INL/EXT-11-23561

Annotated Bibliography for Drying Nuclear Fuel

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September 2011

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Prepared for the
U.S. Department of Energy
Assistant Secretary for Environmental Management
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

ACKNOWLEDGEMENTS

The author would like to thank Matthew Marston, Charlie Pennington, Joy Russell, Robert Einziger, and John Abrefah for sharing their insights, experience and constructive comments on this work. Special appreciation also goes to Sandy Birk for her strategic advice on this task and introductions to helpful contacts.

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INTRODUCTION

Internationally, the nuclear industry is represented by both commercial utilities and research institutions. Over the past two decades many of these entities have had to relocate inventories of spent nuclear fuel from underwater storage to dry storage. These efforts were primarily prompted by two factors: insufficient storage capacity (potentially precipitated by an open-ended nuclear fuel cycle) or deteriorating quality of existing underwater facilities. The intent of developing this bibliography is to assess what issues associated with fuel drying have been identified, to consider where concerns have been satisfactorily addressed, and to recommend where additional research would offer the most value to the commercial industry and the U. S. Department of Energy.

THE SEARCH STRATEGY

The supporting literature search was pursued in two distinct phases. The initial phase was mostly limited to the operation of drying nuclear fuel. A broad preliminary request was made of the INL Library, hard copy and electronic personal files were reviewed, a modest internet search was undertaken, and select secondary references were sought. In the first phase, nearly 200 items were deemed relevant for this review. Each was perused and an associated comment was recorded. The search was not exhaustive. Internet sources included NRC, the International Atomic Energy Agency (IAEA) and nuclear community (conference-specific) web sites where responses were in some cases untenably slow. Material of interest may have been obscured by tens of thousands of hits or not recognized at all with the attempted variations of pertinent search terms.

The strategy for the second phase of the literature search was to identify and evaluate common sources referenced in those documents found in the first phase. These common sources provided a more detailed understanding of the issues and in the context of their significance to drying fuel.

In general, several observations were clear. First, drying as an industrial process is a well established field with engineering publications and forums devoted exclusively to the topic: a couple of general sources (cited in the sources reviewed) were identified among the many thousands available. Second, search efforts including “dry”, as well as, or instead of “drying”, tend to include the much broader topic of dry fuel storage without efficient means to distinguish or prioritize between them. Also, the initial search response was overwhelmed with reports of the incremental development of technical data to support the Hanford N-Reactor fuel relocation project.[95] To manage the search in a manner consistent with the requested scope, emphasis was put on items defining, limiting, or defending fuel drying processes. Third, some fuel drying designs and patents came up in the search and are included for completeness, without confirmation of whether such concepts have been successfully implemented. And finally, the topics of corrosion mechanisms and corrosion products arise repeatedly in conjunction with the discussion of drying. The drying process and operational temperature limitations, must consider the constituents and configuration of the materials to be dried. The more general corrosion citations have been included in the context of dryness achieved, safe handling practices, and the effects on cladding or containment integrity, but specific corrosion topics can (and have been) the subject of their own separate reviews.[106][107][165][251]

SYNOPSIS OF FUEL DRYING ISSUES

The issues fall into several categories: regulations and guidance, demonstrations and experience with dry storage; corrosion (rates, mechanisms, microbially facilitated; fuel, cladding, and containment materials; modes of failure), residual water (estimates, waterlogged fuel, tolerance for adsorbed/absorbed/free water), pressurization concerns (radiolysis, decay product, corrosion, thermal excursion), process limitations (burnup-, temperature-, and hoop stress-related), and handling potentially pyrophoric materials. While there is a lot of overlap among these categories, they provide an outline for a discussion of where concerns have been identified and whether they have been satisfactorily addressed.

Regulations and Guidance

In the United States, federal requirements under the jurisdiction of the Nuclear Regulatory Commission (NRC) prompt drying to preserve fuel and cladding integrity (10 CFR Part 72, Section 120 Paragraph (h)(1) and Section 122 Paragraph (d)). Also, guidance from the NRC directs drying as the basis for a long-standing licensing device relied upon by the commercial nuclear industry (NUREG-1536). By contrast, the Department of Energy (DOE) has custody of much fuel for which the cladding is no longer intact. In either case, the most desirable benefit of drying is the mitigation of the liquid phase corrosion environment that enables or accelerates so many of the fuel, cladding, and containment degradation mechanisms.

A consensus standard has been developed to incorporate drying practices that mitigate chemical reactivity and over-pressurization potential, and (to the extent possible) preserve cladding integrity.[25] However, from a regulatory perspective, commercial interests are currently focused on accumulating data to support re-licensing existing dry storage casks. While there has been some discussion of unforeseen quantities of liquid water in the dry storage environment, there is relatively little concern with the efficacy of current fuel drying practices.

Demonstrations and Experience

Dual purpose casks have been developed to satisfy NRC requirements for storage and transportation of commercial fuel. Patents and cask vendor publications illustrate broad system design considerations, and experimental and operational data benchmark the practical experience for various materials and configurations. Several successful demonstration programs have been undertaken.[135] Hanford has loaded, drained, dried and conditioned thousands of multi-canister overpack (MCO) containers. INL has dried, packaged and relocated canisters of TMI-2 fuel debris from underwater to (sealed) horizontal dry storage and has dried and consolidated the inventories of the MTR Canal and CPP-603 Basin to (vented) dry storage in the Irradiated Fuel Storage Facility (IFSF). Several nations have taken similar steps and others have dried and encapsulated fuel for continued wet storage (in a new location).[41][47][441]

The drying methods are varied, but generally favor vacuum drying technology to facilitate the mass transfer, often in combination with a purge or cyclic backfill to improve heat transfer between the heat source and any remaining water. Forced helium gas drying has also gained acceptance.[453]

The determination of vacuum drying endpoint was usually influenced by, if not established outright by the NUREG-1536 [299][300] illustration of drying in accordance with PNNL-6365 [217], calling for evacuation to $\leq$ 3.0 mm Hg and having the pressure retained through a 30 minute isolation period. The expectation is that any liquid water remaining in the isolated volume would produce a distinct increase in pressure within that time. The forced helium gas drying system establishes dryness based on the moisture content (water vapor pressure) in the outlet gas; this endpoint can be estimated with the number of turnover exchanges of cavity free volume.[396] The perception is that the basis for these criteria for dryness was somewhat arbitrary.

PNNL-6365 documents an analysis of the cover gas constituents measured for representative storage casks in service for < 1 year and shows that reactions over a 40-year operating period would not cause significant cladding degradation for PWR and BWR fuels.[217] Note that the data reflect gas analysis of the in service casks and are intended to show preservation of cladding integrity. These data do not account for equilibrium water vapor pressure over hydrated corrosion products or physically adhering water within a high surface area particle bed (that might be present in badly degraded fuel or fuel debris but the resultant vapor pressure might take longer than 30 minutes to develop).[105] Neither do these data account for a situation where the vacuum impedes the heat transfer from the heat source to any residual water, although NUREG-1536 Rev. 1 does caution against masking effects of icing adding the suggestion of a staged draw down.[299] Likewise, there could be unanticipated moisture holdup after forced helium drying if occluded internal volume is not at equilibrium with the outlet gas.

Corrosion

Corrosion is a fundamental consideration to wet and dry storage and much attention has been given to estimating rates and establishing mechanisms to enable dry storage and repository performance modeling. The corrosion environment in sealed dry fuel container is a complex equilibrium, so predictions incorporate assumptions that may be overly conservative.

Fuel Corrosion

Oxidation of the fuel must be considered in the context of the fuel material. Consistent with their contribution to the overall used fuel inventory, corrosion of uranium metal, UO_2, and uranium aluminide fuels were well represented among the items reviewed. Failure mechanisms included the potential for gross cladding rupture with the expansion that occurs with oxidation to the less dense U_3O_8. Pyrophoric byproducts of uranium metal corrosion (discussed separately) were another consideration. In general, corrosion mechanisms and rates are reasonably well described for temperatures and materials of interest.

Cladding Degradation

Multiple mechanisms have been considered supporting predictions that supported the initial dry storage licenses: primarily creep, stress corrosion cracking, delayed hydride cracking, and oxidation. And significant work has been invested in their study. Within the licensed storage (and transportation) environment, cladding is a redundant means of containment. However, given the history of robust zircaloy cladding performance for commercial light water reactor fuel, there is ongoing interest in the understanding and accurate prediction of cladding failure to enhance confidence in the safety performance of commercial nuclear fuel. Temperature restrictions for drying and dry storage have been instituted to preserve cladding and fuel integrity.

Containment Material Aging

Drying discussions typically focus on fuel and cladding performance, but some consideration has been given to containment throughout storage and transport. The Standard Guide for Evaluation of Materials Used in Extended Service of Interim Spent Nuclear Fuel Dry Storage Systems reflects the current consensus on this topic, complimenting similar efforts in support of repository licensing.[26] However, the fuel and cladding integrity are usually the limiting considerations for drying. Storage system materials such as steel and concrete have been widely studied for general industrial applications. Some work has been done to assess their performance with prolonged exposure to gamma radiation, but in the context of drying containment material aging is not an urgent issue.

Microbially Induced Corrosion

Mostly studied with respect to otherwise inexplicable wet storage corrosion observations, microbes can hasten corrosion. Some thrive in underwater fuel storage pools, and viable biofilms have been confirmed in the high radiation fields on the cladding of used fuel.[51] In the absence of water, microbes are expected to be benign, but they could affect the drying rate or level of dryness attained, and they may be viable to enhance corrosion upon re-wetting. While dry storage is intended to be dry, water ingress events have been detailed for some facilities, and the effectiveness of any drying process can be circumvented by human error. Additional work in this area should be considered, although perhaps it would be of greater interest with respect to repository disposal than in the context of dry storage.

Residual Water

For some cases, fuel defect models have been employed to calculate residual water after drying.[319] Such water remains in one or more of three forms: chemically adsorbed as waters of hydration associated with oxides, physically absorbed on exposed surfaces, or free liquid water. Waterlogged fuel has been addressed both by calculation and experimentally: it dries more rapidly than a credibly configured particle bed and without gross disruption to the cladding regardless of vacuum conditions applied.[99][219][220][249][253] Estimates of residual water tend to have a large uncertainty that depends heavily on knowledge of fuel condition.

Radiolysis and Pressurization Concerns

Radiolysis of residual water has been assessed by calculation (based on published experimental data) for the DOE standard canister, and was found to be a comparatively small contribution to potential canister pressure by comparison to thermal equilibration and corrosion considerations.[264][463] More recent experimental data suggests that the hydrogen yield for radiolysis may be an order of magnitude less than that assumed for the calculation.[102][174] Depending upon the cover gas, radiolysis may affect the corrosion environment in dry storage, but it does not generate quantities of gas from adsorbed moisture. Back reactions appear to limit the overall product gas pressure.[68] Likewise, decay products have been shown to be an insignificant contributor to gas pressure.[403] Corrosion reactions may consume oxygen or generate hydrogen, or both, depending on the species available, but a thermal excursion (as in an accident scenario) is the greater risk for pressurization.

Process Limitations

High burnup commercial fuel shows some additional vulnerability: to stress corrosion cracking, delayed hydride cracking, and thermal stress-related considerations in general. High burnup produces more fission products and contributes to higher hoop stress on the fuel cladding while simultaneously introducing higher levels of CsI (a source for iodine at the inside cladding surface) and higher initial decay heat.[131] High temperature drying of high burnup fuel (>45GWd/MTU where fuel temperature exceeds 400°C) appears to facilitate the hydride reorientation that can lead to embrittlement of the cladding.[242] NRC regulations have constrained fuel temperatures and thermal cycling (both during drying and in storage), but the nature of the concern is preserving the cladding integrity.[451] High burnup fuels may suffer from competing priorities: thermal limitations implemented to mitigate these cladding degradation mechanisms may lead to a problem of insufficient drying. Forced helium drying appears to address this matter, but appropriate conservatism in the predicted performance needs to be validated over extended storage for high burnup fuels.

Other fuel types must be considered on a case by case basis. Aluminum-clad plate-type UAlx fuels are also subject to certain constraints. Of the major fuel materials, aluminum is most susceptible to general and pitting corrosion, prompting early and ongoing interest in its transition to dry storage. Aluminum fuels are susceptible to creep and diffusion mechanisms at relatively low temperatures, hence drying and storage should be kept to temperatures below 200°C.[392]

Potentially Pyrophoric Materials

The transfer of corroded uranium metal fuels from wet to dry storage presents the distinct hazard of handling a potentially pyrophoric material. Study of this issue covers three areas: 1) the burgeoning understanding of the corrosion conditions favorable to the production of uranium hydride,[98][363][447] 2) the ignition and thermodynamic properties of the materials that define the magnitude of the hazard,[328][381] and 3) the development of a conditioning process to mitigate the hazard.[4][158][433] While the literature on this issue is somewhat limited, the pertinent details are adequately described.

RECOMMENDATIONS FOR FUTURE WORK

Estimates of the amount of residual water with the fuel in dry storage carry considerable uncertainty, particularly with materials degraded in service, damaged during handling, or severely corroded in wet storage. This uncertainty has been accepted, and drying procedures have evolved to improve confidence that water is reduced to a satisfactory level. The usual strategy is to address specific issues with a bounding methodology. However, residual water remains a consideration in the context of the expanding storage interim. The availability of water drives general corrosion, contributing to a complex equilibrium that does not readily lend itself to study. The study of individual contributing gas equilibrium mechanisms (such as radiolysis, corrosion, and water vapor pressure over hydrated corrosion products) enhances confidence in dry storage and the subsequent transportation of these fuels. These have been productive venues for research to support drying spent fuel for interim dry storage as well as transportation and ultimate disposition.

Also, demonstration projects have been extremely useful for validating the storage environment and subsequent fuel performance. In some cases, cover gas analysis and materials examination and testing have been used to this effect.[66][118][217] Such ongoing activities demonstrate progress supporting the increasing duration needed for interim dry storage. Wherever practical to do so, efforts at monitoring and surveillance, and periodic examination of fuel in prolonged dry storage are recommended. The need to demonstrate success or recognize and mitigate the conditions leading to failure is prompted by re-licensing efforts as the interim for dry storage expands in the absence of a firm plan for final disposition.

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45	Blackburn, L. D., D. G. Farwick, S. R. Fields, L. A. James, and R. A. Moen, Maximum Allowable Temperature for Storage of Spent Nuclear Fuel, An Interim Report, HEDL-TME-78-37, UC-70, May 1978.	An early assessment of cladding failure mechanisms for dry storage or repository environments and considerations for appropriate temperature constraints. A general temperature limit of 380°C is given for PWR fuel based on stress-rupture considerations, with 395°C for a "very short-term" application. Mention is made that stress corrosion cracking may lead to a lower limit. BWR fuel limits suggested are slightly higher.
46	Boase, D. G., and T. T. Vandergraaf, The Canadian Spent Fuel Storage Canister: Some Materials Aspects, Nuclear Technology, Vol. 32, January 1977, pp. 60-71.	Dry storage instrumented demonstration in sealed carbon steel baskets in concrete canisters at Whiteshell. Concerns assessed are canister structural integrity, corrosion of fuel basket and canister lining, fuel sheath oxidation, and UO_2 oxidation. Eliminated potential for fuel oxidation by using helium backfill in baskets. Suggests that oxidation to U_3O_7 which is accompanied by much less swelling, may dominate below 250°C.
47	Borek, E., W. Bykowski, S. Chwaszczewski, and W. Czajkowski, Encapsulation technology of MR6 spent fuel and quality analysis of the EK-10 and WWR-SM spent fuel stored more than 30 years in wet conditions, Conference: RRFM 2002: 6. International Topical Meeting on Research Reactor Fuel Management, Institute of Atomic Energy, Otwock - Świerk, Poland, 2002, pp. 170-174.	Heated air drying (50-110C) used for fuel with undamaged cladding. Selected as less expensive than US heated vacuum drying methods. (Modified process for damaged fuel - no detail given.) Dry fuel first, then encapsulate. Endpoint at outlet R.H. of 5%. Seal weld, evacuate, leak check, then helium backfill to 0.2 MPa. Encapsulated fuel may be returned to wet storage. First 3 MR-type fuel elements were dried in June 2001.
48	Bradley, E. R., W. J. Bailey, A. B. Johnson, Jr., and L. M. Lowry, Examination of Zircaloy-Clad Spent Fuel After Extended Pool Storage, PNL-3921, UC-85, Pacific Northwest Laboratories, Richland, WA, September 1981.	Examines some of the oldest Zircaloy clad fuel available at the time. After 21 years in underwater storage, no evidence of accelerated corrosion or hydride redistribution in the cladding was observed.
49	Brooks, P., and R. L. Sindelar, Characterization of FRR SNF in Basin and Dry Storage Systems, Westinghouse Savannah River Company, 1998?, pp. 542-545.	No change in cladding condition observed for dry stored aluminum clad fuel except where water inadvertanly entered the storage canisters.
50	Brown, C., J. Guttmann, C. Beyer, and D. Lanning, Maximum Cladding Stresses for Bounding PWR Fuel Rods During Short Term Operations for Dry Cask Storage, Paper 1106, Proceedings of the 2004 International Meeting on LWR Performance, Orlando, Florida, September 19-22, 2004, pp. 459-466.	Presents the technical justification (based on hoop stress) for allowing higher temperature limits for short-term operations (such as drying) for low burnup fuel cladding.

