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EXAMINATION OF SPENT PWR FUEL RODS AFTER 15 YEARS IN DRY STORAGE

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Virginia Power Surry Nuclear Station Pressurized Water Reactor (PWR) fuel was stored in a dry inert atmosphere Castor V/21 cask at the Idaho National Environmental and Engineering Laboratory (INEEL) for 15 years at peak cladding temperatures decreasing from about 350 to 150°C. Prior to the storage, the loaded cask was subjected to extensive thermal benchmark tests. The cask was opened to examine the fuel for degradation and to determine if it was suitable for extended storage. No rod breaches had occurred and no visible degradation or crud/oxide spallation were observed.

Twelve rods were removed from the center of the T11 assembly and shipped from INEEL to the Argonne-West HFEF for profilometric scans. Four of these rods were punctured to determine the fission gas release from the fuel matrix and internal pressure in the rods. Three of the four rods were cut into five segments each, then shipped to the Argonne-East AGHCF for detailed examination. The test plan calls for metallographic examination of six samples from two of the rods, microhardness and hydrogen content measurements at or near the six metallographic sample locations, tensile testing of six samples from the two rods, and thermal creep testing of eight samples from the two rods to determine the extent of residual creep life. The results from the profilometry (12 rods), gas release measurements (4 rods), metallographic examinations (2 samples from 1 rod), and microhardness and hydrogen content characterization (2 samples from 1 rod) are reported here. The tensile and creep studies are just starting and will be reported at a later date, along with the additional characterization work to be performed.

Although only limited prestorage characterization is available, a number of preliminary conclusions can be drawn based on comparison with characterization of Florida Power Turkey Point rods of a similar vintage. Based on this comparison, it appears that little or no cladding thermal creep and fission gas release from the fuel pellets occurred during the thermal benchmark tests or storage. Measurements of the cladding outer-diameter, oxide thickness and wall thickness are in the expected range for cladding of the Surry exposure. The measured hydrogen content is consistent with the oxide thickness. The volume of hydrides varies azimuthally around the cladding, but there is little variation across the thickness, of the cladding. It is most significant that all of the hydrides appear to have retained the circumferential orientation typical of prestorage PWR fuel rods.

KEYWORDS: Spent Fuel, Dry Storage, Fuel Examination

I. INTRODUCTION

Some of the original licenses issued by the United States Nuclear Regulatory Commission (NRC) for 20 years of dry storage of Light Water Reactor (LWR) fuel are coming up for renewal shortly. Due to the delay in opening a deep geological repository, consideration is being given to long-term dry storage of perhaps up to 100 years. There are material-related issues with respect to the potential long-term behavior of the dry cask storage system (cask, basket, seals, fuel, etc). The principal fuel issue is whether the cladding can remain a viable barrier to fission-product release and, if not, to what extent the fuel itself may provide a degree of fission-product retention in the cask storage environment. A corollary issue is whether the fuel rods will retain sufficient integrity after 20+ years of dry storage to be safely transferred to a final repository.

A report recently completed for the Electric Power Research Institute (EPRI), "Data Needs for Long-Term Dry Storage of LWR Fuel"[1], identified a number of specific questions about fuel rod behavior. These include:

1. Has there been a change in the mechanical properties, especially the ductility, of the cladding?
2. What is the extent of cladding creep at storage conditions?
3. Has there been a detrimental hydrogen pickup or hydride reorientation in the cladding?
4. How much cladding annealing has occurred?
5. Has gas release from the fuel to the plenum been enhanced?
6. Has volatile fission-product release been enhanced?

Questions 1 through 4 are all germane to maintaining cladding integrity during storage, subsequent handling during transfers, and off-normal events that may occur in the cask's life cycle. Questions 5 and 6 are germane to environmental effects within the cask and the impacts these effects will have on subsequent safe handling. The cladding mechanical properties are most relevant in addressing the issue of integrity. Environmental effects will have greater relevancy and importance if the mechanical properties have degraded significantly or if degradation phenomena have been enhanced because of the environment.

In the mid-1980s, the U.S. Department of Energy (DOE) procured a Castor V/21 dry-storage cask for testing at the Idaho National Environmental and Engineering Laboratory (INEEL). The primary purpose of the tests was to benchmark thermal and radiological codes and to determine the thermal and radiological characteristics of the cask. The cask was loaded with as-irradiated assemblies from the Surry Nuclear Station and then tested in a series of configurations using a variety of cover gases. The tests were not intended to examine fundamental fuel behavior. Therefore, prior to the tests, the fuel had undergone only minimal characterization: visual examination of the outside of the assemblies and ultrasonic examination to ensure no breached rods would be included. During the tests, the temperature was monitored and the cover gas was periodically analyzed to determine if any leaking rods had developed. No leaks were found. The details of these tests have been reported in a number of documents [2-4]. Subsequently, the cask sat on the storage pad at the INEEL for ≈15 years with the fuel in an inert atmosphere (helium).

