plus 12 intact assemblies that had been in the Turkey Point Reactor, were consolidated into 24 consolidated fuel canisters as part of INEL's Dry Rod Consolidation Technology Project. The consolidated fuel canisters were then used in performance tests of the TN-24P and VSC-17 casks. During the fuel rod consolidation process, the exhaust gases from the consolidation area were monitored to detect the release of radioactive gases from the fuel that would indicate a cladding failure. In the consolidation reports (Vinjamuri 1988a, 1988b), one of the conclusions reached was that all fuel rods from the 48 assemblies were pulled and canisterized without rod failures.

Later, during the performance test of the TN-24P cask using consolidated fuel, ^{85}Kr was released to the cask. Based on a combination of ORIGEN2 predictions and experimental measurements (Barner 1985; Guenther 1988), it was estimated that four or more fuel rods may have developed leaks between the end of cask loading and the beginning of cask performance testing, three or more fuel rods during cask performance testing, and another five fuel rods in the six-month period following testing. The rate of ^{85}Kr release was observed to decrease with time from cask loading. Shortly after the last gas sample was taken from the fully loaded TN-24P cask, 17 canisters of consolidated fuel were removed from the TN-24P cask and loaded into the VSC-17 cask. The performance tests for the VSC-17 cask showed a nominal ^{85}Kr but not enough to indicate a new leaking fuel rod. Since the end of the VSC-17 performance testing in early 1991, until September 1994, the VSC-17 has been undisturbed. Recent gas samples, taken since September 1994, indicate that the atmosphere in the VSC-17 has not changed significantly. There has been a small amount of ^{85}Kr release, below the quantity expected for a single rod release, and there has been buildup of hydrogen in the cask. The amount of hydrogen is consistent with off-gassing of the RX277 neutron shield material in the lid. Similar amounts of hydrogen were observed during cask performance testing.

The amount of ^{85}Kr released during and after the TN-24P cask performance test with consolidated fuel is significantly higher than was released in previous cask testing with unconsolidated fuel. Before this test, four cask performance tests of similar duration and scope had been performed; only two indications of ^{85}Kr release were observed. The magnitude of the releases in the previous tests and surveillance periods indicated that each was limited to a single rod cladding breach. The previous tests involved about 16,700 spent fuel rods, whereas this test involved about 9800 rods. It is

hypothesized that the greater magnitude of ^{85}Kr released in this test and post-test surveillance is due to additional cladding leaks caused by enlargement of incipient cladding flaws during pulling and flexing of the fuel rods in the consolidation process. The enlarged cladding flaws, combined with cladding creep during cask testing and surveillance periods, allowed leak paths to develop. The leakage has not affected operations.

SUMMARY

Radiochemical gamma analysis of gas samples from cask performance tests and subsequent cask surveillance and monitoring activities provides an indication of spent fuel integrity during dry storage. The gas sampling analysis indicates that dry storage of spent fuel in an inert atmosphere is benign. In general, fuel handling activities have a more significant impact on fuel rods than does extended dry storage in an inert atmosphere.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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April 16, 2013

NRC INFORMATION NOTICE 2013-07: PREMATURE DEGRADATION OF SPENT FUEL STORAGE CASK STRUCTURES AND COMPONENTS FROM ENVIRONMENTAL MOISTURE

ADDRESSEES

All holders of, applicants for, and registered users of spent fuel storage system certificates of compliance (CoCs) as well as all holders of and applicants for an independent spent fuel storage installation (ISFSI) license under Title 10 of the *Code of Federal Regulations* (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 inform addressees of recent operating experience on environmental moisture causing premature degradation of structures and components important to safety during spent nuclear fuel storage operations. The NRC expects recipients to review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Peach Bottom Atomic Power Station ISFSI