51	Bruhn, D.F., S.M. Frank, F.F. Roberto, P.J. Pinhero, and S.G. Johnson, "Microbial Biofilm Growth on Irradiated, Spent Nuclear Fuel Cladding," Journal of Nuclear Materials, Vol. 384, 2009, pp. 140-145.	Provides evidence of biofilm survival (Gram-negative organisms and sulfer reducing bacteria) on spent nuclear fuel cladding demonstrating tolerance for the radiation fields associated with such materials.
52	Burtseva, T. A., Y. Yan, and M. C. Billone, Radial-Hydride-Induced Embrittlement of High-Burnup ZIRLO Cladding Exposed to Simulated Drying Conditions, acc: ML101620301, Argonne National Laboratory, June 30, 2010.	Experimental data for high burnup (70MWd/MTU) North Anna Reactor fuel. Examines behavior with heating-cooling cycle representative of drying conditions.
53	Cappelaere, C., R. Limon, T. Bredel, P. Herter, D. Gilbon, P. Bouffioux and J.P. Mardon, "Long Term Behavior of the Spent Fuel Cladding in Dry Storage Conditions," ASME, 8th International Conference on Radioactive Waste Management and Environmental Remediation, Bruges, Belgium, October 2001.	Creep as a contributing mechanism to cladding failure in dry storage. Results of 10-60 day experiments on zircaloy-4, at 380-420°C with hoop stress of 150-250 MPa. Formulation of thermal creep law and observation of radial hydride formation after cooling at 80 MPa.
54	Carlsen, B. W., D. Fillmore, R. McCormack, R. Sindelar, T. Spieker, and E. Woolstenhulme, Damaged Spent Nuclear Fuel at U.S. DOE Facilities, Experience and Lessons Learned, INL/EXT-05-00760, November 2005.	DOE Experience with fuel damage and complicating factors associated with handling failed fuel. Describes NAC-1E event, Tory IIA packaging, and more general corrosion concerns.
55	Caskey, G.R., Jr., Materials Issues in Interim Storage and Direct Disposal of Aluminium-clad Spent Nuclear Fuel, WSRC-TR-93-502, Westinghouse Savannah River Company, Aiken, SC, September 1993.	Recommends that the maximum temperature for'interim storage of aluminum clad SNF should be held below 150°C. The ductility of irradiated 1100 and 6061 aluminum reduces to less than 1% between 150 and 200°C making the cladding susceptible to failure. The protective oxide film on aluminum may be degraded above 120°C. Aluminum may blister in moist air at temperatures above ~250°C.
56	Chao, C.K., J.Q. Chen, K.C. Yang, C.C. Tseng (2007). "Creep Crack Growth on Spent Fuel Zircaloy Cladding in Interim Storage." *Theoretical and Applied Fracture Mechanics*, Vol 47, No. 1, February 2007.	Examines creep crack growth with temperature profile. Shows that a higher rate of temperature increase causes a larger creep growth rate.
57	Chao, C.K., K.C. Yang, C.C. Tseng (2008). "Rupture of Spent Fuel Zircaloy Cladding in Dry Storage Due to Delayed Hydride Cracking." *Nuclear Engineering and Design*, Vol. 238, pp. 124-129.	Delayed hydride cracking was considered and crack propagation appeared to be aggravated if hydride orientation is shifted from the circumferential to the radial direction. ANSYS computer code was used with stain density theory to analyze the relationship.
58	Chen, Kuo-Fu, W. S. Large, R. L. Sindelar, Vacuum Drying Tests for Storage of Aluminum Spent Nuclear Fuel, 1998 international high-level radioactive waste management conference, Las Vegas, May	Vacuum drying (no external heating) feasibility test using mock aluminum fuel (with mock decay heat of 0.0 to 14 Watts), no corrosion product considered.

	1998, pp. 719-720.	
59	Childs, K., A. Christensen, and E. Woolstenhulme, Improved Drying Rate Diagnostics for Saturated Fuel Debris at the INEEL, Topics in the Fuel Cycle, annual meeting of the American Nuclear Society (ANS), September 1999, pp. 48-49.	Operational experience. Endpoint determination to expedite process throughput.
60	Chin, B. A., and E.R. Gilbert, "Prediction of Maximum Allowable Temperatures for Dry Storage of Zircaloy-Clad Spent Fuel in Inert Atmosphere," Nuclear Technology, Vol. 85, April 1989, pp. 57-65.	Creep rupture and cumulative damage modeling under declining dry storage temperatures. The allowable initial storage temperature depends on cladding stress, decay heat (heat source), and heat dissipation from the storage environment.
61	Chin, B. A., M. A. Khan, and J. Tarn, Deformation and Fracture Map Methodology for Predicting Cladding Behavior During Dry Storage, Pacific Northwest Laboratory, PNL-5998, UC-85, DE87 001848, Richland, Washington, September 1986.	Deformation and fracture theories were used to develop maps. Where available, experimental data were used to test their validity. Predictive equations and cumulative damage methodology were used to take credit for declining storage temperatures. Compared predictions with results from tests under abnormal conditions, methodology appears to be conservative (inconsistencies in strain data not withstanding).
62	Chollet, P., and Y. Guenon, Interim Dry Storage System Technologies and Innovations, VARNA 2002.	TN-24 cask family and NUHOMS modular dry storage system discussed.
63	Chopra, O. K., C. Regan, S. Lee, D. C. Ma, and W. J. Shack, Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal, NUREG-1557, October 1996.	Aging issues for dry storage license renewal. Aging pertains specifically to the containment structures (concrete, steel, etc.) and not the fuel.
64	Christensen, A. B, and K. M. Wendt, Drying of Spent Nuclear Fuels and Fuel Debris in a Vacuum Furnace, Radioactive Waste Management and Environmental Remediation, American Society for Testing and Materials (ASTM), 2001, pp. 1193-1198.	Vaporization rate trend as an indicator for monitoring drying, discussed in the context of the three vacuum drying furnace systems used in at INL.
65	Christensen, A. B, EDF-1466, Rev. 2, Validation of Water Content in TMI-2 Canisters During Drying in the Heated Vacuum Drying System, June 28, 2000.	Internal Report. Calculates water content.

66	Christensen, A. B., K. Custer, R. Gardner, and J. Kaylor, Receipt and Storage Issues at the TMI-2 Irradiated Fuel Storage Installation, ICONE10-22649, Proceedings of ICONE10, 10th International Conference on Nuclear Engineering, Arlington, VA, April 2002.	Operational experience. Suggests season and position dependent hydrogen generation due primarily to corrosion at the DSC (rather than radiolysis within the TMI-2 canisters), vented via vent and purge ports with HEPA filters
67	Christensen, A. B., K.D. Fielding, L.W. Madsen and D.R. Teuscher, "Lessons Learned During Start- Up of a Facility for Storing Spent LWBR in Dry Wells at the Idaho Chemical Processing Plant," CONF-860417, Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986, pp. S-156 to S-174.	Second generation dry wells installed and Shippingport LWBR fuel transferred from underwater storage. Discusses observations and gas anaylses from first generation dry wells (in use since 1971) and compares fuel inventory characteristics (for Peachbottom, Fermi LMFBR Blanket, and Shippingport LWBR). Presents design enhancements made to address first generation deficiencies.
68	Christensen, H., and S. Sunder, Current State of Knowledge of Water Radiolysis Effects on Spent Nuclear Fuel Corrosion, Nuclear Technology, Vol. 131, July 2000, pp. 102-122.	Gamma (not alpha) radiolysis UO_2 corrosion kinetic model and comparison of corrosion in O_2 and H_2O_2 without irradiation.
69	Christensen, H., S. Sunder, and D.W. Shoesmith, Development of a Kinetic Model to Predict the Rate of Oxidation and Dissolution of Nuclear Fuel (UO_2) by the Radiolysis of Water, AECL-11102, COG-93-488, CA9500413, Vol. 27, No. 4, March 1994.	Models UO_2 dissolution under conditions expected in a nuclear waste vault (groundwater in a gamma field).
70	Chu, H.C., S.K. Wu, K.F. Chien, R.C. Kuo (2006). "Effect of Radial Hydrides on the Axial and Hoop Mechanical Properties of Zircaloy-4 Cladding." *J. of Nuclear Materials*, 362:1, 30 June 2006, pp. 93-103.	The effect of radial hydrides on mechanical properties was studied. The effect on axial properties was negligible. Hoop tensile properties were scattered with a general trend of degradation with increased fraction of radially oriented hydrides.
71	Chu, H.C., S.K. Wu, R.C. Kuo (2008). "Hydride Reorientation in Zircaloy-4 Cladding." *J. of Nuclear Materials*, V. 373, 15 February 2008, pp. 319-327.	Studied formation of radial hydrides in stress-relief annealed Zircaloy-4 cladding. Percent of radial hydrides increases to saturation with thermal cycles.
72	Chung, H. M., "Fundamental Metallurgical Aspects of Axial Splitting in Zircaloy Cladding," Proceedings of the International Topical Meeting on Light Water Reactor Fuel Performance, American Nuclear Society, Park City, Utah, April 10-13, 2000.	Correlation of microstructural hydride behavior with axial splitting in irradiated zircaloy cladding tubes in a hot cell at 292-325°C in argon.
73	Chung, H. M., Understanding Hydride- and Hydrogen-Related Processes in High-Burnup Cladding in Spent-Fuel-Storage and Accident Situations, Paper 1064, Proceedings of the 2004 International Meeting on LWR Performance, Orlando,	Specifically considers radial-hydride-assisted delayed hydride cracking in the context of vacuum drying high burnup fuel, but hedges on conclusions based on the performance of precracked specimens.

	Florida, September 19-22, 2004, pp. 470-479.	
74	Chung, H.M.; Yaggee, F.L.; and Kassner, T.F., "Fracture Behavior and Microstructural Characteristics of Irradiated Zircaloy Cladding," Zirconium in the Nuclear Industry, Seventh International Symposium, ASTM STP 939, R. B. Adamson, and L. F. P. Van Swam, Eds., American Society for Testing and Materials, Philadelphia, Pennsylvania, 1987, pp. 775-801.	Brittle fracture of zircaloy cladding from commercial fuel (burnup >22 MWd/kg U) at 325°C. Irradiation induced Zr_3O and ZrO_2 precipitates from reactor service. Failures were augmented by bulk hydride precipitation. Differences in loading and crack propagation indicate intrinsic hoop stress owing to the outer suface oxide layer.
75	Chung, Sung-Hwan,, Chang-Yeal Baeg, Byung-Il Choi, Heung-Young Lee, and Yong-Jae Kim, Operational experiences of on-site transport of spent nuclear fuels in Korea, Conference: PATRAM 2004: 14. international symposium on the packaging and transportation of radioactive materials, Berlin, Germany, September 20-24, 2004.	Used KSC-4 casks and CASTOR KN-12 casks for transport of PWR fuel assemblies. Provided for vacuum drying with helium backfill, but fuel transported wet to save whole transport cylce time (on site, wet to wet storage transfer).
76	Clough, M., D. Willey, Z. Banfield, J. Eldridge, and P. Gilchrist, Inactive experiments investigating the vacuum drying of wetted Magnox fuel, GLOBAL 2009 Congress: The Nuclear Fuel Cycle: Sustainable Options and Industrial Perspectives, June 2009.	Abstract only. Rate and extent of water removal experimentally determined. Cold vacuum drying considered, but no detail available.
77	Colmenares, C. A., “Oxidation Mechanisms and Catalytic Properties of the Actinides”, Prog,. Solid State Chem., Vol. 15, Pergamon Press, Ltd, Great Britain, 1984, pp. 257-364.	Details mechanisms, rate expressions, and corrosion product descriptions for uranium reactions with oxygen, air and water. The inhibiting effect of oxygen is detectable at 10 vppm and has full effect at 100 vppm. The degree of inhibition is also proportional to the water vapor pressure.
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80	Cooper, T. D., Spent Nuclear Fuel Project (SNFP) Estimate of Volatile Fission Products Release From Multi-Canister Overpacks, WHC-SD-SNF-TI-033, Rev. 0, EDT/ECN 616633, July 1996.	MCO gas generation evaluated.
81	Cooper, T. D., Spent Nuclear Fuel Project Gas Generation From N-Fuel in Multi-Canister Overpacks, WHC-SD-SNF-TI-028, Rev. 0, July 1996.	MCO gas generation evaluated. Pyrophoric material addressed.
82	Corcoran, V. J., C. Johnston, W. J. Metcalfe, and J. Thorpe, The Water Vapour Corrosion of Uranium and its Prevention, AWRE O-42/65, July 1965.	Examines the reaction between uranium and water vapor in nitrogen and oxygen environments at 25°C and 40°C. Production of hydrogen from uranium and moist nitrogen was suppressed in the presence of oxygen (which was gradually consumed). Also looked at sulphide coating and arsine coating for inhibiting oxidation.
83	Cox, B, "Pellet-Clad Interaction (PCI) Failures of Zirconium Alloy Fuel Cladding – A Review," Journal of Nuclear Materials, Vol. 172, 1990, pp. 249-92.	History (literature search and summary) of pellet-clad interaction (PCI) failures (in reactor service). Stress corrosion cracking (SCC) induced by fission product iodine is considered the probable cause.
84	Cox, B., "Environmentally-Induced Cracking of Zirconium Alloys – A Review," Journal of Nuclear Materials, Vol. 170, 1990, pp. 1-23.	Considers cracking mechanisms in Zr alloys. Hydrogen embrittlement in Zr alloys occurs only where there is precipitated hyrdide.
85	Cox, D. S., I. B. Bainbridge, and K. R. Greenfield, Management of Legacy Spent Nuclear Fuel Wastes at the Chalk River Laboratories: Operating Experience and Progress Towards Waste Remediation, Waste Management 2005, WM'05 Conference, Tuscon, Arizona, February 27 - March 3, 2005.	AECL preparations to relocate 22 tonnes of mostly metallic uranium fuel from tile holes to above ground dry storage. Equipment is used to pierce and vent contained gases and passivate the material in a controlled manner prior to canister handling (removal from tile hole and vacuum drying).
86	Creer JM, RA McCann, MA McKinnon, JE Tanner, ER Gilbert, RL Goodman, D Dziadosz, EV Moore, DH Schoonen, M Jensen, and C Mullen. 1986. The Castor-V/21 PWR Spent-Fuel Storage Cask: Testing and Analyses. EPRI NP-4887, PNL-5917, Interim Report, prepared by Virginia Power Company, published by the Electric Power Research Institute, Palo Alto CA, November 1986.	Surry fuel dry storage cask demonstration, the focus of this report is thermal and shielding performance.
87	Creer, J. M., and D. H. Schoonen, "CASTOR-V/21 PWR Spent Fuel Storage Cask Performance Test," Third International Spent Fuel Storage Technology Symposium/Workshop, CONF-860417, Seattle, Washington, April 8-10,	Abridged version of [86]. Surry fuel dry storage cask demonstration, the focus of this report is thermal and shielding performance. Discusses the details of observed basket cracking.

1986, pp. S-20 to S-42.