The NRC, EPRI, and DOE are interested in determining if relevant information can be gained from examination and testing of the fuel rods that have undergone typical long-term storage. These organizations arranged to open the Castor cask and conduct both a nondestructive and destructive examination of the fuel in an effort to determine any degradation that may have occurred during the storage period.

Twelve rods were removed from one assembly (T11 from the Surry-2 Reactor) for further characterization and testing. This report describes the initial fuel and cladding characterization, as well as the implications of these results for long-term dry cask storage.

II. TEST CONDITIONS AND MATERIALS

A. STORAGE CONDITIONS

The behavior of the spent fuel is governed by its time at temperature and the storage atmosphere. These rods underwent two different types of storage, which contribute to the overall fuel and cladding condition of the rods examined. Initially, the rods were part of a performance-testing program during which they were stored for short periods of time under a variety of atmospheres and orientations, each resulting in a different maximum temperature and temperature profile. At the completion of this testing, the rods were stored for an extended period of time in a He/<1%air atmosphere, during which the temperature continued to decrease.

Table I. Time/Temperature History of Assembly T11 from Castor-V/21 Cask

Configuration	Cover gas	Peak Cladding T,°C	≈Duration, hours
Vertical	Air	Reduced temp	200
Vertical	He	344	119
Vertical	N_2	359	43
Vertical	Vacuum	415	72
Horizontal	He	357	93
Horizontal	N_2	398	72
Vertical	70% He/30% Air	396 → 352	2880
Vertical	He/<1% Air	344 → 155	1.3×10^5

The Castor cask was monitored with thermocouple probes during the thermal benchmark-testing phase to determine the temperature profiles as functions of elevation and radial position with the cask. In an upright orientation, the hottest fuel occurred in the center two assemblies [2]. In addition, there was a thermal gradient along the fuel rods. The hottest elevation was ≈2.5 m above the rod bottom, and dropped off significantly along the bottom 500 mm [4].

Prior to complete loading, the fuel rods spent about 200 h in air at a significantly reduced temperature. During the following cask performance tests [2-4], the fuel rods were in He, N_2, and vacuum atmospheres in horizontal and vertical configurations. The peak cladding temperature adjusted for the position of the assembly in the cask varied in each configuration [3] as indicated in Table I. It should be noted that the estimated cladding temperatures during the initial 2160-h test period in the Castor-V/21 cask ranged from ~25 to 100°C above a PWR cladding temperature during normal reactor operation. At the completion of the cask performance tests, the fuel spent approximately 2880 h vertically in a 70% He–30% air atmosphere [3]. Since that time in March 1986, the fuel has been stored vertically in a He/< 1% air atmosphere. While the temperature was not monitored during this storage period, the peak cladding temperature decreased due to decreasing decay heat. Due to the short duration between the end of the performance testing and the start of the extended dry storage, the initial temperature for the extended storage should be nearly the same as that measured during the performance testing in the same configuration with the same cover gas. (If one compares the measured temperatures in the vertical configuration with He and N_2 and takes a linear interpolation, one can see that the 1% air in the extended cask storage atmosphere has an insignificant effect on the thermal conductivity of the He and hence the temperature.) While the cask lid was removed, a thermocouple was inserted ≈650 mm into the assembly and measured a temperature of ≈155°C after this storage period, with the cask lid still off. Because of enhanced convection with the cask lid off, this temperature is somewhat, but not significantly, lower than that expected in the closed cask.

Cover gas samples were taken periodically throughout the performance-testing period and storage duration to look for cask leaks and fuel rod breaches [3]. In all cases, the intended atmospheres indicated in Table I were maintained. The moisture level in the cask was <0.03 vol. % in all cases.

B. FUEL DESCRIPTION

The Castor cask contained twenty-one 15 x 15 Westinghouse fuel assemblies [3] that had been irradiated in the Surry Reactor. Assembly T11, which had been stored in the hottest part of the cask, was chosen for evaluation. The UO_2 fuel pellets had an initial enrichment of 3.11% and nominal density of 95% theoretical. The slightly cold-worked and partially annealed cladding had a nominal original outside diameter of 10.71 mm, with a wall thickness of 0.62 mm. The rods were pressurized with He to 2.86 MPa [4]. The assembly was irradiated for three cycles to achieve a burnup of 35.7 GWd/MTU. The assembly-averaged fast (E>1 MeV) neutron fluence is calculated to be 6.38×10^{25} n/m^2 [5]. It was discharged in November 1981 and was in water storage until transported to INEEL and loaded into the Castor cask in July 1985, with a decay heat of 1.1 kW. Since the details of the thermal profile were not fine enough to determine rod-to-rod temperature differences, the center 12 rods in the assembly were chosen for examination because of their ease of extraction from the assembly.

C. CHARACTERIZATION OF SURRY FUEL PRIOR TO STORAGE

Ideally, one would perform extensive characterization on the fuel rods prior to storage in order to have a reference point to determine changes that might occur during dry cask storage. Nondestructive profilometry on the actual test rods and gas analysis, ceramography, metallography, hydride analysis, and mechanical property testing on adjacent sibling rods would be conducted. As the original purpose of these tests was to determine the thermal characteristics of the cask systems and not the long-term performance of the rods, only limited characterization was conducted on the Surry rods or assemblies prior to the initial cask performance tests [4].