On October 11, 2010, a cask seal pressure monitoring system low pressure alarm was received for Cask TN-68-01 during storage at the Peach Bottom Atomic Power Station ISFSI. Cask TN-68-01 is a bolted closure cask system with a double mechanical O-ring seal (cask lid seal) that provides confinement between the lid and cask interface. The cask had been in service at Peach Bottom since June 2000. Figure 1 shows a cross sectional view of the lid region for a TN-68 cask. A protective cover was installed on the lid region to protect the system from external weathering. During disassembly and removal of the protective cover, the licensee found streaks of rust on the underside of the cover, a pronounced pattern of rust directly under the access plate, and water or signs of moisture around most of the bolt lid holes and bolts. The licensee found the elastomer O-ring seal on the bottom of the protective cover to be completely intact and sealed against the top of the cask lid. After performing a sequence of helium leak tests, the licensee identified that the outer sealing surface of the main cask lid seal was leaking at a rate greater than allowed by the CoC technical specifications. The licensee returned the

spent fuel assemblies to the spent fuel pool to perform additional inspections on the cask lid and seals. The initial evaluation revealed corrosion of the outer portion of the cask lid seal, lower than expected torque on some of the main lid bolts, and corrosion on the threads of the lid bolts. The inner portion of the cask lid seal remained intact; therefore, the cask's primary confinement was not compromised.

A root cause evaluation concluded that the seal leakage was caused by corrosion of the outer portion of the cask lid seal from water infiltration through the access plate in the protective cover. The water infiltration caused galvanic corrosion of the outer portion of the cask lid seal due to the presence of moisture at the interface of the aluminum-clad cask lid seal and the stainless steel clad cask body sealing surface. The presence of the moisture at the interface of the two dissimilar metals set up a galvanic cell that caused the aluminum to corrode and allowed helium to leak through the outer portion of the cask lid seal. The root cause evaluation further stated that the helium leak was attributed to inadequate sealing of the access plate in the protective cover and a lack of any verification of the integrity of the water-tight cover. The primary corrective actions developed by the cask vendor and the licensee involved improving the access plate design and developing a method for verifying protective cover seal integrity. Additional corrective actions, which were incorporated into operating procedures immediately after the event, included a change to the lid bolt torquing process and ensuring access plate gaskets and O-rings were inspected at installation.

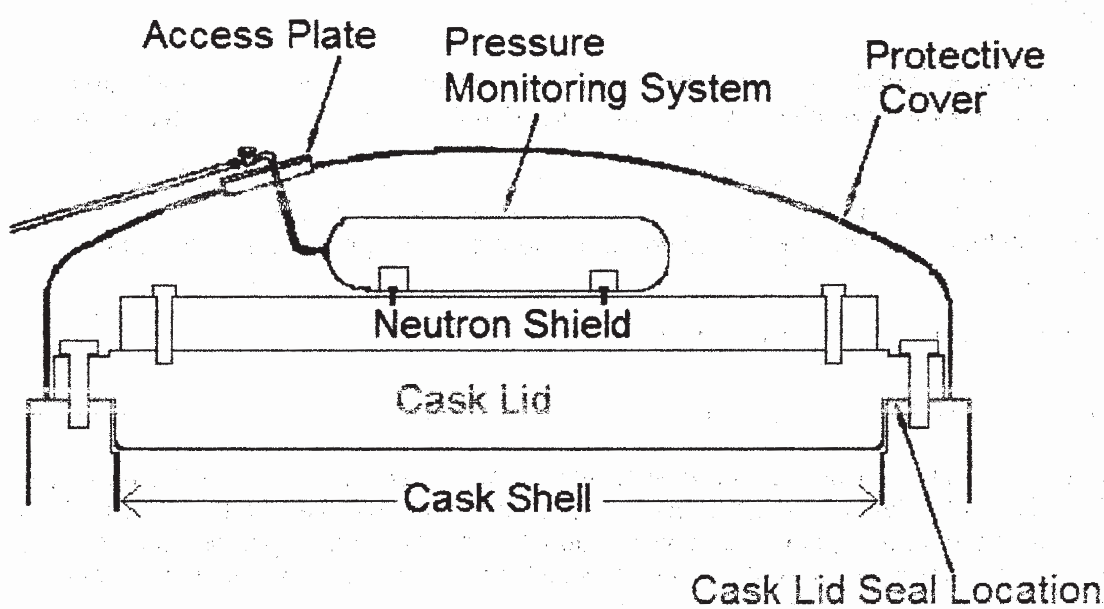

Figure 1

Additional information is available in "Peach Bottom Atomic Power Station—NRC Inspection Report 05000277/12010010," dated July 8, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML111890441).