88	Creer, J. M., T. E. Michener, M. A. McKinnon, J. E. Tanner, E. R. Gilbert, R. L. Goodman, D. A. Dziadosz, E. V. Moore, H. S. McKay, D. P. Batalo, D. H. Schoonen, M. F. Jensen, and C. K. Mullen, TN-24P PWR Spent Fuel Storage Cask Performance Testing and Analyses, PNL-6054, UC-85, EPRI NP-5128, Research Project 2406-4, Electric Power Research Institute, April 1987.	Performance test of the TN-24 Cask with Surry PWR fuel. One or more fuel rod(s) developed a leak possibly due to cask rotation from vertical to horizontal orientation during the test.
89	Cubicciotti, C., and R. L. Jones. 1978. EPRI-NASA Cooperative Project on Stress Corrosion Cracking of Zircaloys. EPRI NP-717 (RP 455-l) , Electric Power Research Institute, Palo Alto, California, March 1978.	Studied stress corrosion cracking, mechanisms of crack initiation and propagation, and in-service pellet cladding interaction failures associated with local and overal power change. Iodine (from radiolysis of CsI), cadmium, and irom (in liquid cesium) were likely agents in PCI failures.
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91	Cunningham, M. E., E. P. Simonen, R. T. Allemann, I. S. Levy, R. F. Hazelton, E. R. Gilbert, Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere, PNL-6364, UC-85, Pacific Northwest Laboratory, October 1987.	The original basis for NUREG-1536. Zircaloy creep rupture, stress corrosion cracking, and hydride reorientation prompting temperature limitations for drying and dry storage.
92	D. T. Reed and R. A. Van Konynenburg, "Effect of Ionizing Radiation on the Waste Package Environment", DCRL-JC--107180-Rev.l, DE91 015125, Lawrence Livermore National Laboratory, in High Level Radioactive Waste Management: Proceedings of the Second International Conference, pp. 1396-1403, American Nuclear Society and American Society of Civil Engineers, Las Vegas, Nevada, April 28-May 3, 1991.	The potential effects of nitrogen acid and ammonia formation need to be considered for the expected repository (air) environment. Hydrogen and atomic hydrogen are the most important radiolytic products in the water vapor system.
93	Daum, R. S., S. Majumdar, and M.C. Billone, "Experimental and Analytical Investigation of the Mechanical Behavior of High-Burnup Zircaloy-4 Fuel Cladding," Journal of ASTM International, Vol. 5, No. 5, Paper ID JAI101209, May 2008,	Discusses radial hydrides, corrosion-induced oxides, ring-compression-type loading, and finite element modeling and analysis.

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94	Daum, R. S., S. Majumdar, Y. Liu, and M. C. Billone, Radial-hydride Embrittlement of High-burnup Zircaloy-4 Fuel Cladding, Journal of Nuclear Science and Tecnology, Vol. 43, No. 9, Paper 4309O 2006, pp. 1054-1067.	Assesses the threshold hoop stress to precipitate radial hydrides (embrittlement) in unirradiated and high-burnup zircaloy cladding.
95	DeVine, J. C., R. G. Ballinger, T. L. Bradley, R. P. Denise, R. D. Filbert, C. E. Foreman, J. C. Janus, A. B. Johnson, Jr., J. D. Martin, J. V. Robinson, K. A. Simpson, and R. F. Williams, Dry Storage of N Reactor Fuel, Independent Technical Assessment, Volume II, September 1994.	Forms the foundation for the extensive testing program used to demonstrate the cleaning, packaging, drying and conditioning processes for N Reactor fuel. Dryness discussion indicates Duke Power estomated remaining moisture at 0.25%.
96	Dienst, W., S. Nazaré, and F. Thümmler, Irradiation Behaviour of UAl_x-Al Dispersion Fuels for Thermal High Flux Reactors, Journal of Nuclear Materials, Vol. 64, 1977, pp. 1-13.	Blister tests (swelling under post-irradiation annealing) at higher than normal operating temperatures and uranium aluminide fuel performance discussion.
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100	Dyer, R. S., E. Barnes, R. L. Snipes, S. Høibråten, H. C. Gran, E. Foshaug, and V. Godunov, Completion of the First Integrated Spent Nuclear Fuel Transshipment/Interim Storage Facility in NW Russia, Waste Management 2003 Symposium, Tuscon, Arizona, February 23-27, 2003.	Residual water in the cask may be up to 3.2 liters. Improvements are under consideration since this much water could contribute to significant degradation over an extended (50 year) storage period.
101	E. R. Johnson Associates, A Preliminary Assessment of Alternative Dry Storage Methods for the Storage of Commercial Spent Nuclear Fuel, JAI-180, DOE/ET/47929-1, UC-85, DE82 002303,	Assesses the use of casks, drywells, concrete silos, and air-cooled vaults for at-reactor dry storage options. Cask storage receives top ranking for cost.

	U.S. Department of Energy, November 1981.	
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103	Ebner, M. A., EDF-2441, Rev. 0, Composition of Cover Gases in TN-REG and TN-BRP Casks, Project File No. 7652.4.9.1, August 29, 2002.	Internal Report. Discusses observed rate of hydrogen formation and pressure rise in R. E. Ginna and Big Rock Point fuel shipped from West Valley.
104	Ebner, M. A., EDF-2604, Potential Residual Water in Vacuum-Dried West Valley Spent Nuclear Fuel, WV-FRC-0002A, transmitted by letter to John L. Mahoney of West Valley Nuclear Services from Larry J. Ferrell (INEEL), CCN 20444, April 12, 2001.	Internal Report. Estimates potential for residual water in West Valley TN-REG and TN-BRP fuel casks.
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106	Ebner, M. A., R. A. Jordan, and S. O. Bates, Literature Review of the Drying Characteristics of Uranium, Aluminum, Iron, and Spent Fuel Corrosion Products, INEEL/EXT-2000-01038, February 2001.	Considers waters of hydration and potential for corrosion products to hold up water during (and after) drying.
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108	Ebner, M. A., The Potential Pyrophoricity of BMI-SPEC and Aluminum Plate Spent Fuels Retrieved From Underwater Storage, INEL-96/0235, August 1996.	Discussion of uranium hydride pyrophoric material: potential for generation and recommendations for conditioning and handling.
109	Ebner, M. A., to J. M. Mines, Evaluation of the Pyrophoricity of the CFRMF Core Filter, MAE-06-96, Lockheed Idaho Technologies Company, Interdepartmental Communciation, April 3, 1996.	Hydride content and potential for ignition are limited somewhat by the massive, monolythic nature (and comparatively limited surface area) of this item.

110	Ebner, M. A., to R. A. Schiffern, Basis for the Drying and Passivation Protocol, MAE-01-97, Lockheed Idaho Technologies Company, Interdepartmental Communciation, January 22, 1997.	INEL passivation with 1-3% oxygen in argon at 150°C is an equivalent procedure to that used in the UK for Magnox metal fuels. Attachment 1 includes an annotated bibliography. Attachment 2 includes a detailed discussion of reaction kinetics for metal particulates and hydride particulates.
111	Einfeld, K., and F. W. Popp, "Dry Storage Systems Using Casks for Long Term Storage in an AFR and Repository," CONF-860417, Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986, pp. S-99 to S-116.	Considers a 500 year storage cask design for repository emplacement.
112	Einfeld, K., J. Fleisch, A. Lührmann, and K. Ramcke, "Operating Experience of Cask-Type Dry Storage and Spent Fuel Performance," in "Proceedings of 7th International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM 83)," Conference, No. 830528, 1983, pp. 240-241.	Pilot demonstration of BWR fuel storage in a CASTOR cask.
113	Einziger, R. E, and R. E. Woodley, Evaluation of the Potential for Spent Fuel Oxidation Under Tuff Repository Conditions, HEDL-7452, DE86 015889, Hanford Engineering Development Laboratory, March 1985.	Concludes based on dry air oxidation data and model that a cladding breach must not occur in the first 161 years to prevent spallation (considering that higher temperatures and radiation field decays with time). A cladding breach after 160 years may lead to partial oxidation that enhances leach rate without splitting cladding. If stored in dry air, spallation can be expected in the first 300 years for rods that are initially breached. Insufficient data were available to support conclusions on radiolysis and moist air oxidation.
114	Einziger, R. E., "Technical Bases for Extended Dry Storage of Spent Nuclear Fuel," EPRI Final Report 1003416, Electric Power Research Institute, December 2002.	Provides an overview of degradation issues, existing data and an outline of surveillance programs. Assesses strategies to address re-licensing issues and finds that essentially all are covered either by the initial application or by ongoing monitoring.
115	Einziger, R. E., and J. A. Cook, "LWR Spent Fuel Dry Storage Behavior at 229"C," Electric Power Research Institute, Palo Alto, CA, NUREG/CR-3708, 1984.	Breached PWR and BWR rods (with artificially induced defects) stored in either unlimited air or inert atmosphere for 5962 hours. One BWR rod in unlimited air did show cladding deformation and crack growth. X-ray analysis showed UO_2 had oxidized to U_3O_8.

116	Einziger, R. E., and R. Kohli, “Low Temperature Rupture Behavior of Zircaloy-Clad Pressurized Water Reactor Spent Fuel Rods under Dry Storage Conditions,” Westinghouse Hanford Company, HEDL-7400 DE84 007170, (full text), Nuclear Technology, Vol. 67, October 1984, pp. 107-123 (abridged).	Spent fuel rods (Turkey Point PWR) were artificially pressurized to 150 MPa and tested at 323°C for up to 2101 hours. Hydride reorientation was observed.
117	Einziger, R. E., and R. V. Strain, Oxidation of Spent Fuel at Between 250 and 360°C, EPRI NP-4524, Project 2062-10, Topical Report, Electric Power Research Institute, April 1986.	Early work to investigate temperature and environment on the oxidation of UO2 and determine whether the size of a cladding defect affects the oxidation behavior.
118	Einziger, R. E., H. C. Tsai, M. C. Billone, and B. A. Hilton., Examination of spent PWR fuel rods after 15 years in dry storage, NUREG/CR-6831, acc: ML032731021, Argonne National Laboratory, August 2003.	Evaluation of Surry PWR fuel after 15 years in dry storage. Peak drying temperatures over 400C, burnup up to 36 GWd/MTU. No degradation or undesireable effects observed (including unacceptable creep and hydride reorientation that could lead to embrittlement).
119	Einziger, R. E., H. Tsai, M.C. Billone, and B.A. Hilton, “Examination of Spent Pressurized Water Reactor Fuel Rods after 15 Years in Dry Storage,” Nuclear Technology, Vol. 144, Nov. 2003, pp. 186-200.	Peak temperatures of over 400°C were attained for this benchmark test. Very little creep was observed. No additional fission gas release was detected. No evidence of hydrogen uptake or hydride reorientation was found, Little if any cladding annealing occurred during the prestorage performance or extended-storage periods.
120	Einziger, R. E., L. Thomas, H. C. Buchanan, R. Stout, "Oxidation of Spent Fuel in Air at 175 to 195°," Presented at the Third Annual International High-Level Radioactive Waste Management Conference, Las Vegas, Nevada, April 12-16,1991, pp. 1449-1457, revised for publication, Journal of Nuclear Materials, Vol. 190, 1992, pp. 53-60.	Preliminary data indicates quasi-stable formation of U_4O_{9+X} at O/M = 2.4 in air at 175-195°C. Further oxidation to U_3O_8 would be very slow taking over 2000 years at 95°C.
121	Einziger, R. E., M. A. McKinnon, and A. J. Machiels, Extending Dry Storage of Spent LWR Fuel For Up To 100 Years, ANL/CMT/CP-96494, International Symposium on Storage of Spent Fuel From Power Reactors,Vienna, Austria, November 1998.	Points to temperature limits for stress-driven cladding degradation mechanisms with concerns for higher burnup fuel (>45GWd/MTU). Discusses relevant conditions for dry storage over 100 year period.
122	Einziger, R. E., S. D. Atkin, D. E. Stellbrecht, and V. Pasupathi, “High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods Under Dry Storage Conditions.” Nuclear Technology, Vol. 57, April 1982, pp. 65-80.	No cladding breaches after a year of dry storage at 482, 510, and 571°C in inert/limited air atmosphere. This demonstrates how conservative Blackburn's analyses (previous performance model, 1978) are for clad lifetime prediction.

123	EINZIGER, R.E., STRAIN, R., Behavior of Breached PWR Spent-Fuel Rods in an Air Atmosphere Between 250 and 360°C, Nuclear Technology, Vol. 75, October 1986, pp. 82-95.	Defects in fuel rod segments with lower burnup propagated sooner than those with higher burnup from the same parent rod. Breached PWR fuel rods will not split open from fuel oxidation during 100 years oof of storage if the rod is not exposed to air until the temperature drops below 230°C. Defect shape may be more important than size in determining time-to-cladding-splitting.
124	Elias, E., and C. B. Johnson, Radiological Impact of Clad and Containment Failures in At-Reactor Spent-Fuel-Storage Facilities, Research Project 2062-1 Final Report, EPRI-NP-2716, DE83 900514, Electric Power Research Institute, Palo Alto, California, October 1982.	Analyzed failure scenarios for dry cask storage systems. Krypton-85 was found to be the limiting isotope for dose at the site boundary. Taking no credit for the cask confinement barrier, cladding failure at 7% per year (far in excess of expected failure rate) could occur before causing a 2 mrad beta dose at 1 km from the storage site. (This represents ~10% of the off-site dose limit in 10 CFR 50.)
125	Enderlin, V. R., and K. D. Domina, Update on Spent Nuclear Fuel Retrieval at the DOE Hanford Site, HNF-7785, Rev. 0, March 2001.	Status report. 2100 metric tons of fuel moved to new dry storage facility as of March 2001.
126	Environmental Assessment -- Test Area North pool stabilization project update, DOE-EA-1217, Idaho National Engineering Laboratory, August 1997.	Basic description to support FONSI.
127	Eslinger, L. E., and R. C. Schmitt, "Spent-Fuel Storage Cask Testing and Operational Experience at the Idaho National Engineering Laboratory," EGG-M-89478, Transactions of American Nuclear Society, Vol. 60, No. 2, November 1989, p. 124.	Discusses the Castor V/21, Transnuclear TN-24P, and Westinghouse MC-10 Commercial fuel casks at Test Area North (TAN).
128	Farhataziz, and M. A. J. Rodgers, Radiation Chemistry Principles and Applications, VCH Publishers, Inc., New York, 1987.	Summarizes some of the earlier work as addressed in [21] and [97] and goes on to consider radiation processing and sterilization.
129	Ferry, C., C. Poinssot, C. Cappelaere, L. Desgranges, C. Jegou, F. Miserque, J.P. Piron, D. Roudil, and J.M. Gras, "Specific Outcomes of the Research on the Spent Fuel Long-term Evolution in Interim Dry Storage and Deep Geological Disposal," Journal of Nuclear Materials, Vol. 352, 2006, pp. 246-253.	Alpha radiolysis accelerates fuel dissolution in anoxic conditions, the effect may be counteracted by other environmental conditions like hydrogen production (from canister corrosion). Raises the question of the long-term effect of daughter product helium on grain boundary stability.
130	Flaherty, J. E., A. Fujita, C. P. Deltete, and G. J. Quinn, A Calculation Technique to Predict Combustible Gas Generation in Sealed Radioactive Waste Containers, GEND-041, UC-78, DE86 011395, May 1986.	Develops the calculation and compares to measured gas generation in dewatered TMI-2 canisters prior to shipment.