All the rods had been ultrasonically examined for leaks in the Surry basin; no leaks were found. Full-length black and white videos of assembly T11 and seven other assemblies, and color still pictures of two assemblies (V05 and V27), were taken of all four sides of each assembly. The videos provided little information on the integrity of the rods. The photos showed an orange/reddish crud, which was later analyzed to be Fe_2O_3, on assembly V05. This is somewhat unusual since prior analysis of crud from Surry rods indicated a very thin whitish-gray Fe_3O_4 hematite, which is typical of PWR rods [6]. Videos and color stills of the same assemblies were taken after the Castor-V/21 performance tests, just prior to the commencement of long-term storage.

During an assembly consolidation exercise, linear profilometry at 90° orientations was performed on selected rods from 36 similar Surry assemblies. The maximum, minimum, and average diameters in each direction for each of the 36 assemblies were reported [6]. The difference between the maximum and minimum for any one Surry assembly ranged between 0.03 and 0.06 mm (0.3 and 0.6% of the as-fabricated diameter). The average rod diameter for all the assemblies varied by as much as 0.06 mm or 0.6%. Therefore, the minimum average outward creep that occurred during dry storage that could be determined by comparison of a particular post-storage rod diameter trace with an average prestorage trace from the collection of rods would be about 0.8%.

Some of the 15 x 15 Westinghouse rods irradiated in the Turkey Point reactor, which were manufactured to similar specifications as the Surry rods, were irradiated under similar conditions as the Surry rods for 3 cycles, but to a lower burnup of ≈26-28 GWd/MTU. These rods were extensively characterized [7,8]. Since phenomena such as cladding creepdown and radiation-induced hardening tend to saturate in the first few cycles, the characteristics of the Turkey Point rods are expected to be similar to those of the Surry rods. The in-reactor gas release of the Surry rods might be slightly higher due to the increased burnup; but it is still expected to be ≤1% and well within the scatter of gas-release data found for the general population of PWR rods in this burnup range. Cladding oxide thickness and hydrogen uptake should be greater for the Surry rods because of their longer residency time (≈1190 hours vs. ≈850 hours for Turkey Point). Attributes of interest include oxide layer thickness, hydrogen content, hydride distribution and orientation, and hardness. At the modest burnup levels of these rods, 26-36 GWd/MTU, burnup is not a strong consideration in fuel rod behavior. Likewise, because of the lack of post-irradiation destructive examination data on exact sibling Surry rods, comparisons of post-storage data obtained from the examinations will be made against open literature data for PWR fuel rods. The post-irradiation data on the Turkey Point rods [6,7] will also be used for comparative purposes.

Five rods from two 3-cycle Turkey Point assemblies were destructively analyzed [8]. All the rods have the same isotopic distribution of fission gas and a fission gas release (fgr) of ≈0.22%. The hydrogen content in the cladding ranges from 40±10 wppm H_2 at 610 mm from the rod bottom to 90±5 wppm H at the 3050-mm elevation. The hydrides are circumferentially oriented. The as-fabricated cladding thickness is nominally 0.617 mm with a low of 0.597 mm and a high of 0.643 mm. The oxide thickness varies both azimuthally and axially, and there could be as much difference in oxide thickness as a factor of two between two rods at the same axial location.

III. EXAMINATIONS

Twelve rods were transferred from the INEEL Test Area North (TAN) facility to the Argonne-West Hot Fuel Examination Facility (HFEF) where profilometry was conducted on all the rods to determine if any in-storage creep had occurred. All of the rods exhibited similar cladding outer diameter profiles. Four rods with marginally higher diameters were chosen for fission gas release, internal pressure measurements, and void volume determination. These data are needed to determine the cladding stress during storage, which is the driving force for creep. The three rods with the highest internal pressure were marked and sectioned into five lengths (≈33 inches) per rod for transport to the Alpha Gamma Hot Cell Facility (AGHCF) at Argonne-East.

The rod segments were sectioned to obtain specimens for creep testing and metallographic samples for oxide thickness, cladding thickness, hardness, and hydride analysis. Samples for tensile testing and hydrogen determination were also prepared. Limited ceramographic examination of the fuel was conducted. As the rods were unbreached, no change in the structure of the irradiated fuel was expected at the storage temperatures.

A. PROFILOMETRY

Linear profilometry traces of each rod were made at 0, 45, 90, and 135 degrees relative orientation at ≈2.5-mm intervals along the length, starting at ≈330-mm from the top of the rod. The cladding outer diameters were measured to an accuracy of 2.5 x 10^{-3} mm. The cladding outer-diameter profile for rod H9 (Fig. 1) was generated by averaging the four linear traces. While not apparent in the individual traces, the averaged profile reveals discernible, albeit small, dips associated with the assembly grid spacers. Similar profiles were obtained for all the rods. The average cladding diameter decrease for the 12 rods at a location of 1 m from the bottom of the rod is ≈0.06±0.01 mm relative to the nominal as-fabricated diameter of 10.71 mm. This creep-down (≈0.6%) is typical for PWR rods with a burnup of ≈36 GWd/MTU [7].