<u>Three Mile Island, Unit 2 ISFSI at the Idaho National Laboratory Site</u>

The Three Mile Island, Unit 2 ISFSI uses NUHOMS-12T horizontal storage modules (HSMs). The HSMs were delivered to the Idaho National Laboratory site in 1999 as precast concrete

components. The storage system consists of an external rectangular reinforced concrete vault (i.e., HSM) with a storage canister resting horizontally on internal rails inside the HSM. The prefabricated modules consist of a body and a roof joined together by anchor bolts. All sections were a minimum of 0.6-meters (2-feet) thick. In 2000, the licensee noted cracks in the HSMs, and concluded they were cosmetic and insignificant. However, in 2007, the licensee observed continued cracking, crazing and spalling as well as increased efflorescence on the HSM surfaces. The efflorescence was a solid, whitish crystalline material which was determined through sampling and analysis to be calcium carbonate. The licensee performed an evaluation in 2007, during which it determined that the HSMs were capable of performing their design basis functions. In 2008, the licensee noted that 28 of the 30 HSMs had cracks, mostly emanating from the anchor bolt blockout holes with widths up to 0.95 centimeters (0.38 inches). At that time, the licensee determined that the HSMs appeared to be prematurely deteriorating and that continued crack growth could impact the ability of the HSMs to fulfill their originally planned 50-year design service life. Subsequent evaluations by the licensee initiated the development of an annual inspection plan for the HSMs and base mat as well as an examination of the inside of the HSMs. The evaluation also recommended that the licensee retain the services of a company experienced and qualified in testing and evaluating concrete to determine the degradation mechanism and make recommendations both for repairs and to prevent further degradation. Although the cracking was discussed with the storage system vendor, the licensee chose an independent vendor to perform an evaluation of the HSMs and base mat concrete in 2009. The evaluation included a field investigation and laboratory analysis to evaluate the concrete material quality, strength, and long-term durability potential. The conclusion reached was that water had entered the anchor bolt blockout holes on the roof of the HSMs. Subsequent freeze and thaw cycles initiated the crack formation. Repetition of the process resulted in both continued crack growth and the efflorescence growth identified in 2007. In addition to identifying the root cause of the cracking, the report also suggested repairs (injecting resin into the cracks), preventative actions (e.g., installing caps over the anchor bolt blockout holes), and monitoring (use of crack gauges). The licensee incorporated the suggested corrective actions.

Additional information is available in "Three Mile Island, Unit 2, ISFSI—NRC Inspection of the Independent Spent Fuel Storage Installation—Inspection Report 07200020/2012-001," dated August 14, 2012 (ADAMS Accession No. ML12228A457).

DISCUSSION

The instances described above illustrate how the intrusion of water can potentially decrease the effective life of both the structures and components of a spent fuel storage system. In one instance, the presence of water not only caused chemical degradation through oxidation of one metal, but it also facilitated the formation of a galvanic cell between two dissimilar metals that contributed to the degradation of the secondary confinement barrier of the storage system. In another instance, water contributed to an accelerated aging process of concrete structures of the spent fuel storage system. Water entered cracks and crevices around the anchor bolt blockout holes in the concrete structure, and when subjected to freezing temperatures, generated mechanical forces that produced cracks in the concrete. These cracks provided additional and larger pathways for water to enter the interior of the concrete which resulted in larger cracks from subsequent freezing temperatures and promoted efflorescence. If remedial actions had not been taken, this accelerated aging process could have inhibited the ability of the concrete structure to perform its design function of protecting the canister system containing the

radioactive material, as well as protecting personnel from ionizing radiation, during normal and accident conditions.