131	Frost, C. R., A. K. Miller, A. K., M. Brooks, T. Y. Cheung, A. Tasooji, J. C. Wood, J. R. Kelm, and B. A. Surette, Estimates of Zircaloy Integrity During Dry Storage of Spent Nuclear Fuel, EPRI NP-6387, Research Project 2062-9, Stanford University, Atomic Energyof Canada Ltd., and Ontario Hydro. Electric Power Research Institute, Palo Alto, California, May 1989.	Considers cladding failure mechanisms and estimates probability of failure over a 100 year storage period with temperature in air or inert atmosphere. Assesses maximum temperature limitations in the context of this failure probability plot and the dominant failure mechanisms for these considerations. Assumptions are conservative: higher than typical fission gas release, 10% availability of iodine from CsI at inner clad surface, isothermal temperature profile, and no decrease in hoop stress from creep.
132	Fukuda, K., W. Danker, J. S. Lee, A. Bonne, and M. J. Crijns, IAEA Overview of Global Spent Fuel Storage, IAEA-CN-102/60, 2003.	High level document - no detail on drying.
133	Funk, C. W., and L. D. Jacobson, Spent Fuel Integrity During Transportation, HEDL-TME78-58, UC-85, Hanford Engineering Development Laboratory, Richland, Washington, January 1980.	Considers failure mechanisms in transport, but does not successfully confirm or refute any failed rods. Observes that radioactivity generally increases during transportation.
134	Funke, Th., and Ch. Henig, CASTOR® 1000/19: Development and Design of a New Transport and Storage Cask, International Youth Nuclear Congress, IYNC 2008, Paper No. 216, Interlaken, Switzerland, 20-26 September, 2008.	German dual use transportation and storage cask pursuing license for use in Czech Republic for Temelin NPP fuel. Dewater, vacuum dry, then helium backfill.
135	Further analysis of extended storage of spent fuel, Final Report of a Co-ordinated Research Programme on the Behaviour of Spent Fuel Assemblies during Extended Storage (BEFAST III) 1991-1996, IAEA-TECDOC-944, International Atomic Energy Agency, Vienna, Austria, May 1997.	Tabular summary of international experience with dry storage. Dry storage technology has matured since the inception of the BEFAST program and emphasis is shifting from dry storage and capacity for the back end of the fuel cycle to accommodation of higher burnup fuel. While dry storage experience has been positive for more than two decades, extrapolation to extended storage (>50 year period) has yet to be confirmed.
136	Garvin, L. J., Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Report, HNF-SD-SNF-SP-012, Rev. 0, EDT/ECN 607687, May 1997.	Guidance to allow DOE to satisfy NRC regulatory equivalence for conditioning of spent nuclear fuel in accordance with DOE policy (Grumbly, Memorandum EM-36-3.1.6.7, 1995). Gives tables of items to be addressed.
137	Garvin, L. J., Preliminary Safety Evaluation For The Spent Nuclear Fuel Project's Cold Vacuum Drying System, WHC-SD-SNF-PSE-003, Rev. 0, ECN 607673, May 1996.	Documentation to support drying and MCO loading operations.
138	Garvin, L. J., Spent Nuclear Fuel Project - Criteria Document Cold Vacuum Drying Facility Phase 2 Safety Analysis Report, SNF-2983, Rev. 0, July 1998.	Explains how items identified in Ref. [136] are to be addressed in the SNF CVDF Phase 2 SAR.

139	Garvin, L. J., Spent Nuclear Fuel Project Multi-Canister Overpack, Additional NRC Requiremens, HNF-SD-SNF-DB-005, Rev. 3, ECN 630498, September 1998.	See also Reference [136]
140	George, A. H., J. Rogers, and J. Fleming, Vacuum Drying of Fuel Containers, prepared for D. L. Peters, Idaho National Engineering Laboratory, Subcontract No. C85-110754, Task Order 34, 1 June 1993 - 14 January 1994.	Montana State University vacuum drying design and equipment testing. Includes data for controlled leaks and residual water as detected and dried through holes of varying diameters.
141	Geuther, W. J., Removal Plan for Shippingport Pressurized Water Reactor Core 2 Blanket Fuel Assemblies from T Plant to the Canister Storage Building, Los Alamos-Technical Associates, Inc., for Westinghouse Hanford Company, EDT 615201, WHC-SD-WM-ES-394, Rev. 0, September 1996.	Identifies dryness criteria (3 torr for 60 minutes) and estimate of liquid water initially present. Mentions dip tube use for draining loaded canister. The amount of oxygen in the T Plant Canister will be less than 5 volume percent of the gases contained in the gaseous phase to prevent hydrogen deflagration. The vacuum drying system will be required to measure oxygen content within the T Plant Canister to determine when the described limit has been reached.
142	Gibson, G. W., K. C. Sumpter, M. Zukor, and D. W. Knight, Plates and Elements Made With Powdered UAl, and Fabricated by Atomics International, CI-1025, Idaho Nuclear Corporation, January 1967.	Details ATR fuel element parameters and illustrates modes of failure.
143	Gilbert, E. R., C. A. Knox, and G. D. White, "Assessment of Nitrogen as an Atmosphere for Dry Storage of Spent LWR Fuel," PNL-5569, UC-85, DE86 001944, Pacific Northwest Laboratories, Richland, WA, September 1985.	Examines reactions and performance of nonirradiated fuel pellets and UO_2 spent fuel fragments over 7 weeks of tests at up to 380°C in an external gamma field in storage in nitrogen. Negligible weight change and no changes in physical appearance were observed.
144	Gilbert, E. R., W. J. Bailey, A. B. Johnson, Jr., and M. A. McKinnon, Advances in Technology for Storing Light Water Reactor Spent Fuel, Nuclear Technology, Vol. 89, February 1990, pp. 141-161.	Refers to 10CFR72 for requirement that gross degradation of fuel and release of radioactive particulates (fuel through cladding defects or crud dislodged from surfaces) be prevented. Temperature limited such that cladding breach by stress rupture expected to be <0.5% of the spent fuel rods. Provides overview of concerns and international dry storage experience to date.
145	Gilbert, E.R., C. E. Beyer, E. P., Simonen, P. G. Medvedev, “Update of CSFM creep and creep rupture models for determining temperature limits for dry storage of spent fuel”, paper presented at Embedded Topical Meeting, International Congress on Advanced Nuclear Power Plants, Hollywood, Florida, 2002.	Method to determine storage temperature limit based on creep deformation. Data suggest creep rate for some mechanisms may be reduced in irradiated cladding. Large variability in (unirradiated) data apparently due to manufacturing variations. Relatively fast strain rate data suggest that high oxide and hydride layers in high burnup cladding cannot bear load.

146	Glang, R., Chapter 1, Vacuum Evaporation, Handbook of Thin Film Technology, L. I. Maissel and R. Glang, Eds., McGraw-Hill, New York, 1970.	General text on drying.
147	Goldmann, L. H., J. J. Irwin, and C. R. Miska, The Hanford Spent Nuclear Metal Fuel Multi-Canister Overpack and Vacuum Drying & Hot Conditioning Process, WHC-SA-3097-FP, SPECTRUM `96: international conference on nuclear and hazardous waste management, May 1996.	MCO dimensions, fuel cleaning, loading, cold vacuum drying, hot vacuum conditioning, passivation, storage (and final disposition).
148	Goll, W., H. Spilker and E. H. Toscano, "Short-Term Creep and Rupture Tests on High Burnup Fuel Rod Cladding," Journal of Nuclear Materials, Vol. 289, 2001, pp. 247-254.	Tests at 573 and 643 K and hoop stresses of 400 and 600 MPa followed by 5 days at 423 K and 100 MPa. Tests showed at ~600 K, uniform plastic strain of at least 2% is reached without cladding failure.
149	Gorpani, B., Assembly Transfer System Description Document, SDD-ATS-SE-000001 REV 00, ICN 01, June 26, 2000.	References and interprets NUREG-1536 to reguire "The system shall dry SNF assemblies to a combined concentration of less than or equal to 0.25 volume percent for 0 2 , H20, C02, and CO (TBV-094) for a loaded disposal container."
150	Graedel, T. E., "Corrosion Mechanisms for Aluminum Exposed to the Atmosphere," Journal of the Electrochemical Society, Vol. 136, No. 4, April 1989, pp. 204C-212C.	Discusses aluminum corrosion and adsorbed and absorbed water.
151	Gravenor, J. G., A. Blanchard, D. S. Kendall, and P. A. Jackson, Post-irradiation testing of AGR element components in support of the Scottish Nuclear Limited dry-store project, Paper #22, Fuel management and handling, Proceedings of the international conference organized by the British Nuclear Energy Society, Edinburgh, 20-22 March 1995.	Drying AGR graphite fuel for interim storage using 200C flowing hot air or argon. Irradiated fuel tested (1 pin at a time) to assess activity release.
152	Gray, W. J., and R. E. Einziger, Initial Results from Dissolution Rate Testing of N-Reactor Spent Fuel Over a Range of Potential Geologic Repository Aqueous Conditions, DOE/SNF/REP-022, Rev. 0, PNNL-11894, UC-802, April 1998.	Intrinsic solubility of N-Reactor fuel is higher than LWR fuel under the conditions tested. However, N-Reactor fuel is expected to have 100 times lower exposed surface area than LWR, so dissolution rates would be of the same order of magnitude on a per mass basis. Burnup of N-Reactor fuel is about ten times less than LWR fuel, so radionuclide release rates for N-Reactor fuel would be about 10% of that for LWR fuel. Radionuclide inventory varies greatly with location within the fuel rod, so care must be taken in intrepreting results and to provide like for like comparisons.

153	Gruss, K. A., and M. W. Hodges, Regulatory issues associated with the dry storage and transportation of high burnup fuel, U.S. Nuclear Regulatory Commission, INIS-FR--807, OSTI ID: 20241550, July 1, 2001.	Estimates cask loading capacity and time and temperature limitations for drying for licensed casks.
154	Gruss, K. A., G. Hornseth, and M. W. Hodges, U.S. Nuclear Regulatory Commission acceptance criteria and cladding considerations for the dry storage of spent fuel, IAEA-CN-102/55, International conference on storage of spent fuel from power reactors, July 2003, pp. 229-239.	Refers to ISG-11, Rev 2, stating a limit of 400C for normal conditions of storage and vacuum drying in the context of retaining the integrity of SNF geometry within analyzed configuration(s). Drying is only addressed in the context of potential damage (primarily to the cladding) that could affect the licensing basis.
155	Guenther, R. J., A. B. Johnson, Jr., A. L. Lund, E. R. Gilbert, S. P. Pednekar, F. M. Berting, L. L. Burger, S. A. Bryan, and T. M. Orlando, Initial Evaluation of Dry Storage Issues for Spent Nuclear Fuels in Wet Storage at the Idaho Chemical Processing Plant, INEL-96/0140, November 1994.	PNL review of INL issues, worst case failed fuels in antiquated wet storage: degradation mechanisms by fuel or cladding type, waterlogged elements, discusses (remotely relevant) existing regulation.
156	Guenther, R. J., Results of Simulated Abnormal Heating Events for Full-length Nuclear Fuel Rods, PNL-4555, UC-85, January 1983.	Considered overheating in a dry storage cask environment and looked at rod temperatures and internal pressures. Metallography of the cladding provided data on grain growth, hydriding, oxidation, cladding stresses, and the general nature of the failures. Shows the importance of heating rate on cladding microstructure and performance.
157	Gustov, V., G. Korotkov, E. Barnes, R. Snipes, Improvement of operational safety of dual-purpose transport packaging set for naval SNF in storage, ICEM '07, International Conference on Environmental Remediation and Radioactive Waste Management, Bruges, Belgium, 2-6 September 2007.	Abstract only. Cask TUK-108/1 fulfillment of safety requirements. Results of tests of pilot facility at Far Eastern plant Zvezda.
158	Hands, B. J. A Summary of BNFL's Experience in the Storage And Handing of Spent Metallic Fuel, Addendum: Summary of Reference Paper Contents, RD Department Report 0326, Rev. 2, Client in Confidence, Contract Number MRY-SWV-383520, BNFL UK Group, April 1995.	This item offers an annotated bibliography of various references associated with uranium fires and uranium corrosion from the international nuclear community, primarily the UK. To the extent that the source documents could be found and reviewed (many appear to be internal or unpublished works), they are included in this listing as well.

159	Hanson, B. D., The Burnup Dependence of Light Water Reactor Spent Fuel Oxidation, PNNL-11929, July 1998.	TGA studies examine the formation of U_3O_8 from UO_2 and the relative speed of dissolution for release considerations in a repository environment. Accounts for at least half of the burnup dependence of reaction rate with the impurity substitutions in the lattice structure (as shown by oxidation of doped unirradiated UO_2). Concluded that burnup is not a well-defined characteristic with respect to oxidation rate, because of variations in fission products with burnup.
160	Hayes, T. A., R.S. Rosen, and M. E. Kassner, Critical Analysis of Dry Storage Temperature Limits for Zircaloy-Clad Spent Nuclear Fuel Based on Diffusion Controlled Cavity Growth, UCRL-ID-131098, TIC: 254551, 164598, Lawrence Livermore National Laboratory, December 1999.	Corrosion mechanisms and temperature attained with drying.
161	Hazelton, R. F., Characteristics of Fuel Crud and Its Impact on Storage, Handling, and Shipment of Spent Fuel. PNL-6273, UC-85, DE88 000914, Pacific Northwest Laboratory, Richland, Washington, September 1987.	Crud is a contamination and radiological exposure concern resulting from in-service corrosion deposited on the fuel. While in one case, BWR fuel from West Valley left residual crud in a shipping cask (even with extensive flushing), generally crud problems are manageable. Tests did not produce excess spallation of crud from dry stored rods.
162	Hermann, A., H. Wiese, R. Bühner, M. Steinemann, and G. Bart, Hydrogen Distribution Between Fuel Cladding Metal and Overlying Corrosion Layers, (Proceedings of the International Topical Meeting on Light Water Reactor Fuel Performance, American Nuclear Society, Park City, Utah, April 10-13, 2000.)?	Presents an improved technique for assesssing the hydrogen concentration in the cladding. Shows how the fraction of hydrogen in the oxide layer can obscure (lead to the overestimation) of hydrogen in the metal.
163	Hilliard, R. K., *Effect of Heating Irradiated Uranium: A Literature Survey,* Hanford Atomic Products Operations, HW-52753, Reactors-Production C-42 M-3679, 20th Ed., Rev. 1, DE88 008448, November 1, 1957.	Mechanisms for uranium reactions with air and water are not well defined as yet. Interest for this effort was radionuclide release under accident conditions.

164	Hillner, E., Corrosion and Hydriding Performance Evaluation of Three Zircaloy-2 Clad Fuel Assemblies After Continuous Exposure in PWR Cores 1 and 2 at Shippingport, Pa., WAPD-TM-1412, Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania, January 1980, Addendum December 1983.	Examines post-irradiation corrosion rates on zircaloy cladding. Maximum corrosion product was found on a bundle exposed to slightly lower temperatures but greater fast flux. Hydrogen pickup varies with time and is not a constant fractio of oxide thickness. Additional testing documented in the Addendum indicates that the hydrogen pickup does not vary with neutron exposure and suggests that some of the earlier data may be anomalous.
165	Hilton, B. A., Review of Oxidation Rates of DOE Spent Nuclear Fuel, Part 1: Metallic Fuel, DOE/SNF/REP-054, Rev. 0, September 2000, Argonne National Laboratory, ANL-00124, November 2000.	Oxidation mechanisms and rates associated with (uranium) metal fuel.
166	Hilton, D. A., Some Observations on the Effect of Internal Surfaces on the Ignition Behavior of Irradiated Uranium in Carbon Dioxide, Journal of Nuclear Materials, Vol. 40, 1971, pp. 116-119.	Uranium samples with Irradiation induced swelling appear to deviate from the normal Arrhenius type rate dependency between external specific surface area and temperature.
167	Hodges, B. W., Ed., J. T. Collopy, D. L. Faber, B. W. Hodges, D. R. Roof, E. H. Schwer, C. J. Simon, and K. V. Smith, Preparation of LWBR Spent Fuel for Shipment to ICPP for Long Term Storage (LWBR Development Program), WAPD-TM-1601, Bettis Atomic Power Laboratory, October 1987.	Liner blowdown, vacuum drying, neon backfill, pressure and leak test. Includes process lessons learned from dewatering efforts.
168	Holtec International, HI-STORM 100 Final Safety Analysis Report, Rev. 9, Appendix 2.B The Forced Helium Dehydration (FHD) System, February 13, 2010.	Describes the design criteria and basic implementation of forced helium drying.
169	Holtec International, Holtec Highlights, A summary Report to Our Clients, Suppliers, and Company Personnel, HH 25.16, December 16, 2010.	Makes claims and provides contact information regarding forced helium drying. Mostly a promotional brochure.
170	Huang, F. F., Release of Water Trapped in Damaged Fuel Subsurface Voids, HNF-2191, Rev. 0, EDT/ECN 620804, March 1998.	Water removal flow models for water trapped in waterlogged elements.
171	Huang, F. H., and L. A. Lawrence, Pressurization of whole element canister during staging, HNF-2047, Rev. 0, EDT/ECN 615131, January 1998.	Experimental data to support N-reactor fuel drying and MCO loading operations.
172	Huang, F. H., Container materials in environments of corroded spent nuclear fuel, Journal of Nuclear Materials, Vol. 231, 1996, pp. 74-82.	An overview of the related issues in the context of canister material selection (Type 304L SS).