Some of the profiles show an increase in diameter on the upper half of the rod relative to the lower half of the rod. One would normally expect the profile of the rod in an as-irradiated state to be reasonably flat in the middle 2.5 m where the fast fluence is relatively uniform. The increase in the upper half of the rods after storage may be due to the increased temperature in this location in the storage cask or, more probably, an increased oxide layer thickness in this area of the rod.

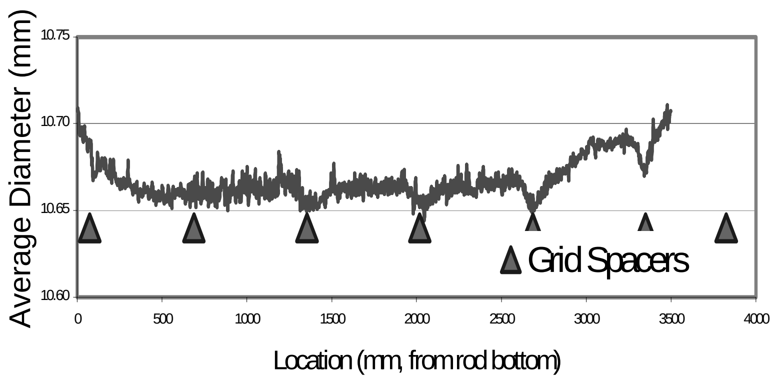


Fig. 1. Averaged outer diameter profile for Rod H9 (from Surry-2 Assembly T11) after extensive thermal benchmark testing (≈350 to 415°C) and ≈15 years of dry cask storage in He (≈350 → ≈150°C). The nominal as-fabricated cladding outer diameter is 10.71 mm.

Oxide layer thickness has been measured at about the midplane and ≈20 inches above the midplane for one rod. Assuming a linear extrapolation from these two points, a correction can be made to the diameter profiles. The upturn of the profile on the upper end of the rod is then decreased progressively as more oxide is subtracted from the diameter. Doing this calculation for the H9 Surry rod gives a relatively uniform creepdown of 0.6% for the middle 2.5 m of the fuel column. This correction will be recalculated for rods H9 and G6 after the remaining four cladding characterizations are performed.

B. FISSION GAS ANALYSIS AND VOID VOLUME DETERMINATION

Cladding creep during dry cask storage is driven by the stress caused by the rod internal gas pressure. This pressure is due primarily to the initial He fill gas and, to a lesser extent, the fission gas released from the fuel into the rod void volume. The void volume, internal rod pressure, gas composition, and fission gas isotopic composition were measured for four of the rods to determine the internal rod stress and to estimate the extent of fission gas release during the storage period.

The measured internal gas pressures range from 3.43 to 3.61 MPa at 27°C, which is approximately 0.7 MPa higher than the as-fabricated gas pressure (Table II). The void volumes range from 19.53 to 20.39 x 10^{-6} m^3, which is typical of this vintage of rods. The decrease in void volume with burnup is caused by a combination of cladding creepdown and fuel swelling. The relative errors of the void volumes and of the gas pressures are less than 3%. The internal gas composition in all four rods is essentially the same: 96-98% He fill gas with a small amount of O_2 and N_2. The average Xe/Kr ratio is ≈10 compared to a ratio of ≈9 in Turkey Point fuel, although there was only 0.5% fission gas in the Turkey Point void gas. The post-storage isotopic composition of the Xe and Kr are, within experimental error, the same as in the post-irradiation ratios for the Turkey Point fuel. The fission gas release values range from 0.4 to 1.1%, which is larger than the post-irradiation fission gas release of 0.22% measured in the Turkey Point rods, but well within the range reported in the literature for rods of this type and burnup.

C. FUEL ROD SELECTION AND SAMPLING LOCATIONS

The rationale for the selection of the 12 Surry-2 T11 PWR rods has been discussed in the Introduction. The Activity Plan calls for: selecting four of the 12 rods with the largest diameters for gas analysis; selecting three of these four rods with the highest internal gas pressure for segmentation and shipment to ANL-E; and selecting two of these three rods with the largest diameters and highest gas pressures for characterization and determination of tensile and creep properties. In spirit, this logic was designed to focus on rods that may have experienced some thermal creep during post-reactor performance testing and storage. In practice, the differences found in the profilometry proved to be insignificant relative to experimental uncertainty.