The effects of weathering and environmental moisture may lead to degradation of structures, systems, and components. Several phenomena are discussed in NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," and Table D-1 of NUREG-1927, "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance" (ADAMS Accession Nos. ML101040620, ML003686776, and ML111020115, respectively). Identifying potential moisture entry points, such as cracks, crevices and joints in both vertical and horizontal storage systems, can facilitate the incorporation of gaskets and sealing materials into both the design of and maintenance of spent nuclear fuel storage systems to minimize premature degradation of structures and components important to safety. Adequate drainage of the base mat (i.e., ISFSI pad) may also prove advantageous for the following reasons. First, pooled water may cause premature degradation of the base mat. Second, since humidity and deliquescence have been shown to contribute to stress corrosion cracking in marine environments (see Information Notice IN2012-20, ADAMS Accession Nos. ML12139A440), the combination of pooled water and heat from canisters containing irradiated spent nuclear fuel could produce humid conditions within the storage system.

These examples show the importance of periodically monitoring the physical condition of a spent nuclear fuel storage system. By obtaining baseline measurements and performing periodic evaluations, accelerated degradation can be detected before the structures and components of a storage system are unable to perform their intended function, and corrective actions can be implemented. Such information may prove useful in assessing aging management in license renewal applications.

CONTACT

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or to the appropriate Spent Fuel Storage and Transportation (SFST) project manager.

/RA/

Mark Lombard, Director
Division of Spent Fuel Storage
and Transportation
Office of Nuclear Material Safety
and Safeguards

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under NRC Library.

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bcc: [illegible] E. Eslinger
[illegible]

72-1005
PDR

October 15, 1990

Ms. M. L. Fisher, Branch Chief
[illegible] Radioactive Waste Branch
Idaho Operations Office - DOE
785 DOE Place
Idaho Falls, ID 83402

EXPERIENCES IN TRANSFER OF CANISTERS FROM THE TN-24P CASK TO THE VSC-17 CONCRETE CASK - [illegible]

Dear Ms. Fisher:

The following will serve to document the operational experiences encountered in the transfer of consolidated fuel canisters from the TN-24P metal spent fuel storage cask to the VSC-17 concrete spent fuel storage cask. This work was part of the activities leading to the testing of the VSC-17 cask in accordance with Cooperative Agreement DE-FC06-[illegible] and the terms and conditions of that Agreement.

The essence of this letter shall also become part of the data report provided at the conclusion of the test activities for the VSC-17 and, therefore, a part of the final report documentation related to the DOE-OCRWM effort on small concrete cask testing.

Preparatory to fuel transfer, the casks were staged in accordance with procedures in place. A problem developed with placement of the TN-24P in the predetermined location because a protrusion on the [illegible] could not be negotiated with the cask transporter as projected. Accordingly, delays were encountered while the project had the Hot Shop floor reanalyzed for placement of the TN-24P cask 12 in. to the south of grade beam 134 centerline instead of directly on beam centerline as planned.

Other activities preparatory to fuel transfer went according to projections and [illegible], placement of plastics, and so forth. The transfer operation was [illegible] by mid-day, September [illegible]. The transfer operation was initiated with the intent to complete the operation including replacement of lids for a stable configuration using overtime hours as required.

Jim — Please note that the experiences noted are with the consolidated canisters which are dimensionally a tighter fit than with normal spent fuel assemblies. D. L. Schmitt

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October 8, 1990
RCS-99-92
Page 2

The assemblies (three of four tried) could not be removed from the TN-24P with a lift equivalent to that which had been used several years ago to place them in the cask, and subsequently retract them and move them around for a better thermal setup (the lift fixture limit was established at [illegible] lbs corresponding to that used in the previous work. The information about retracting and moving of canisters was established from conversation with project personnel from that time and verified through PNL personnel). The net force applied to lift the assemblies was not large since canister weight is nominally about 2350 lbs; nevertheless, the canisters had [illegible] thermally, twisting, bowing, corrosion, or other. The activity was stopped while project personnel examined engineering files to evaluate lift fixture and canister lifting lug capacities. Based on that review, a decision was approved by the project, the facility manager, and a safety representative to revise procedures to increase net force to canister lift to [illegible] lbs. Operations resumed late in the day with instructions to record breakaway forces, constant lift force and so forth since it was recognized that we would be acquiring previously unencountered data. Subsequent operations that evening resulted in partial lifts of all assemblies needed for testing the VSC-17 at which time operations were stopped because of lateness and the need to have time to stop in a stable configuration (lids on casks, etc.).