173	Huang, F. H., Studies on Dry Storage of Damaged Spent Nuclear Fuel, Fluor Daniel Northwest, Inc., ISBN-0-9646655-7-3, Avante Publishing, Richland, Washington, 1998.	Details testing and technical considerations leading up to dry storage of N Reactor fuel in MCOs.
174	Icenhour, A. S., and L. M. Toth, Alpha Radiolysis of Sorbed Water on Uranium Oxides and Uranium Oxyfluorides, Nuclear Science and Technology Division, Oak Ridge National Laboratory, ORNL/TM-2003/172, September 2003.	This companion study to [175] addresses the effects of alpha radiation on adsorbed water. The overall equilibrium mitigates pressure buildup in a closed system. Extrordinary measures are not needed to remove adsorbed moisture from the oxide.
175	Icenhour, A. S., L. M. Toth, and H. Luo, Water Sorption and Gamma Radiolysis Studies for Uranium Oxides, Nuclear Science and Technology Division, Oak Ridge National Laboratory, ORNL/TM-2001/59, February 2002.	Examines water adsorbed/retained (in increasing quantities) with UO_2, U_3O_8, and UO_3. Complete removal at 650°C is unnecessary. System is tolerant of quantities (several percent sorbed moisture) remaining after drying at 150°C. Observes only slight pressure increase in gamma field due to radiolytic production of hydrogen - and system at least partly compensated with a simultaneous loss of oxygen pressure. Back reactions were also clearly in evidence.
176	Icenhour, A. S., L. M. Toth, and H. Luo, Water Sorption and Gamma Radiolysis Studies for Uranium Oxides, Nuclear Technology, Vol. 147, August 2004, pp. 258-268.	Demonstrates that efforts to remove all traces of moisture from U_3O_8 are not necessary. Some H_2 was produced, but O_2 was depleted from the initial atmosphere leading to and overall pressure decrease, even for samples with 9 wt. % moisture. Back reactions were evident limiting product gas pressure in the system.
177	Ioltukhovskiy, A. G., A.V. Vatulin, I.M. Kadarmetov, N.B. Sokolov, and V.P. Velioukhanov, Validation of dry storage modes for RBMK-1000 spent fuel assembles (SFA), IAEA-CN-102/39, Storage of Spent Fuel from Power Reactors, 2003 Conference Proceedings, organized by the International Atomic Energy Agency in co-operation with the OECD Nuclear Energy Agency, Vienna, Austria, October 2003, pp. 422-430.	Discusses degradation mechanisms including cladding oxidation, creep, "hydrogen pickup", delayed hydride cracking (DHC), corrosion effected cracking of steel, and expansion of UO2 with its oxidation to U3O8 (at temperatures above 200C).
178	Ioltukhovskiy, A. G., A.V. Vatulin, I.M. Kadarmetov, N.B. Sokolov, and V.P. Velioukhanov, Validation of dry storage modes for RBMK-1000 spent fuel assembles (SFA), IAEA-CN-102/39, Storage of Spent Fuel from Power Reactors, 2003 Conference Proceedings, org	Discusses degradation mechanisms including cladding oxidation, creep, "hydrogen pickup", delayed hydride cracking (DHC), corrosion effected cracking of steel, and expansion of UO2 with its oxidation to U3O8 (at temperatures above 200C).

179	Irwin, J. J., C. R. Miska, C. C. Pitkoff, and R. Whitehurst, Spent Nuclear Fuel Project Cold Vacuum Drying Facility Supporting Data and Calculation Database, SNF-3001, Rev. 0, EDT/ECN 625782, February 1999.	Documentation to support drying and MCO loading operations.
180	Irwin, J. J., C. R. Miska, S. A. Brisbin, C. C. Pitkoff, and R. Whitehurst, SNF Project Cold Vacuum Drying Facility Operations Manual, SNF-2356, Rev. 2, EDT/ECN 653480, April 1999.	Step by step procedure. Time constraints and estimates for drying.
181	Irwin, J. J., D. M. Ogden, B. C. Fryer, and M. J. Thurgood, Spent Nuclear Fuel Vacuum Drying Thermal-Hydraulic Analysis and Dynamic Model Development Status Report, WHC-SD-WM-ER-607, Rev. 0, EDT/ECN 614687, March 1995.	Documentation to support drying and MCO loading operations.
182	Irwin, J. J., Spent Nuclear Fuel Project Cold Vacuum Drying Facility Vacuum and Purge System Design Description, SNF-3062, Rev. 0, EDT/ECN 625179, August 1998.	Includes list of review comments.
183	Itooka, S., T. Yokoyama, and M. Kato, Study on Evaluation of Contents Integrity Stored in Dry Cask During Interim Storage, Paper No. 576, Proceedings of Global 2005, Tsukuba, Japan, October 9-13, 2005	Vacuum drying LWR fuel with inert backfill. Concerns for cladding creep, hydride reorientation, annealing of aluminum basket/internals, and exernal load (handling accident).
184	J. Henscheid, S. S. Kim, T. Larson, R. Lords, D. Marts, T. Criddle, and P. Winston, Feasibility Study on Appropriate Technologies for Storage and Disposition of Spent Nuclear Fuel from Soviet-Designed Research Reactors, CCN 30166, S. M. Modro to I. Bolshynski, INEEL National Security Programs Counterterrorism and Law Enforcement, February 21, 2002.	Highlights US drying and storage technologies relevant to specific soviet fuels. Summarizes drying considerations.
185	J. W. McWhirter and J. E. Draley, Aqueous Corrosion of Uranium and Alloys: Survey of Project Literature, ANL-4862, Metallurgy and Ceramics, 26771, Argonne National Laboratory, Chicago, Illinois, May 14, 1952.	Tests at 50-226°C were conducted in boiling distilled water to produce an oxidation rate expression. Aerated water showed slower initial rate than hydrogen saturated water. Alloys were tested in an effort to identify a more corrosion resistant uranium. Heat treatment may also improve performance.
186	J.C. Crepeau, S. Reese, H.M. McIlroy, and R.E. Lords, "Drying of Mock Spent Nuclear Fuel Elements," *Drying Technology*, Vol. 16, No. 3-5, pp. 545-560, 1998.	Presents an overview of simulated drying studies performed with IFSF Canning Station mockup.
187	Jackson, P. A., The conditioning of AGR fuel elements for long term storage, Paper #21, Fuel management and handling, Proceedings of the international conference organized by the British Nuclear Energy Society, Edinburgh, 20-22 March 1995.	Modular dry vault storage with inert backfill for 100 year interim. Expects to retain fuel integrity. Discusses how dry is dry and process considerations. Recommends induction heating.

188	Johnson, A. B, Jr., A. L. Lund, and S. P. Pedneker, Estimates of Durability of TMI-2 Core Debris Storage Canisters and Cask Liners, PNL-9457, UC-804, April 1994.	Suggests a conservative estimate of stainless steel (Type 304L, 219 mils thick at the thinnest) performance under relatively benign conditions from performance under more aggressive corrosion conditions. The expectation is for no pinhole penetration and loss of up to 50% wall thickness to general corrosion over 4600 years in water or 11,500 years in air. By comparison the (Type 1020, 750 mils thick) carbon steel liners are likely to see pentration in 750 years in water and loss of 50% wall thickness in 375 years in water or 1900 years or 3800 years in air (with 40% or 10% relative humidity respectively).
189	Johnson, A. B., Jr., and E. R. Gilbert, "Reaction of Fuel Cladding with Cover Gases Under Dry Storage Conditions," Proceedings of Spent Fuel /Cladding Reaction During Dry Storage, NUREG ICP-0049, Gaithersburg, Maryland, 1983.	General survey of oxidation behavior of cladding materials, including zircaloy, inconel and stainless steel, as relevant to storage in air, argon, carbon dioxide, helium, or nitrogen.
190	Johnson, A. B., Jr., E. R. Gilbert, and J. C. Dobbins, "Evaluation of a PWR Fuel Assembly Subject to Air Storage," in "Proceedings of the Third International Spent Fuel Storage Technology Symposium/Workshop," Conference-860417, Pacific Northwest Laboratories, Richland, WA, PNL-SA-13878, April 1986, pp. W55-W62.	Results of post-test fuel examination (after 24 months) of dry storage for one PWR fuel assembly in air (with temperature declining from 275°C) at EMAD (Nevada Test Site). One rod of the assembly was breached 2 months into the test.
191	Johnson, A. B., Jr., E. R. Gilbert, and R. J. Guenther. 1983. Behavior of spent nuclear fuel and storage system components in dry interim storage. PNL-4189, Rev. 1, Pacific Northwest Laboratory, Richland, Washington, February 1983.	The US, UK, Canada, and Germany each had some manner of dry storage. Identifies dry storage for extended management of spent fuel as going back to 1964. Discusses storage technology, fuel performance, and the characteristics of LWR fuel relevant to dry storage.
192	Johnson, A. B., Jr., E. R. Gilbert, D. R. Oden, D. L. Stidham, J. E. Garnier, D. L. Weeks, and J. C. Dobbins, "Simulated Dry Storage of a Spent PWR Nuclear Fuel Assembly in Air," Proceedings of Waste Management '85, Vol 1, 1985, pp. 513-519.	Gas analysis of a Turkey Point PWR rod in simulated dry storage conditions (air at 200-275°C) appeared to leak during the second month of the test.

193	Johnson, A. B., Jr., J. C. Dobbins, and F. R. Zaloudek, "Assessment of Integrity of Spent Fuel Assemblies Used in Dry Storage Demonstration at the Nevada Test Site," Pacific Northwest Laboratories, PNL-6207, UC-85, DE87 014699, Richland, Washington, July 1987.	Cover gas monitoring for krypton-85 provided a sensitive measure of fuel rod integrity. Detects "background" krypton-85 from (slow) surface desorption after (in reactor) exposure to leak(s). New leaks detected at orders of magnitude higher levels. Of 3468 PWR rods tested only one failed. Companion rods which had been overheated up to 480-570°C did not leak.
194	Johnson, A. B., Jr., mpacts of Reactor-Induced Defects on Spent Fuel Storage, Battelle, Pacific Northwest Laboratories, Richland, Washington, 1978?, pp. 235-247.	Promotes dry storage surveillance as the interim for dry storage expands.
195	Johnson, A. B., Jr., W. J. Bailey, and E. R. Gilbert, Materials Behavior In Interim storage of Spent Fuel, PNL-SA-10553, American Nuclear Society Transactions, Materials Science and Technology, Vol. 43, 1982, pp. 314-315.	Mentions early demostration for bounding dry storage conditions (in air and inert gas), for max temperature ~400°C. Main concern for air storage was oxidation of UO_2 at cladding defects that developed in reactor.
196	Johnson, D. M., Performance Specification Fuel Drying and Canister Inerting System for Shippingport Pressurized Water Reactor (PWR) Core 2 Blanket Fuel Assemblies Stored Within Shippingport Spent Fuel Canisters, HNF-3043, Rev. 1, ECN 657936, February 2000.	Potentially useful set of applicable documents and references listed. Specifically calls out drying endpoint determination as described in NUREG-1536. Canister inerting to be provided with helium backfill.
197	Johnson, D. M., Shippingport Spent Fuel Canister System Description, SNF-5809, Rev. 1, Hanford, June 2001.	Vacuum drying followed by helium backfill. See also [196]. Expands calculations from [217] PNL-6365 to apply to LWR fuel.
198	Johnson, Jr., A. B., and E. R. Gilbert, Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases, PNL-4835, UC-85, Pacific Northwest Laboratory, September 1983.	Analysis of PWR dry storage indicating that storage in inert gas environment precludes cladding degradation mechanisms.
199	K. Hayashi, K. Iwasa, K. Araki, and R. Asano, Developing New Transportable Storage Casks for Interim Dry Storage, 14th International Symposium on the Packaging and Paper # 047Transportation of Radioactive Materials (PATRAM 2004), Berlin, Germany, September 2004.	Hitz casks design. Vacuum drying with helium backfill.
200	K. Vinjamuri, K. and R. R. Hobbins, Aqueous Corrosion of Uranium Aluminide Fuel, Nuclear Technology, Vol. 62, August 1983, pp. 145-150.	Examination of failed ATR fuel: pitting corrosion and erosion of the UAl_x. Other failure modes may include blistering due to excessive fission gas buildup or buckling due to thermal stress or irradiation growth stress.

201	Kadarmetov, I. M., Behaviour of Spent WWER Fuel under Long Term Storage Conditions, A. A. Bochvar All-Russia Research Institute of Inorganic Materials, IAEA Regional Training Course on WWER Fuel, Bratislava, Slovakia, 21 June - 2 July, 1999.	Summarizes cladding degradation mechanisms.
202	Katz, J. J., and E. Rabinowitch, The Chemistry of Uranium, The Element, Its Binary and Related Compounds, National Nuclear Energy Series, Dover Publications, Inc., New York, 1951.	This is the seminal work on uranium chemistry.
203	Kaye, C. J., Conditioning of metallic Magnox fuel element debris, IAEA-SM-261/33, Conditioning of radioactive wastes for storage and disposal. Proceedings of an international symposium organized by the IAEA, the CEC and the OECD NEA and held in Utrecht, the Netherlands, 21-25 June 1982.	Prep for sea disposal. Activated contaminants not fuel or cladding per se. Heat conduction drying of debris prior to sorting is proposed.
204	Keese, H., W. Anspach, and R. Christ, "The TN 1300 Transport/Storage Cask System," in "Proceedings of 7th International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM 83)," Conference, No. 830528, 1983, pp. 242-247.	Describes the TN 1300 Cask for storage of PWR or BWR fuel.
205	Keese, H., W. Anspach, and R. Christ, "The TN 1300 Transport/Storage Cask System," in "Proceedings of 7th International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM 83)," Conference, No. 830528, 1983, pp. 242-247.	Describes the REA 2023 Cask and auxilliary skid.
206	Keey, R. B., Drying Principles and Practice, International Series of Monographs in Chemical Engineering, Volume 13, Pergamon Press, 1972.	General text on drying.
207	Kelley, J. G., The Myths and Realities of Vacuum Drying, Trivis Inc., NEI Dry Storage Information Forum, Clearwater Beach, Florida, May 15-17, 2007.	Indicates that drying time is independent of heat load.
208	Kenneally, R. M. and J. H. Kessler. "Cooperative Research Program on Dry Cask Storage Characterization," Proceedings of ICONE 8, 8th International Conference on Nuclear Engineering: Baltimore, MD, April 2, 2000.	Examination program to look at VSC-17 and CASTOR V/21 casks and contents in support of license renewal requirements.