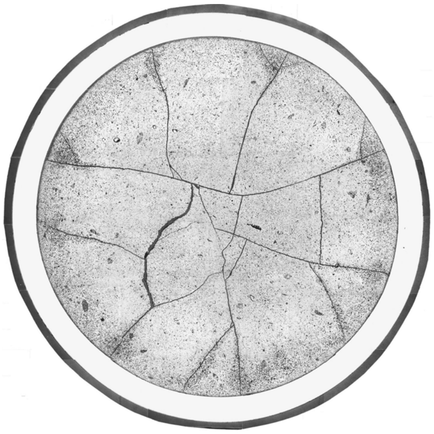
D. METALLOGRAPHY AND HYDROGEN ANALYSIS

1. Fuel Pellet Condition

Due to the lower temperature of the fuel during storage compared to in-reactor temperatures, and no indication of cladding breach, changes in the condition of the fuel pellets themselves were not expected. Transverse ceramographic samples were taken from Rod H9 at the rod midplane and ≈510 mm above the midplane. Cross-sectional mosaics (see Fig. 3) depict a pellet cracked into 10-25 pieces, which is prototypic of this fuel at this burnup.

Table II. Rod Volume, Internal Pressure (at 27°C) and Fission Gas Release (fgr)

Rod	Void Vol, m^3 x 10^6	Internal Gas Pressure, MPa	fgr, %	Fission Gas Contribution to Void Pressure, %	Fission Gas Partial Pressure, MPa
T11-H9	19.76	3.61	1.08	3.61	0.129
T11-G6	19.53	3.51	0.39	1.36	0.047
T11-H7	20.29	3.43	0.88	3.01	0.102
T11-G9	20.02	3.44	0.49	1.65	0.056
Turkey Point	22±1	3.5±0.1	0.22±0.01	0.50	0.017



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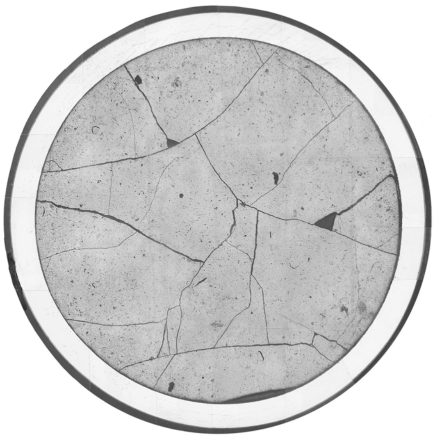
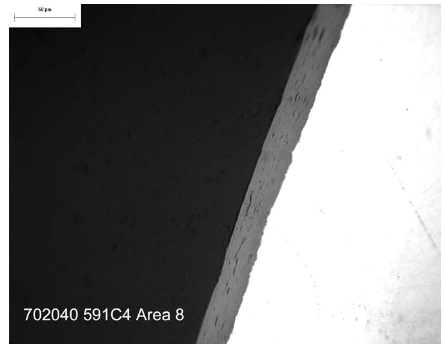


B

Fig. 3. Cross-sectional mosaics of Rod H9: A) midplane, B) 510 mm above midplane.

2. Oxide Layers, Fuel-Cladding Gap, and Cladding Thickness

The cladding outer-surface oxide layer thickness was measured from the photomicrographs at eight azimuthal locations around the cladding, covering about 45% of the cladding circumference. At the fuel column axial midplane, the thickness varies between 20 and 28 μm with an average of 24 μm. At the higher elevation, consistent with a higher in-reactor temperature, the oxide layer ranges from 25 to 42 μm thick with an average value of 33 μm (see Fig. 4). This is about 3 times as large as observed on the Turkey Point "D" rods at the same elevation, but well within the range of oxide thickness of 5-40 μm for this burnup fuel [8]. Some fine circumferentially oriented microvoids were observed in the oxide layer. There was limited spallation of the oxide, apparently due to the linkage of the microvoids. The oxide morphology is consistent with the observations on the Turkey Point rods. A fuel-cladding gap was observed around the inner cladding circumference, probably formed during cooldown. Small (<10 μm), discontinuous oxide layers were observed on the cladding inner surface, but there was no evidence of fuel-cladding chemical interaction. Once again, this is typical of the behavior of rods at this nominal burnup.



A

B

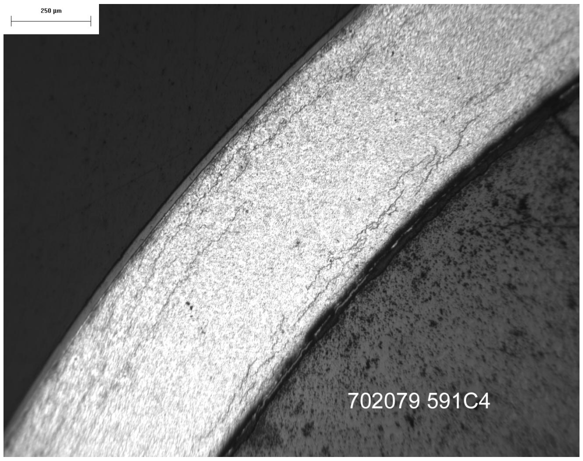
Fig. 4. Cladding outer-surface oxide layer for Surry-2 rod H9 from assembly T11 at: A) axial midplane, B) ≈510 mm above the midplane.