Activities resumed on October 1, 1990, with further review of the lift fixture. Evaluation determined the need for a new load test in conformance with the DOE Hoisting and Rigging Manual. Discussion with Westinghouse Electric Company, fabricators of the fixture, confirmed that the fixture stress analysis was based on a 4000-lb design load. Subsequently, the fixture was load tested to 6000 lbs in conformance with the requirement of 150% load test such that a 4000-lb lift could be performed.

Discussions were held with PNL personnel to identify alternate canisters to be used in the VSC-17 in case canisters could not be removed from the TN-24P cask.

Transfer operations resumed October 4, 1990. One of the canisters, canister number 18, could not be removed from the TN-24P cask. The canister was raised approximately [illegible] in. at which time the limiting force of 4000 lbs was reached. This assembly was lowered back into the TN-24P cask and an alternate assembly was selected. The "breakaway" forces required to remove the assemblies are provided in attached Table 1. Video tapes were made and observations recorded such as scraping on all four sides of canister 18 during the removal attempt.

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[illegible]

Page 2

The [illegible] requirements for extricating these fuel canisters from the TN-24P [illegible] The canisters had apparently taken [illegible] probably thermally induced although possibly including other [illegible] twisting or other [illegible] The most probable effect was [illegible]

[illegible] a problem within the present budget constraint or at the cost of the major focus, testing of the VSC-17. It should be clear, nevertheless, that the experience encountered should receive future focus since the inability to extract at least one of the assemblies with existing equipment is [illegible] The experience we have encountered [illegible] has direct impact on future [illegible] handling equipment [illegible]

[illegible] problem [illegible] Hot Shop [illegible]

Sincerely,

[illegible] Manager
Cask [illegible] and Testing [illegible]

Hot Shop
Air Trans.

[illegible] from the TN-24P [illegible] years ago to [illegible] move them around [illegible] the canisters had [illegible] Based on that [illegible]

[illegible] The canister was [illegible] force of 4000 lbs was [illegible]

TABLE 1. TN-24P CANISTER PULLING FORCE

TN-24P Location	Canister Number	Breakaway Force (lbs)
A1	12	3180
A2	21	3020
A4	16	3020
A5	24	3250
A6	8	3002
B1	10	3010
B2	19	3331
B3	18	3110*
B5	17	3350
B6	9	3030
C1	11	3000
C2	15	3038
C3	6	3010
C6	2	3090
D1	13	3290
D3	4	3050**
D5	23	3360
D6	3	3060

18

* After fuel canister was lifted approximately 3 inches, pull forces started to increase until the maximum force of 4000 was reached.

** Fuel canister number 4 was used as a spare for canister 18.

INEL
Idaho National Engineering Laboratory

bcc: J. G. Olneen, MS 9109
L. E. Eslinger, MS 9109
W. A. Franz, MS 9206
L. P. Leach, MS 3940
G. R. Trohkinoinen, MS 3920
R. C. Schmitt File
Central Files

November 21, 1990

Ms. M. W. Fisher, Branch Chief
Civilian Radioactive Waste Branch
Idaho Operations Office - DOE
785 DOE Place
Idaho Falls, ID 83402

EVALUATION OF OPTIONS, ASSOCIATED COSTS, AND RECOMMENDATIONS FOR:
(A) THE "STUCK" FUEL CANISTER IN THE TN-24P CASK, (B) RELATED CASK MOVE OPERATIONS, AND (C) RESOLUTION OF VSC-17 PRESSURE BEHAVIOR - RCS-111-90

Dear Ms. Fisher:

The following provides an assessment of options, costs, and recommendations regarding the "stuck" fuel assembly, related cask operations, and the VSC-17 cask pressure behavior.

Review of Stuck Fuel Assembly Issue

Please recall that during transfer of fuel from the TN-24P cask to the VSC-17 cask, consolidated canister number 18 was withdrawn about 12 inches from the TN-24P basket at which time the 4,000 lb lifting limit of the lifting fixture and the canister lifting lugs was reached. Subsequently, the canister was returned to it's seated position and an alternative canister was selected for testing in the VSC-17. The lifting limit had been increased from the 3,000 lbs of previous operations to 4,000 lbs to overcome the "breakaway" force that was unexpectedly encountered in the removal of canisters after their several years of storage in the TN-24P. (The increase to a 4,000 lb lift limit was derived based on the initial design and load test of the lifting fixture of 6,000 lbs and the load test of the canister at 4,000 lbs.)