209	Kessler, S. F., Criticality Safety Evaluation Report for the Multi-Canister Overpack, HNF-SD-SNF-CSER-005, Rev. 4, ECN 647522, October 1999.	Analysis to support N-reactor fuel drying and MCO loading operations.
210	Kim, Y.S. (2009). "Hydride Reorientation and Delayed Hydride Cracking of Spent Fuel Rods in Dry Storage," Metallurgical and Materials Transactions A, Journal of ASTM International, Vol. 40A, December 2009, pp. 2867-2875.	Examines thermal creep during drying and potential for hydride reorientation and delayed hydride cracking (with cooling). Experimentally demonstrates that prior creep deformation facilitates nucleation of reoriented hydrides in spent fuel rods.
211	Kim, Y.S., "Delayed Hydride Cracking of Spent Fuel Rods in Dry Storage," Journal of Nuclear Materials, Vol. 378, 2008, pp. 30-34.	Predicts cladding failure for rods in dry storage upon cooling to ~180°C if they have stress raisers. Suggests that the gamma to delta hydride phase transformation provides the difference in hydrogen concentration.
212	Kimball, C., and M. Billone, Dry Cask Storage Characterization Project, EPRI TR-1002882, Electric Power Research Institute, Palo Alto, California, September 2002.	Examines fuel from CASTOR V/21 dry cask storage demonstration after 15 years of storage. Gas analysis shows neither signs of air ingress to the container nor cladding failure. No evidence was found to indicate degradation to any of the systems significant to safety. All materials in the cask, including the fuel assemblies appear as they did upon loading in 1985. Detailed characterization was performed on three rods finding 1) very little creep (supporting predictions of <0.1%), 2) no additional fission gas release, 3) no evidence of hydrogen pickup or hydride reorientation, 4) little or no cladding annealing, and 5) residual creep strain of ~6% for 400°C sample after raising the stress level to 250 MPa for ~700 hours, but while creep rate increased the deformation was uniform.
213	Klinger, G. S., B. M. Oliver, J Abrefah, S. C. Marschman, P. J. MacFarlan, and G. A. Ritter, Drying Results of K-Basin Fuel Element 5744U (Run 4), PNNL-11821, UC-602, July 1998.	Drying tests to support N-reactor fuel drying and MCO loading operations.
214	Klinger, G. S., B. M. Oliver, J Abrefah, S. C. Marschman, P. J. MacFarlan, and G. A. Ritter, Spent Fuel Drying System Test Results (First Dry Run), PNNL-11814, UC-602, July 1998.	Drying tests to support N-reactor fuel drying and MCO loading operations.
215	Klinger, G. S., B. M. Oliver, J Abrefah, S. C. Marschman, P. J. MacFarlan, and G. A. Ritter, Spent Fuel Drying System Test Results (Second Dry Run), PNNL-11838, UC-602, July 1998.	Drying tests to support N-reactor fuel drying and MCO loading operations.

216	Klinger, G. S., B. M. Oliver, S. C. Marschman, J. Abrefah, L. R. Greenwood, P. J. MacFarlan, and G. A. Ritter, Drying Results of K-Basin Fuel Element 6603M (Run 5), PNNL-11841, UC-602, September 1999.	Drying tests to support N-reactor fuel drying and MCO loading operations.
217	Knoll, R. W., and E. R. Gilbert, Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel, PNL-6365, UC-85, Pacific Northwest Laboratory, November 1987.	The original basis for NUREG-1536. Makes corrosion calculation based on gas analysis from commercial casks after <1 year of operation loaded with PWR or BWR fuel.
218	Kobayashi, M., Ph.D. Thesis, Northwestern University, Evanston, Illinois, Pulsed-Bed Approach to Fluidization, ANL-7592, Argonne National Laboratory, October 1969.	Studies heat transfer through particulate material.
219	Kohli, R., and V. Pasupathi, Investigation of Water-Logged Spent Fuel Rods Under Dry Storage Conditions, PNL-5987, September 1986.	Damaged BWR and PWR fuel elements tested for moisture retention and rate of release. Neither reactor-induced nor intentionally prepared defects led to water retention that would cause degradation of fuel or storage system components.
220	Kohli, R., and V. Pasupathi, Investigation of Water-Logged Spent Fuel Rods Under Dry Storage Conditions, PNL-SA-13880, CONF-860417, Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986.	Conference paper on same topic as [219].
221	Kohli, R., D. Stahl, V. Pasupathi, A. B. Johnson, Jr., and E. R. Gilbert, "The Behavior of Breached Boiling Water Reactor Fuel Rods on Long-Term Exposure to Air and Argon at 598K," Nuclear Technology Vol. 69, May 1985, pp. 186-197.	Water-logged fuel rods were evacuated, drilled and re-evacuated. 5-10 grams of water were removed. Rods were stored in a capsule with air replenished ~daily. Exposed fuel at the cladding defect oxidized rapidly (at 598K) causing swelling and extending cracks in the cladding. Oxidation was local to the breach with no oxidation found beyond 2.5 cm from the end of the breach.
222	Kondo, T, E. D. Verink, F. H. Beck, and M. G. Fontana, Gas Chromatographic and Gravimetric Studies of Uranium Oxidation Mechanism, Corrosion, Vol. 20, 1964, pp. 314t-320t.	Discusses uranium oxidation kinetics under ambient conditions, including time dependency of the reaction in two phases, surface morphology and hydride formation as a transitory intermediate. Considers oxidation in dry oxygen and dry nitrogen and deduces the involvement of humidity in ambient oxidation.
223	Kondo, T., F. H. Beck, and M. G. Fontana, A Gas Chromatographic Study on the Kinetics of Uranium Oxidation in Moist Environments, Corrosion, NACE, Vol. 30, No. 9, September 1974, pp. 330-339.	Discusses uranium oxidation kinetics under ambient conditions, including surface morphology and hydride formation.

224	Krahn, D. E., Cold Vacuum Drying Facility Hazard Analysis Report, HNF-SD-SNF-HIE-004, Rev. 1, ECN 630494, February 1998.	Documentation to support drying and MCO loading operations.
225	Kuhn, W. L., J. Abrefah, A. L. Pitner, M. Plys, J. P. Sloughter, J. C. Wiborg, and D. W. Damschen, Technical Review of the Characteristics of Spent Nuclear Fuel Scrap, PNNL-13751, December 2001.	More on the same theme as References [261] and [233], includes fuel washing
226	Kühnel, R., R. Bokelmann, M. Scholz, K. Gebke, and M. Schüler, U. S. Patent No. 4,366,114, Dry Storage for Spent Fuel Assemblies, Dec. 28, 1982.	Kraftwerk Union AG, German patent assignee, uses sealed (gas tight) storage boxes. Patent appears to be for storage facility and does not discuss storage box loading or drying other than to include a heat transfer medium suggesting a granular graphite or gas backfill.
227	Kustas F . M., Bates, S. O., Opitz, B. E., Johnson, A. B., Perez, J. M., Farnsworth, R. K., Investigation of the Condition of Spent Fuel Components, PNL-3513, Pacific Northwest Laboratory, Richland, Washington, September 1981.	Materials conditions as a result of underwater storage:Yankee Rowe, Point Beach, Zion, La Crosse, RBOF, and TMI-1 (piping).
228	Lam, P-S., Sindelar, R.L., and Peacock, H.B, Vapor Corrosion of Aluminum Cladding Alloys and Aluminum-Uranium Fuel Materials in Storage Environments, WSRC-TR-97-0120, April 1997, Savannah River Site.	Results and models can be generally applied to predict the degradation of fuel materials. In a sealed system containing finite amounts of water vapor, the rate of corrosion slows with time to less than the parabolic rate for unlimited species. The corrosion reactions stop after water is consumed down to the critical relative humidity level. Corrosion equations allow calculation of hydrogen and its partial pressure in a sealed system. Blistering of the metal matrix is observed at several localized regions and is attributed to hydrogen evolved at high rates.
229	Large, W. S., and R. L. Sindelar, Review of Drying Methods for Spent Nuclear Fuel, WSRC-TR-97-0075, April 1997.	Requirements based overview of DOE SNF drying programs at the time, estimates (and bases) for water content, removal, endpoint dtermination, process duration, potential for subsequent corrosion including hydrogen and hydride generation. Proposes a drying test program and suggests evaluation of hot air drying as an alternative to vacuum drying.
230	Lauer, H., H. Michoux, H. Hüggenberg, K. Janberg, D. Rittscher, and J. M. Renavot, "Super Phenix Spent Fuel Storage Cask, CASTOR SPX," in "Proceedings of 7th International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM 83)," Conference, No. 830528,	Sodium cooled reactor fuel, washed in nearby water pool and transferred to dry cask storage. Relatively high cladding and storage temperature limits had been established (650°C/425°C) to avoid creep.

1983, pp. 238-239.

231	LaVerne, J. A., and L. Tandon, H_2 Production in the Radiolysis of Water on UO_2 and Other Oxides, Notre Dame Radiation Laboratory, NDRL-4453, Journal of Physical Chemisty B, Vol. 107, No. 49, 2003, pp. 13623-13628.	Radiolysis hydrogen yield increases with decreasing layers of adsorbed water on oxides.
232	LaVerne, J. A., H_2 Formation from the Radiolysis of Liquid Water with Zirconia, Notre Dame Radiation Laboratory, NDRL-4572, Journal of Physical Chemisty B Letters, Vol. 109, No. 12, 2005, pp. 5395-5397.	Shows dependence of radiolysis yield on the crystalline form of the zirconia introduced to the water.
233	Lawrence, L. A., B. J. Makenas, R. P. Omberg, D. J. Trimble, R. B. Baker, S. C. Marschman, and J. Abrefah, Characterization of Hanford N-Reactor spent fuel and associated sludges, SPECTRUM '96: international conference on nuclear and hazardous waste management, April 1996, pp. 2453-2458.	Much evidence of fuel corrosion, sludges did not contain pyrophoric constituents, passivation in 2% oxygen in argon improved (raised) ignition temp (corroded fuel ignited at 300°C) but did not return it to that of irradiated uncorroded fuel (ignited at 650°C)
234	Lawrence, L. A., Drying Damaged K-West Fuel Elements (Summary of Whole Element Furnace Runs 1 though 8), HNF-3377, Rev. 0, EDT/ECN 620816, October 1998.	Drying tests to support N-reactor fuel drying and MCO loading operations.
235	Lee, Chan Bock; Chung Chan Lee, Oh Hwan Kim, Jin Gon Chung, and Chong Chul Lee, Analysis of Fuel Behavior During Rod Ejection Accident in the Korea Standard PWR, Korea Atomic Energy Research Institute, 1996?, pp. 626-633.	Failure of high burnup fuel occurring by PCMI (pellet-clad material interaction).
236	Leibowitz, L., L. Baker, Jr., G. Schnizlein, L. W. Mishler, and J. D. Bingle, Burning Velocities of Uranium and Zirconium in Air, Nuclear Science and Engineering, Vol. 15, 1963, pp. 395-403.	Foil and wire strips of uranium and zirconium were burned in air. Propagation velocities matched a similar theory to flame propagation in gases. Variations with thickness and width accommodated by theory.
237	Lessing, P. A., Effects of Water in Canisters Containing DOE Spent Nuclear Fuel, DOE/SNF/REP-017, Rev. 0, October 1998.	Analysis of potential problems associated with residual water in dried stored DOE fuel.

238	Levy, I.S., B.A. Chin, E.P. Simonen, C.E. Beyer, E.R. Gilbert, and A.B. Johnson, Jr. "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas," PNL–6189, UC-85, Pacific Northwest Laboratory, Richland, Washington, May 1987.	Concludes that the earlier recommendation for a single-valued temperature limit of 380°C needs to be replaced by multiple limits (to accommodate fuel and configuration-specific characteristics). Develops the CSFM model and compares results with NRC's DCCG model as well as the German work, with good agreement and explanation for variations.
239	Lietava, P., Licensing Aspects of Spent Fuel Storage as a Function of Timescales, Management of Spent Fuel from Nuclear Power Reactors, IAEA proceedings, Vienna, Austria, 19-22 June 2006, pp. 221-235.	Uses Castor 440/84M Cask.
240	Little B. and P. Wagner, "An Overview of Microbiologically Influenced Corrosion of Metals and Alloys Used in the Storage of Nuclear Wastes." Canadian Journal of Microbiology, Vol. 42, No. 4, Ottawa, Canada: National Research Council of Canada, 1996, pp. 367-374.	Discusses specific mechanisms for microbial effects on corrosion for various metals and alloys. Identifies the presence of water as a necessary factor.
241	Liu, Y. et al., Life Cycle Management of Radioactive Materials Packagings, Transportation, Storage, and Security of Radioactive Material, Vol. 18, No. 4, 2007, pp. 219-225.	Discusses gas generation mechanisms, cladding degradation and temperature limits, and more general aging concerns. Suggests radiofrequency identification (RFID) technology for life cycle package tracking.
242	Liu, Y. Y., High-Burnup Fuel Issues for Long-Term Dry Storage and Transportation, Argonne National Laboratory, DOE/NE Fuel Cycle Technologies Program, Office of Used Fuel Disposition Research and Development, December 2010.	Testing of surrogate pre-hydrided zircaloy cladding samples with hoop stress to determine effects of drying temperature on embrittlement.
243	Liu, Y. Y., Managing Aging Effects on Used Fuel Dry Casks for Long-term Storage, Argonne National Laboratory, DOE/NE Fuel Cycle Technologies Program, Office of Used Fuel Disposition Research and Development, December 2010.	Discusses prevention, mitigation, and monitoring of aging in dry storage.
244	Locke, D. H., The Behaviour of Defective Reactor Fuel, Nuclear Engineering and Design, Vol. 21, 1972, pp. 318-330.	The operating power of the fuel pin is related to the deterioration of the zircaloy cladding (small defect can progress to gross failure), while stainless steel cladding does not indicate the same deterioration mechanism. Hydriding of the zircaloy is implicated in its catastophic failure.
245	Lombard, K. E., A. B. Christensen, and K. M. Wendt, EDF-2727, Rev. 0, Drying Requirements for TMI Canisters D153 and	Internal Report. Examines the logic and likely consequenses of drying or not drying TMI debris canisters including epoxy met mounts.

	D388, July 16, 2001.	
246	Long term storage of spent nuclear fuel — Survey and recommendations, Final report of a co-ordinated research project, 1994-1997, IAEA-TECDOC-1293, International Atomic Energy Agency, Vienna, Austria, May 2002.	Water quality, corrosion, and residual water issues.
247	Longhurst, G. R., Pyrophoricity of Tritium-Storage Bed Materials, EG&G Idaho Report, EGG-FSP-8050, March 1988.	Pyrophoricity experiments on depleted uranium and hydrides. Establishes that risk of fire in a tritium storage bed is real, especially if uranium or zirconium-cobalt are used as storage materials. Fires may not occur until the bed is heated.
248	Lords, R. E., and R. K. McCardell, Technical Basis for Drying Damaged Spent Nuclear Fuel, INEEL/INT-01-01414, October 2001.	Internal Report. Considers the practical and compliance reasons for drying, describes the drying process and endpoint determination, and justifies drying fuel for vented dry storage as a means to reduce the likelihood of ongoing corrosion.
249	Lords, R. E., Editor, J. C. Crepeau, S. Reese, and H. M. McIlroy, Drying Studies of Mock Spent Nuclear Fuels, INEEL/INT-97-01106, September 1997.	Internal Report. Discusses drying tests and bursting tests performed at University of Idaho (Idaho Falls extention) with IFSF Canning Station mockup. No destruction of water-filled aluminum coupons even at the most extreme conditions tested (sudden vacuum after pre-heat to >90C).
250	Lords, R. E., INEEL Process Drying Report, INEEL/INT-2000-01728, September 2000.	Internal Report. Provides a brief description and proposes an approach to drying issues associated with sealed storage and transport of degraded fuel materials.
251	Lords, R. E., Literature Review of the Drying Characteristics of Thorium Spent Fuel Corrosion Products, INEEL/EXT-02-00488, April 2002.	Considers waters of hydration and potential for corrosion products to hold up water during (and after) drying.
252	Lords, R. E., W. E. Windes, J. C. Crepeau, and R. W. Sidwell, Drying Studies for Corroded DOE Aluminum Plate Fuels, INEL-96/00134, CONF-960804--29,	Drying temperature monitored in aluminum plates sandwiched with wet simulated corrosion sediment.
253	Lords, R. E., W. E. Windes, J. C. Crepeau, and R. W. Sidwell, Drying Studies of Simulated DOE Aluminum Plate Fuels, INEL-95/00437, Topical Meeting on DOE Spent Nuclear Fuel and Fissile Material Management, American Nuclear Society, June 1996.	Illustrates the potential for hydrates associated with corrosion products and clay to retain water and obscure endpoint with drying at up to 100C. Qualitative dryness determined visually. The process removed free water but not bound water. The occluded void in a simulated plate showed no damage during drying tests.