3. Hydrogen Content and Orientation

The cladding was etched to elucidate the hydrides. Eight azimuthal regions were imaged at each axial location. The radial location and the density of hydrides vary both axially and azimuthally (See Fig. 5). In all cases, the hydrides were circumferentially oriented. Eight determinations of the hydrogen content were made at each elevation using Leco fusion extraction analysis. At the midplane, there is 250 ± 40 wppm of H and ≈510 mm above the midplane, there is 300 ± 25 wppm H (error equals one standard deviation of measurements). The 20% higher content at the higher elevation is about 14% lower than would be expected, based on 37% more oxide growth at the higher elevation. Using a Pillings-Bedworth Ratio of 1.75, appropriate for oxide formed in-reactor [9], consistency of oxide growth with hydrogen content would require a hydrogen uptake of 24% at the midplane and 21% at the higher elevation.

E. HARDNESS MEASUREMENTS

The cladding hardness was measured across the radius of the cladding at both axial locations using a 200 g Vickers indenter. The measurements are performed at four azimuthal locations (≈90° apart) at each axial location. There is no discernable variation across the cladding with a hardness of 239 ± 5 at the axial midplane and 236 ± 10 at ≈510 mm above the midplane. There are no comparative hardness data on as-irradiated Surry cladding, but some data are available for Turkey point cladding. These two data sets are compared in Section IV.E.



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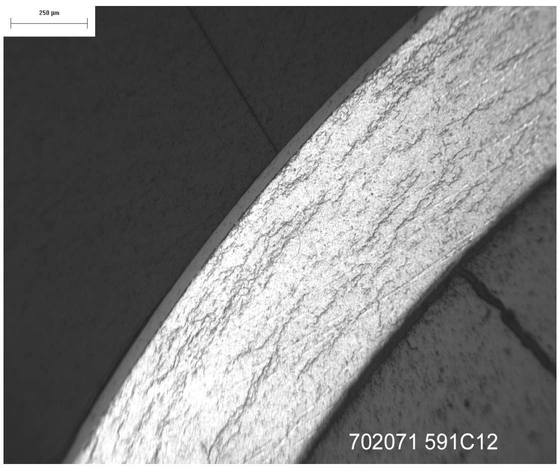

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Fig. 5. Hydride structure in cladding of Rod H9 Surry-2 Assembly T11 at two elevations: A) midplane, B) ≈510 mm above midplane.

IV. IMPLICATIONS FOR DRY STORAGE

The full examination of the Surry-2 PWR fuel and cladding from the Castor-V/21 Cask has not yet been completed. Metallography at four additional locations (1 more from Rod H9 and 3 from Rod G6) will be performed, along with microhardness and hydrogen determination at the four additional metallographic locations. Tensile (6 samples from H9 and G6) and thermal creep testing (8 samples from H9 and G6) have yet to be completed. But even before the completion of these tests and without the additional characterization results, one can draw a number of preliminary conclusions from the current data regarding the potential for further dry storage. Visually, the rods appear to be in excellent condition.

A. CREEP

Ideally, we would like to compare profiles before and after storage to determine the amount of creep that occurred. This would eliminate correction factors such as oxide growth, crud variability, and systematic measurement error. In this particular situation, there are no pre-storage profiles on the actual rods measured after storage, so profiles from rods that have similar properties and irradiation histories might be considered for comparative purposes. There is profilometry data from sister Surry rods taken during rod consolidation, and Turkey Point rods measured at Battelle Memorial Institute (BMI) prior to shipment and at INEEL during rod consolidation. Using these diameter measurements as a baseline entails a considerable degree of uncertainty.

When the measurements taken on the same rods at BMI and INEEL are compared, there is a variation of ±0.2%. This may be a result of different measurement machines or accuracies. Vinjamuri [6] gives the maximum, minimum, and average diameters for the Surry and Turkey Point rods used in the consolidation project. The variations are given in Table III below.

Table III. Variation of Rod Diameters within an Assembly and from Assembly to Assembly.

Reactor	Assembly to assembly variation in averaged rod diameter	Rod to rod variation in average diameter within an assembly
Surry	0.62%	0.62%
Turkey Point	0.31%	0.46%

Since we do not know which rod or assembly to baseline with and we are measuring in a different facility, any profile we chose for comparative basis may be as much at 0.8 to 0.9% different from the actual profile before storage for the rod we measure after storage. In other words, there must be greater than 0.8% creep before we can say with reasonable certainty that any creep occurred during performance testing and storage.

An estimate of the creep can be obtained by comparing the diameters after storage with the minimum diameter that may have been prior to storage. Twenty Turkey Point rods were profilometered after irradiation and water storage (transported in a water-filled cask) prior to any dry storage. Average profile plots are available [7] with error limit indicative of the rod ovality. The hot region in the vertical storage cask with a He atmosphere was between 2 and 3 m from the bottom of the fuel. This elevation is where one would expect the maximum creep to occur during storage. Based on eyeball estimates from the plots, the maximum, minimum, and average diameter for each rod in the 2-3 m elevation was tabulated. The global (worst case for collection of rods) and average values over the set of rods in these tables is summarized in Table IV below.