The lift operation was video taped and scrape marks were observed on all four sides of canister number 18. Whereas, thermal expansion of the canister is the most probable cause, bowing, twisting or other mechanisms cannot be eliminated as possibilities; we presently have little capability to determine the root cause because accessing the assembly or the basket is not feasible with fuel in the cask. For the other six canisters in the TN-24P, it is possible, although not probable, that additional canisters

canisters?

EG&G Idaho, Inc. P.O. Box 1625 Idaho Falls, ID 83415

This activity currently ongoing as of 11/9/92.

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The activity currently ongoing as of [illegible]

may be unremovable. It is also possible that canister number 18 is no longer stuck because of thermal unloading of the basket following the removal and placement in the VSC-17 cask of 17 fuel canisters.

Costs for Additional Analyses and Operations for the Stuck Assembly

One possibility for removing canister number 18 from the TN-24P cask is simply to apply more lift. The major risks are that the assembly might become stuck in a partially withdrawn position or that canister damage might be incurred. To assess applying more lift involves reanalyzing the design of the canister and obtaining one-time exemptions to exceed DOE Hoisting and Rigging Manual restrictions. The CCTP has received an estimate of [redacted] from Engineering to analyze the lifting fixture and canister design and determine methodology to remove the stuck fuel canister. The estimate includes conservatism based on the uncertainties involved and, unfortunately, there is no assurance of a useful result. The CCTP has independently prepared an assessment of the task and believe that the estimate from Engineering is about a factor of two too large. Accordingly, [redacted] is expected to be more realistic. Costs could be less if a "dead end" is encountered early.

A second part of the problem would involve what to do with the canister assuming success in extracting it from the TN-24P. One solution would be to place it in the VSC-17 where the canister ports are dimensionally relaxed (more clearance). This requires a number of operations: removal of the VSC-17 and TN-24P lids, removal of a canister from the VSC-17 and placement in the TN-24P, removal of canister 18 from the TN-24P and placement in the VSC-17, replacement of lids, pumpdown and backfill of both casks, and etc. Additionally, partial lifts of the other six canisters in the TN-24P would be completed to assure removability. Our estimate for these operations is about [redacted].

Issue of Permanent Lids on TN-24P

Another issue is the Hot Shop inquiry (or request) that the permanent lids be installed on the TN-24P when it is returned to the pad this time. Our estimate to place the permanent lids on the cask is [redacted] which includes the fact that lid placement fixtures need to be designed and fabricated, special shielding and/or remote handling capabilities are required, and numerous procedural changes are necessary (please recall that the permanent lids include a thin inner lid where shielding is inadequate for personnel to work on bolting).

Issue of Unusual Pressure Behavior in the VSC-17 Cask

The final issue is the unusual pressure behavior observed in the VSC-17 cask, principally during the initial startup and early test runs. The issue includes your request to evaluate the possibility of laboratory scale testing to research the cause of the pressure behavior. It is the project's

VSC-17 Pressure Beha
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Your consideration of the
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cc: D. Hixon, DOE-ID, MS
DOE Project Engineer

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November 28, 1990
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Page 4

VSC-17 Pressure Behavior. The project recommends no action regarding laboratory testing for evaluating the VSC-17 cask pressure behavior unless the evaluation of the data obtained during testing of the VSC-17 cask proves unacceptable for identifying the observed behavior. Since we are still obtaining data as of November 28, 1990, a completion schedule for the evaluation is not projected at this time. The evaluation includes the efforts of PNL and others not in the project.

Your consideration of the above recommendations and your directions accordingly are requested.

Sincerely,

R. C. Schmitt

R. C. Schmitt, Manager
Cask Transport and Testing Project

tcm

cc: D. Hixon, DOE-ID, MS 1110
DOE Project Engineer

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