254	Loscoe, P. G., Transitioning Metallic Uranium Spent Nuclear Fuel from Wet to Dry Storage, Waste Management 2000, WM'00 Conference, Tuscon, Arizona, February 27 - March 2, 2000.	Provides an overview of issues and resolution of difficulties in preparation for the dry storage of N-Reactor fuel in MCOs.
255	Louthan, M. R., N. C. Iyer, H. B. Peacock, Jr., and R. L. Sindelar, Transitioning Aluminum Clad Spent Fuels from Wet to Interim Dry Storage, WSRC-MS-94-0617, International high-level radioactive waste management conference: progress toward understanding, CONF-950570--12, Las Vegas, Nevada, 1995.	Highlights the corrosive effects of chloride ions on aluminum and the need to mitigate these effects by rinsing aluminum plate fuel before drying/canning for interim dry storage.
256	Lundberg, L. B., Corrosion of Spent ATR Fuel Elements Relative to Their Dr Storage, Internal Technical Report, ERA-NRE-94-096, EG&G Idaho, Inc., August 1994.	Identifies degradation concerns for ATR fuel stored under water in CPP-603 and an absence of dry storage data for ATR fuel. Published data suggests aluminum corrosion rates in air drop by a factor of 100 after periods from 6 months to 2 years. Recommends dry storage tests of longer than 2 years (and up to 10 years).
257	M. K. Adler Flitton, M. K., T. S. Yoder, and P. K. Nagata, The Underground Corrosion of Selected Type 300 Stainless Steels After 34 Years, INL/CON-09-15408, NACE Expo 2009, March 2009.	Steel coupon performance in soil with native microbes.
258	M. R. Louthan, Jr., N. C. Iyer, R. L. Sindelar, and H. B. Peacock, Jr., "Characterization of Aluminum-Clad, Aluminum-Uranium Core Fuels for Interim Dry Storage," in proceedings of International High Level Radioactive Waste Conference at Las Vegas, Nevada, 05/01/95 - 05/05/95, pp. 645-646.	Evaluation of aluminum clad fuel demonstrates that the "clad-core system" should act as a confinement barrier to the release of radionulides even if the cladding has been breached. Direct placement of uncanned fuels into sealed dry storage will provide double confinement without NDE to assure that the clad has not been penetrated.
259	Machiels, A., "Dry Storage of High-Burnup Spent Fuel: Responses to Nuclear Regulatory Commission Requests for Additional Information and Clarification," EPRI TR-1009276, Electric Power Research Institute, November 2003.	Responses are relevant to [356] and [357] in the interpretation and understanding of temperature limitations to preserve cladding integrity.
260	Makenas, B. J., A. J. Schmidt, K. L. Silvers, P. R. Bredt, C. H. Delegard, E. W. Hoppe, J. M. Tingey, A. H. Zacher, T. L. Welsh, and R. B. Baker, Supplementary Information on K-Basin Sludges, HNF-2367, Rev. 0, UC-2070, February 1999.	An update from [416]. Detailed analysis of sludge with consideration for effect on fuel handling.

261	Makenas, B. J., R. P. Omberg, D. J. Trimble, and R. B. Baker, Characterization of Hanford N Reactor spent fuel and K Basin sludges, WHC-SA-2952, CONF-960212-81, prepared for Waste Management '96, American Nuclear Society, Tucson, Arizona, 1996.	Slightly more procedural detail and analysis, fewer but similar conclusions as in Reference [233].
262	Marschman, S. C., J. Abrefah, G. S. Klinger, B. M. Oliver, P. J. MacFarlan, and G. A. Ritter, Drying Results of K-Basin Fuel Element 1990 (Run 1), PNNL-11817, UC-602, June 1998.	Drying tests to support N-reactor fuel drying and MCO loading operations.
263	Marschman, S. C., T. D. Pyecha, and J. Abrefah, Metallographic Examination of Damaged N Reactor Spent Nuclear Fuel Element SFEC5,4378, PNNL-11438, UC-602, Pacific Northwest National Laboratory, Richland, Washington, August 1997.	Observations of hydride inclusions in damaged N-Reactor fuel.
264	Marschman, S. C., T. M. Orlando, T. E. Madey, N. G. Petrik, J. P. Cowin, J. Yarmoff, and R. Dobrozemsky, Radiolytic and Thermal Processes Relevant to Dry Storage of Spent Nuclear Fuels, Final Report, EMSP-60392, U. S. Department of Energy, September 2000.	G-values of adsorbed water appear to be film thickness dependent with thin layers behaving almost like water vapor and thick layers like condensed water. Kinetics of water desorption from (UO3 hydrates) corroded SNF (TGA/DSC/MS). Many progress reports - this claims to be "final".
265	Marshall, R. P., Louthan, M. R., "Tensile Properties of Zircaloy with Oriented Hydrides," Syposium on Zirconium Alloy Development, Transactions of ASM, Vol. 56, 1963, pp. 693 to 700.	Hydride platelets oriented perpendicular to the tensile stress drastically reduced strength and ductility. Parallel oriented platelets had little effect. At and above 50 ppm hydrogen oriented as perpendicular hydrides the zircaloy-4 exhibited no ductility irrespective of the total hydrogen content.
266	Marston, M. J., Commercial Power Experience in Getting Fuel From Wet to Dry Storage, TSSD Services, Inc., September 14, 2010.	Discusses experience in vacuum drying with Connecticut Yankee, Maine Yankee, and Yankee Rowe fuels. No correlation evident between average decay heat and drying time.
267	Matsuo, Y., "Thermal Creep of Zircaloy-4 Cladding Under Internal Pressure," Atomic Energy Society of Japan, Journal of Nuclear Science and Technology, Vol. 24, No. 2, February 1987, pp. 111-119.	Studies creep characteristics of Zircaloy-4 cladding with temperature and hoop stress. Derives equations to model creep. Suggests that self-diffusion of Zr in the Zr-Sn alloy is the control mechanism.

268	McCoy, J. K., Analysis of Degradation Due to Water and Gases in MPC. BB0000000-01717-0200-00005 REV 01. Las Vegas, Nevada: CRWMS M&O. 102829. ACC: MOL.19960419.0202, 1995.	Concludes that MPC dryness requirements are sufficiently stringent for the MPC cavity environment after loading and closure operations. However, they are found to be inconsistent with the interpretation that up to 0.25 volume percent of the interior void volume of the loaded MPC may be filled with liquid water. Suggests changing "residual water content of the MPC interior" to "water content of the gas in the filled MPC" for clarity. The effects of oxidation/reduction reactions, corrosion, hydriding, and radiolysis on the fill gas have been analyzed and found not to produce significant harmful effects. Damage to the package by the fill gas are expected to be negligible.
269	McCracken, K. J., and E. A. Smith, Hanford spent nuclear fuel cold vacuum drying proof of performance test procedure, HNF-2402, Rev. 0, EDT/ECN 606764, June 1998.	Step by step procedure. No conclusions or results - intended to demonstrate design choice function and provide data for initial start up parameters.
270	McCracken, K. J., Hanford Spent Nuclear Fuel Cold Vacuum Drying Test Specification and Test Plan, WHC-SD-SNF-TP-018, Rev. 0, EDT/ECN 600538, June 1998.	Documentation to support drying and MCO loading operations.
271	McCracken, K. J., K. J. Cleveland, and G. A. Ritter, Hanford spent nuclear fuel hot conditioning system test procedure, HNF-SD-SNF-TC-010, Rev. 0, EDT/ECN 606760, September 1997.	Tests to support N-reactor fuel conditioning and MCO loading operations.
272	McCracken, K. J., L. R. Schroeder, and B. C. Fryer, Cold vacuum drying proof of performance (first article testing) test results, HNF-4057, Rev. 0, EDT/ECN 140132, March 1999.	Details basis of duration required and helium use for MCO vacuum drying (removal of free water).
273	McCullum, R., Integrated Used Fuel Management, Industry Perspectives, Nuclear Energy Institute, Nuclear Waste Technical Review Board, Las Vegas, Nevada, www.nwtrb.gov, June 11, 2009.	Recommends centralized interim storage (contractor provided with DOE as the customer).
274	McGillivray, G. W., D. A. Geeson, and R. C. Greenwood, *Studies of the Kinetics and Mechanism of the Oxidation of Uranium by Dry and Moist Air - A Model for Determining the Oxidation Rate Over a Wide Range of Temperatures and Water Vapor Pressures,* Journal of Nuclear Materials, Vol. 208, 1994, pp. 81-97.	A model is developed on the assumption that the surface concentration of water determines the rate of reaction and that the adsorption of water onto the oxide follows a Langmuir type isotherm. Found good agreement with experimental data for 115-350°C and 0-47 kPa.

275	McKinnon, M. A, J. E. Tanner, R. J. Guenther, J. M. Creer, J. W. Doman, and C. E. King, "REA-2023 BWR Cask Performance Test," Third International Spent Fuel Storage Technology Symposium/Workshop, CONF-860417, Seattle, Washington, April 8-10, 1986, pp. S-45 to S-60.	Thermal and shielding performance of the REA-2023 cask. Heat transfer performance was excellent, but shielding did not meet design expectations. Shielding modifications are needed.
276	McKinnon, M. A. and V. A. Deloach. "Spent Nuclear Fuel Storage - Performance Tests and Demonstrations." PNL-8451, UC-510, Pacific Northwest Laboratory, Richland, Washington, April 1993.	Evaluation of shielding and thermal performance of Castor V/21, Transnuclear TN-24P, and Westinghouse MC-10 Commercial fuel casks at INEL.
277	McKinnon, M. A., and A. L. Doherty, Spent Nuclear Fuel Integrity During Dry Storage — Performance Tests and Demonstrations, PNNL-11576, UC-810, Pacific Northwest National Laboratory, Richland, Washington, June 1997.	Fuel integrity surveillance from gas sampling during and after dry storage demonstration (Castor V/21, Transnuclear TN-24P, and Westinghouse MC-10 Commercial fuel casks at INEL, expanded to include VSC-17 and REA-2023 casks, also). Gas sampling analysis indicates that fuel storage in an inert fill gas is benign. In general, handling operations are more likely to cause damage than dry storage. Only two leaking rods were identified (by Kr-85 detection) in unconsolidated storage. By comparison, about 10 of the consolidated rods began to leak after consolidation.
278	McKinnon, M. A., and M. E. Cunningham, Dry Storage Demonstration for High-Burnup Spent Nuclear Fuel – Feasibility Study, EPRI, Palo Alto, CA, and U.S. Department of Energy, Washington, DC: 2003. 1007872, PNNL-14390, August 2003.	Feasibility study highlights the concerns associated with the trend toward higher burnup fuel and the need to dry store (& later transport) such material.
279	Minshall, P. C., A Summary of Nuclear Electric's Experience in the Storage and Handling of Spent Metallic Fuel, Task TEEZZ/35020, Report TEE/REP/0060/94, Commercial-Confidential, Nuclear Electric, November 1994.	Possibly a precursor to [158]. BNFL subcontracted with Nuclear Electric to survey metal fuel (Magnox fuel and its transfer from wet to dry storage) experience relevant to Hanford's then-anticipated fuel treatment effort.
280	Miska, C. R., Hot Conditioning Equipment Conceptual Design Report, WHC-SD-SNF-CDR-007, Rev. 0, EDT/ECN 616136, August 1996.	Two volumes. Specifies equipment and configuration but not operational details.
281	Mizia, R. E., Hydrogen Damage in DOE Spent Nuclear Fuel Packages, DOE/SNF/REP-019, Rev. 0, August 2000.	Evaluates potential for hydrogen generation for ATR fuel over 40 years of storage.
282	Mizia, R. E., R. K. McCardell, and W. J. Dirk, EDF-1752, Rev. 0, Evaluation of West Valley BWR and PWR Fuel Assemby and Fuel Rod Integrity, Project File No. 7852.4.7.1.52, October 16, 2001.	Internal Report. Observations of R. E. Ginna and Big Rock Point fuel cladding conditions as loaded into West Valley TN-REG and TN-BRP fuel casks.

283	Morimoto, T., M. Nagao, and F. Tokuda, Desorbability of Chemisorbed Water on Metal Oxide Surfaces. I Desorption Temperature of Chemisorbed Water on Hemitite, Rutile and Zinc Oxide, Bulletin of the Chemical Society of Japan, Vol. 41, July 1968, pp. 1533-1537.	Adsorption and desorption of water on alpha-Fe_2O_3, TiO_2, and ZnO.
284	Morissette, R. P., P. E. Schneringer, R. K. Lane,R. L. Moore, and K. A. Young, Commercial Radioactive Waste Management System Feasability with the Universal Canister Concept, GA-A-18320, Vol. l, DE86 010108, January 1986.	"Standard" dewatering practices. Basic drain, pull vacuum, backfill with helium, etc.
285	Mouradian, E. M., and L. Baker, Jr., 1963, Burning Temperatures of Uranium and Zirconium in Air, Nuclear Science and Engineering, Vol. 15, 1963 pp. 388-394.	Reaction rate assessed assuming boundary layer diffusion control.
286	Müller, A, "Safety-Related Aspects of Interim Dry Storage in the Scope of Atomic Licensing Procedures," Third International Spent Fuel Storage Technology Symposium/Workshop, CONF-860417, Seattle, Washington, April 8-10, 1986, pp. S-118 to S-135.	Discusses CASTOR type cask used in the Federal Republic of Germany, primarily thermal and shielding considerations. No discussion of drying.
287	Musgrave, L. E., Theory of Burning Curve Ignition of Nuclear Metals, Journal of Nuclear Materials, Vol. 43, 1972, pp. 155-163.	Develops an expression relating metal ignition temperature to specific surface area, oxidation rate, and mass of oxide on the surface. Validates with experimental data for uranium and plutonium.
288	Nagase, F., and T. Fuketa, Journal of Nuclear Science and Technology, Vol. 41, No. 12, December 2004, pp. 1211-1217.	Concludes that precipitation of radially-oriented hydrides can lead to axial crack propagation.
289	Nakamura, J., T. Otomo, T. Kikuchi, and S. Kawasaki, Oxidation of Fuel Rod under Dry Storage Condition, Journal of Nuclear Science and Technology, Vol 32. No. 4, April 1995, pp. 321-332.	Air and air-argon tests on rods with artificial defect. Suggested the oxidation difference between irradiated and un-irradiated rods was due to FP gas bubble accumulation at grain boundary and FP accumulation in UO_2 matrix. The irradiated rod in argon (low oxygen) showed very little deformation.
290	Nassini, H. E., C. Fuenzalida Troyano, A. M. Bevilacqua, J. Bergallo, M. Silva, and A. Blanco, Conceptual Design of a Modular System for the Interim Dry Storage of PHWR Atucha Spent Fuel, Management of Spent Fuel from Nuclear Power Reactors, IAEA proceedings, Vienna, Austria, 19-22 June 2006, pp. 267-279.	Conceptual design for dry storage with passive cooling. See also [291].

291	Nassini, H. E., C. Fuenzalida, A. M. Bevilacqua, and J. Bergallo, Conceptual Design of a Modular Facility for the Long-Term Dry Storage of PHWR Atucha Spent Fuels, IAEA-CN-144/6, Conference: International conference on management of spent fuel from nuclear power reactors, Vienna, Austria, 19-22 June, 2006.	Conceptual design for dry storage with passive cooling. Does not explicitly address removal of residual water.
292	Ni, N., S. Lozano-Perez, M.L. Jenkins, C. English, G.D.W. Smith, J.M. Sykes and C.R.M. Grovenor, "Porosity in Oxides on Zirconium Fuel Cladding Alloys, and its Importance in Controlling Oxidation Rates," Scripta Materialiia, Vol. 62, No. 8, 2010, pp. 564-567.	Correlates nanoscale porosity in oxide scale (observed with Fresnel imaging) on zirconium alloy to different stages of oxidation.
293	Nichols, F. A., Behavior of Gaseous Fission Products in Oxide Fuel Elements, WAPD-TM-570, UC-25: Metals, Ceramics, and Materials, Bettis Atomic Power Laboratory, Pittsburg, Pennsylvania, October 1966.	Fission gas swelling and fission gas release theories are reviewed. A general scheme of bubble formation, gravitation to grain boundaries, and stress-induced migration is suggested. No simple model applies.
294	Nigrey, P. J., An issue paper on the use of hydrogen getters in transportation packaging, SAND2000-0483, Sandia National Laboratory, February 2000.	Discusses getter material poisoning, the use of getters in packaging, the effects of radiation on getters, the compatibility of getters with packaging, design considerations, regulatory precedents, and general recommendations. Includes annotated bibliography on getters.
295	Nöring, R., R. Hüggenberg, I. Tripputi, and M. Pietrobon, Adaptation of approved transport and storage cask lines on individual customer requirements, RAMTRANS, Vol. 13, No. 3-4, pp. 355-360, 2002.	Transport and storage (dual use) GNB CASTOR casks - no detail on drying. Intended for fuel with reduced decay heat.
296	Northwood, D. O., Kosasih, U., "Hydrides And Delayed Hydrogen Cracking In Zirconium And Its Alloys," International Metals Reviews, Vol. 28, No. 2, 1983, pp. 92-121.	Detailed review of zirconium hydride, cracking, and hydride orientation and reorientation under stress.
297	Novak, J., and I. J. Hastings, "Post-Irradiation Behavior of Defected U02 Fuel Elements at 220°C to 250°C in Air," Proceedings of Spent Fuel/Cladding Reaction During Dry Storage, NUREG/CP-0049. Gaithersburg, Maryland, 1983.	Experimetal work shows relative dimensional stability at 220-230°C for up to 685 hours (consistent with oxidation to U_3O_7), while severe cladding sheath splitting occurred after about 200 hours at 250°C (oxidation to U_3O_8).
298	Novikov, A., Development of Approaches for the long term storage of Damaged Nuclear Fuel, Chernobyl Nuclear Power Plant, Kiev, Ukraine, pp. 385-393.	Summary of dry storage issues. Recommends drying to remove free and absorbed [adsorbed] water but includes no process details or specific criteria.