Table IV. Global and Average Profilometry Values at 2-3 m Elevation

	Maximum Diameter mm	Average Diameter mm	Minimum Diameter mm
Turkey Point global	10.765	NA	10.638
Turkey Point average	10.721	10.678	10.650
Surry global	10.700	NA	10.645
Surry average	10.679	10.670	10.656

The worst case would be to compare the global maximum from the Surry rods with the global minimum from the Turkey Point rods, which is 0.6%. If we compare the average Surry maximum to the average Turkey Point minimum, this drops to 0.3%. Comparison of the averages implies that no creep occurred during performance testing and storage. The uncertainty in the baseline as shown above indicates these values are all essentially zero. Even these upper bounds are probably an overestimation of the creep since the oxide layer thickness has not yet been accounted for in the diameter profiles.

Another way of looking at the creep is to compare the after-storage profiles with the minimum diameter that could have occurred due to in-reactor creep. During irradiation at ≈350°C, the as-fabricated pellet diameter of 9.385 mm (Surry drawings) would have increased to 9.408 due to thermal expansion. If this formed the mandrel for the cladding, the outer cladding diameter (not accounting for oxide growth) would be the pellet diameter plus two times the cladding thickness (0.617mm) or 10.643 mm. Depending on whether this is compared with the global or average maximum Surry diameters (see Table IV), the maximum creep would be between 0.3 and 0.5%. Once again, this does not account for oxide growth, which may lower these estimates by as much as 0.3%. This correction will be made in the final analysis.

These creep estimates are below the current limits of 1% creep strain. Most of this creep probably occurred during either the performance testing or initial storage years when the temperature was the hottest and the stress highest. The current temperature is substantially below that in the initial storage period and will continue to drop during extended storage. Creep calculations to compare with the above estimates will be made after a storage temperature history of the Surry T11 rods is provided. Preliminary thermal creep predictions using the performance testing temperatures in Table 1 and an assumed storage temperature history give <0.1% thermal creep for irradiation-hardened Zircaloy-4 at the Surry fast fluence.

B. FUEL ROD STRESS

Stress on the cladding due to the pressurization of the rod is the driving force for cladding creep. It is almost impossible to partition the amount of fission gas released during in-reactor operation and during dry storage. The release occurs both by a diffusive process driven by the temperature gradient across the fuel and cracking/rehealing during in-reactor startups and shutdowns. In-reactor, the temperature gradient is greater, the temperature is higher, and hence the diffusion coefficient is higher. In all likelihood, most if not all of the release occurred in-reactor. The measured fission gas release in the surry rods was within the range one would expect for the in-reactor release of fission gas. Any further release will be substantially lower due to the lower diffusion constants, and hence the increase in partial pressure due to fission gas release will be less significant, subsequently reducing the stress on the cladding even more.

C. HYDRIDE REORIENTATION

As the cladding is heated during the drying of the cask and initial storage at higher temperatures, much of the hydrogen introduced into the cladding during irradiation goes into solution. As the cladding cools during storage and the solubility limit is exceeded, the hydrogen will precipitate as hydrides. Depending on the stresses in the cladding, these hydrides will usually be circumferential but may be radial. Excessive hydrogen content or hydrides in the radial direction may degrade the mechanical properties of the cladding. Calculations indicate that after years of storage, the hydrogen content is not excessive and there are very few, if any, radial hydrides. The general circumferential hydride structure, seen in Figure 5, supports this conclusion. The level of the hydrogen is as expected from in-reactor hydrogen pick-up, and there was no new source of hydrogen in the inert atmosphere of the cask.

The maximum hoop stress in the rods during 15 y dry storage (see Table 1 for conditions) is estimated to be ≈62 MPa, which is below the threshold at which one would expect reorientation to occur based on quenching experiments. There is some evidence [10] that reorientation could occur below the threshold stress if the cooling rate were slow enough. The maximum cooling rate, which occurred at the start of the storage period, in this test was ≈0.002°C/h. At this extremely slow cooling rate, no reorientation was observed even though the estimated hydrogen content was near the solubility limit. Between 350-400°C, the solubility limit is 120-200 wppm. At 150°C, the solubility limit of hydrogen in Zircaloy is in the range of 20-30 wppm [11]. Even if some reorientation of the precipitates of the remaining hydrogen had occurred, there should be little detrimental effect on the mechanical properties [12].

D. CLADDING ANNEALING

Knoop hardness measurements with a 200-g weight were taken on as-irradiated "B" and "D" type Turkey Point cladding and cladding of those two types annealed at 571°C, 482°C, and 323°C [11]. As with the Surry cladding, no variation in hardness was found across the Turkey Point cladding radius under any of the test conditions. Both as-irradiated cladding samples and all those annealed at 323°C for 2100 h had a Knoop hardness of 270±20 and those annealed at 571°C for 740 h and 482°C for 4656 h had lower hardness of 185±5. The agreement between the as-irradiated and as-annealed-at-323°C hardness results, as well as agreement with the values in the literature [11] suggests that little annealing should take place at 323°C for 2100 h. There is also agreement with the literature that indicates that significant annealing of irradiation damage should occur in a short time above 400°C.

Using a 200-g weight, at least for stainless steel, the Vickers hardness is about 5% lower than the Knoop hardness [13]. Making the 5% correction between hardness scales, the Surry cladding hardness (≈250 Knoop), based on two samples, is slightly lower than the Turkey Point as-irradiated cladding indicating only a slight annealing, if any. If one assumes a linear relationship between hardness and indentation number, the amount of recovery is estimated to be <20%. This recovery may have occurred during the performance testing when the temperature reached as high as 415°C for a short time. At the lower temperatures expected for extended storage (<150°C), no additional recovery is expected.

V. PRELIMINARY CONCLUSIONS

Preliminary evaluation of Surry-2 PWR fuel rods -- with a burnup of 35.7 GWd/MTU -- that were stored for ≈15 years at an initial temperature of ≈350°C (with temperatures reaching as high as 415°C during ≤72 hours of the performance testing) in a Castor-V/21 cask is reported. After visual examinations, the diameters of the rods were measured, the fission gas was analyzed for volume, composition, and isotopic analysis. Three rods were sectioned and shipped to ANL-E. Metallography, microhardness, and hydrogen determination have been completed at two axial locations of Rod H9. The main conclusions that can be drawn at this time are:

1) The maximum creep that may have occurred is <0.6%; the actual creep may be considerably less. This is significantly below the level at which creep is deemed to be a problem. The creep will probably not increase during additional storage due to the low temperature after 15 years and the continual decrease in temperature from decay heat.

2) Within experimental uncertainty, there appears to be no additional fission gas release during the storage period.

3) There is no evidence of hydrogen pickup or hydride reorientation during the storage period.

4) Little, if any, cladding annealing occurred during the pre-storage performance period or storage.

Additional examinations to determine the tensile properties and residual creep strain are underway. The results of these tests, along with the results of the additional characterization work, will be reported at a later date.

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VII. REFERENCES

1. Einziger, R. E., D. L. Baldwin, and S. G. Pitman, April 1998. Interim report. Data Needs for Long-Term Dry Storage of LWR Fuel. TR-108757, Electric Power Research Institute, Palo Alto, CA.

2. McKinnon, M. A., and V. A. Deloach, April 1993. Spent Nuclear Fuel Storage – Performance Tests and Demonstrations. PNL-8451, Pacific Northwest Laboratory, Richland, WA.

3. McKinnon, M. A., and A. L. Doherty, June 1997. Spent Nuclear Fuel Integrity during Dry Storage – Performance Tests and Demonstrations. PNNL-11576, Pacific Northwest National Laboratory, Richland, WA.

4. Dziadosz, D., E. V. Moore, J. M. Creer, T. A. McCann, M. A. McKinnon, J. E. Tanner, E. R. Gilbert, R. L. Goodman, D. H. Schoonen, M. Jensen, and C. Mullen, 1986. CASTOR-V/21 PWR Spent Fuel Storage Cask: Testing and Analysis. EPRI NP-4887/ PNL-5917, Electric Power Research Institute, Palo Alto, CA.

5. Moore, E, July 13, 2001. Personal communication to M.C. Billone of ANL.

6. Vinjamuri, K., M. C. Feldman, C. K Mullen, B. L. Griebenow, A. A. Arave, and R. C. Hill, April 1988. Dry Rod Consolidation Technology Project at the INEEL. EGG-WM—8059.

7. Davis, R. B., January 1981. Pre-Test Nondestructive Examination Data Summary Report on Turkey Point Spent Fuel Assemblies D01, D04, and D06 for the Climax – Spent Fuel Test. HEDL-TME 80-83, UC-70, Hanford Engineering Development Laboratory.

8. Atkin, S. D., June 1981. Destructive Examination of 3-Cycle LWR Fuel Rods from Turkey Point Unit 3 for the Climax – Spent Fuel Test. HEDL-TME 80-89, UC-70, Hanford Engineering Development Laboratory.

9. Van Swam, L. F., A. A. Strasser, J. D. Cook, and J. M. Burger, March 1997. “Behavior of Zircaloy-4 and Zirconium Liner Zircaloy-4 Cladding at High Burnup,” Proc. 1997 Int. Topical Mtg. on LWR Fuel Performance, Portland, OR. p. 421.

10. Einziger, R. E., and R. Kohli, Oct. 1984. “Low-Temperature Rupture Behavior of Zircaloy-Clad PWR Spent Fuel Rods Under Dry Storage Conditions,” Nucl. Tech, 67, p. 107.

11. Northwood, D. O., and U. Kosasih, 1983. “Hydride and delayed hydrogen cracking in zirconium and its alloys” Inter. Metals Rev. 28, No.2, p. 92.

12. Marshall, R. P., and M. R. Louthan, 1963. “Tensile Properties of Zircaloy with Oriented Hydrides,” Trans ASM, 56, p. 693.

13. Tate, D. R., 1945. Trans. Am. Soc. Metals, **35**.