299	NUREG-1536 Standard Review Plan for Dry Cask Storage Systems, Final Report, Rev. 1, U. S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safegards, July 2010.	Defines temperature limitations based on not allowing more than 90 MPa hoop stress to zirconium alloy and steel cladding. Specifically references PNL-6365. But makes allowance for forced helium drying as an alternative to heated vacuum drying. Also mentions potential for freezing. Searchable full text at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1536/.
300	NUREG-1536 Standard Review Plan for Dry Cask Storage Systems, Final Report, U. S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safegards, January 1997.	Acceptance criteria intended to reduce the amount of water present to protect cladding against degradation, pre-dates serious consideration for drying already degrade materials. Specifically references PNL-4835, PNL-6189, PNL-6364, and Section 8 calls out drying procedurally on the basis of PNL-6365. Full text at https://www.osti.gov/src/servlets/purl/453763-RYHRil/webviewable/453763.pdf or at http://libsearch/Reports/NUREG-1536_OSTI.html.
301	NUREG-1609 Standard Review Plan for Transportation Packages for Radioactive Material, U. S. Nuclear Regulatory Commission, Spent Fuel Projetc Office, March 1999.	Section 4.5.2.3" Confirm that the application demonstrates that any combustible gases generated in the package during a period of one year do not exceed 5% (by volume) of the free gas volume in any confined region of the package. No credit should be taken for getters, catalysts, or other recombination devices." Full text at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1609/final/sr1609.pdf
302	O'Brien, J., D. Phetteplace, M. Sprenger, and H. Welland, ICPP-603 Fuel Canning Project May Street South Vacuum Drying Mockup Station Test Report Document, Project File No. 015635, September 1994.	Basic tests and results (temperatures and pressure responses) for evacuating water from a mockup of the CPP-603 Fuel Canning Project drying station.
303	Ogden, D. M., MCO Pressurization Analysis of Spent Nuclear Fuel Transportation and Storage, WHC-SD-SNF-ER-014, Rev. 0, EDT/ECN 614689, September 1996.	Results of indicate that overpressurization of the MCO can occur within hours given the bounding reaction surface area and 3.0 Kg of residual water during shipping or 2.5 Kg of residual water during storage. Overpressurization can be prevented if the teactive surface area is shown to be less than 80,000 cm^2 or by limiting the available water.

304	Olander, D. R., D. Sherman, and M. Balooch, Retention and Release of Water by Sintered Uranium Dioxide, Journal of Nuclear Materials, Vol. 107, 1982, pp. 31-45.	Water adsorbed on open porosity of the UO_2 is released at ~100°C. Model describes the process as "teakettle pore" diffusion as described by a BET isotherm. Occluded gases from pellet sintering not removed until temperatures in excess of 1000°C. (Not diffusion governed, model based on desorption kinetics from binding sites with a distribution of binding energies.)
305	Oliver, B. M., G. S. Klinger, J. Abrefah, S. C. Marschman, P. J. MacFarlan, and G. A. Ritter, Drying Results of K-Basin Fuel Element 0309M (Run 3), PNNL-11840, UC-602, July 1998.	Drying tests to support N-reactor fuel drying and MCO loading operations.
306	Oliver, B. M., G. S. Klinger, J. Abrefah, S. C. Marschman, P. J. MacFarlan, and G. A. Ritter, Drying Results of K-Basin Fuel Element 1164M (Run 6), PNNL-11896, UC-602, August 1998.	Single element drying test for element split along the length in places and slightly bowed with ends intact. Compares estimates for water removal from gas flow & pressure rise data with corresponding quantities observed in condenser. Informs hydrogen release and hydride decomposition also.
307	Oliver, B. M., G. S. Klinger, J. Abrefah, S. C. Marschman, P. J. MacFarlan, and G. A. Ritter, Drying Results of K-Basin Fuel Element 6513U (Run 8), PNNL-11969, UC-602, July 1999.	Drying tests to support N-reactor fuel drying and MCO loading operations.
308	Oliver, B. M., G. S. Klinger, J. Abrefah, S. C. Marschman, P. J. MacFarlan, and G. A. Ritter, Spent Fuel Drying System Test Results (Dry-Run in Preparation for Run 8), PNNL-12136, UC-602, July 1999.	Drying tests to support N-reactor fuel drying and MCO loading operations.
309	Oliver, B. M., G. S. Klinger, J. Abrefah, S. C. Marschman, P. J. MacFarlane, and G. A. Ritter, Drying Results of K-Basin Fuel Element 2660M (Run 7), PNNL-11897, UC-602, July 1999.	Single element drying test for element with extensive damage to cladding. Compares estimates for water removal from gas flow & pressure rise data with corresponding quantities observed in condenser. Informs hydrogen release and hydride decomposition also.
310	Olsen, C. S., "The Performance of Defected Spent LWR Fuel Rods in Inert Gas and Dry Air Storage Atmospheres," EG&G Idaho, Idaho Falls, Idaho, NUREG/CR-4074, TI85 006891, January 1985.	Presents results of the (third) nondestructive examination, conducted to find any degradation in eight fuel rods (4 PWR and 4 BWR, 2 each having artificial defects) that had been subjected to 13,168 hours at 229-217°C. The BWR rods developed cracks at some (but not all) of the defect sites. The PWR rods did not.

311	Operating Experience Feedback Report - Analysis of Corrosion-Related Occurrences, DOE/DP-0116, DE93 010477, Office of the Assistant Secretary for Defence Programs, Office of Self-Assessment Safety Diagnostic Division, U.S. Department of Energy, October 1992.	General observations of corrosion-related problems in DOE facilities. Identifies concerns for fuel storage at the CPP-603 basin due to corrosion.
312	Orman, S., L. W. Owen, and G. Picton, The Corrosion Behavior of Nickel-Plated Uranium, Corrosion Science, Vol. 12, 1972, pp. 35-44.	Nickel plate is initially protective to the uranium. Over time in a humid environment uranium hydride developes at the nickel-uranium interface either from hydrogen penetration at the base of a pore or by diffusion through the nickel.
313	Orman, S., The Effect of Certain Gases on the Rate of Oxidation of Uranium by Water Vapour, Chemistry and Industry, Vol. 42, October 1963, pp. 1692-1693.	Examines effects of nitrogen, hydrogen, and oxygen on uranium oxidation (in degassed water vapor) at 70°C.
314	Ospina, C, "Intermediate Dry Storage of the Spent Fuel of Reactor DIORIT," Swiss Federal Institute for Reactor Research (EIR), 1981?, pp. 173-187.	Discusses plans for the dry storage of DIORIT research reactor fuel. Identifies materials issues by comparison with LWR systems.
315	Ospina, C., "Dry Storage Cask - DIORIT - Swiss Experience," CONF-860417, Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986, pp. P-128 to P-134.	Emphasis on fuel transfer and dry loading to CASTOR-DIORIT cask. No detail on drying.
316	Ospina, C., H. Baatz, and D. Methling, "Planning and Experience on the Dry Storage of DIORIT Spent Fuel," in "Proceedings of 7th International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM 83)," Conference, No. 830528, 1983, pp. 231-237.	Emphasis on selection of, fuel transfer and dry loading to CASTOR-DIORIT cask. No detail on drying.
317	Owen, L. W., and J. R. Alderton, The Oxidation and Protection of Uranium at Temperatures Near Ambient, Atomic Weapons Research Establishment, Aldermaston, Berkshire, England, 1963?, pp. 865-871.	Missing pp. 868-869.
318	Paget, F. T. W., and R. D. Tolmie, Method and equipment for drying and dry-storage of spent fuel and high-level radioactive waste [Verfahren und Vorrichtung fuer die Trocknung und Trockenlagerung von bestrahltem Kernbrennstoff oder stark radioaktivem Abfall], Application No. 8204251, Patent GB 2096389, March 1981.	Facility stores fuel in an air circulating closed primary circuit. At some humidity level, a fraction of said air is drawn off and moisture is condensed. Patent ceased throgh non-payment of renewal fee.

319	Pajunen, A. J., Cold Vacuum Drying Residual Free Water Test Description, HNF-1851, Rev. 0, EDT/ECN 616193, December 1997.	Residual free water estimates based on three alternate models of degraded fuel and operating constraints to limit residual water in an MCO to less than 200 grams.
320	Pajunen, A. J., Spent Nuclear Fuel Project Product Specification, HNF-SD-SNF-OCD-001, Rev. 5, ECN 662257, December 2000.	Itemizes specifications for MCO with regard to cold vacuum drying in preparation for a 40 year interim storage period.
321	Palmer, A. J., and A. E. Arave, Drying Tests Performed on TMI-2 Fuel Canisters, INEL-94/053, December 1994.	Considers exit stream dew point for dryness determination, assesses time to dry, pore water in Licon, removal of liquid water, amount of water present after dewatering without drying.
322	Palmer, A. J., Drying Tests Conducted on TMI Fuel Canisters Containing Simulated Debris, CONF-9606116-18, INEL-95/00431, Topical Meeting on DOE Spent Fuel and Fissile Material Management, Reno, Nevada, June 16, 1996.	Tests on LICON and other simulated TMI-2 debris materials and configurations.
323	Palmer, A. J., EDF-797, Rev. 0, Water Ingress into TMI DSCs During Storage, March 2, 1999.	Internal Report. Calculates bounding amount of atmospheric moisture that could enter DSCs during a 40 year storage period.
324	Park, S.Y., J.H. Kim, M.H. Lee, and Y.H. Jeong, “Stress-Corrosion Crack Initiation and Propagation Behavior of Zircaloy-4 Cladding Under an Iodine Environment,” Journal of Nuclear Materials, Vol. 372, 2008, pp. 293-303.	Proposes crack initiation and propagation mechanisms of Zircaloy-4 claddings, which had different microstructures, by a grain-boundary pitting model and a pitting-assisted slip cleavage model. The iodine environment increased the crack rate 10,000 times over an inert environment.
325	Pasupathi, V., and D. Stahl, Expected Performance of Spent LWR Fuel Under Dry Storage Conditions, Research Project 2062-1 Final Report, EPRI-NP-2735, DE83 901104, Electric Power Research Institute, Palo Alto, California, December 1982.	Suggests a cladding temperature limit of 250°C for storage in air to prevent oxidation that could lead to gross cladding rupture. Alternative of storage in inert gas proposed to mitigate oxidation instead.
326	Patterson, D. W., and D. S. Hoye, "Canadian Experience Storing Irradiated CANDU Fuel in Concrete Canisters," CONF-860417, Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986, pp. S-136 to S-155.	Discusses Canadian above ground concrete canister design and demonstration program (initiated in 1975). Builds commercial licensing case from 10 years of similar experience for R&D site. Used forced air drying prior to sealing basket and loading basket to canister (via transfer flask).
327	Peacock, H. B., and R. L. Frontroth, Properties of Aluminum-Uranium Alloys, WSRC-RP-89-489 DE90 003889, Westinghouse Savannah River Company, Aiken SC., August 1989.	Discusses aluminum-uranium fuels, their performance, and failure mechanisms.

328	Peacock, H. B., *Pyrophoricity of Uranium,* Report WSRC-TR-92-106, Savannah River Laboratory, March 1992.	Concludes spontaneous ignition of spherical particles larger than 1/16 inch in diameter would not be expected to occur in air at room temperature. The ignition temperatures calculated from ANL data for 1/16, 1/4, and 1/2 inch diameter spherical particles are 333, 375, and 399°C, respectively (estimated to be within ±10%).
329	Peacock, H. B., R.L. Sindelar and P.S. Lam, Temperature and Humidity Effects on the Corrosion of Aluminum-Base Reactor Fuel Cladding Materials During Dry Storage, XA04CO098, 2004.	Corrosion mechanisms for aluminum materials in dry storage, including radiolysis, expected performance at 100% humidity and 200C.
330	Peacock, H. B., R.L. Sindelar, P.S. Lam, and T. H. Murphy, Evaluation of Corrosion of Aluminum Based Reactor Fuel Cladding Materials During Dry Storage (U), WSRC-TR-95-0345, November 1995.	Detailed discussion of aluminum corrosion experiments and radiolysis evaluation at SRS. Extrapolation with assumptions to estimate corrosion over 50 years of dry storage.
331	Peacock, H. B., R.L. Sindelar, P.S. Lam, and T. H. Murphy, Experiments for Evaluation of Corrosion to Develop Storage Criteria for Interim Dry Storage of AL, WSRC-MS-94-0564, CONF-941207-12, DOE Spent Nuclear Fuel - Challenges and Initiatives, Salt Lake City, Utah, December 12-16, 1994.	Aluminum coupons corroded in an autoclave.
332	Pearce, R. J., M. J. Bennett, and J. B. Price, "Oxidation of irradiated uranium in moist air," Nuclear Energy, Vol. 27, No. 5, October 1988, pp. 305-309.	Develops an oxidation rate expression for irradiated U metal in dry and moist air at 75 to 300°C.
333	Pearson, H. E., *AEC Uranium Fire Experience,* Hanford-64841, Richland, Washington, September 17, 1954.	Assesses anecdotal experience with uranium fires and correlates presence of hydrogen or hydride with each occurrence, acknowledging that moisture may be complicit in the reaction.
334	Peehs, M. and J. Fleisch, "LWR Spent Fuel Storage Behaviour," Journal of Nuclear Materials, Vol. 137, 1986, pp. 190-202.	Concludes that dry storage under inert gases is not expected to cause any cladding failures over the interim storage period as long as the dry storage temperature is limited to 450°C.
335	Peehs, M., and K. Knecht, Examples of Remote Handling of Irradiated Fuel Assemblies in Germany, IAEA-TECDOC-1061, Siemens AG Power Generation, Nuclear Fuel cycle, Erlangen, Germany, International Atomic Energy Agency, 1999.	Vacuum drying with additional heat.

336	Peehs, M., G. Kaspar and E. Steinberg, “Experimentally Based Spent Fuel Dry Storage Performance Criteria,” Third International Spent Fuel Storage Technology Symposium/Workshop, CONF-860417, Seattle, Washington, April 8-10, 1986, pp. S-316 to S-331.	Models cladding performance in dry storage suggesting that damage functions accounting for creep, oxidation, iodine stress corrosion cracking and propagation of individual cracks in the clad level out quickly at non-problematic values.
337	Peehs, M., R. Bokelmann, and J. Fleisch, "Spent Fuel Dry Storage Performance in Inert Atmosphere," CONF-860417, Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986, pp. S-215 to S-230.	Investigates fission product release; oxide, creep, hoop strain, and SCC in cladding; defects and water-logged fuel (water successfully removed with vacuum drying process); and thermal performance. One experiment flooded (2 minutes from 210°C) and re-dried the test assembly several times without damage to its integrity.
338	Pennington, C. W., and B. R. Teer, "Dry Storage Technology from Transnuclear," CONF-860417, Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986, pp. P-91 to P-98.	Discusses the transition from transportation cask to (dual purpose) dry storage cask and indentifies dry storage cask design requirements.
339	Pennington, C. W., NAC's Modular, Advanced Generation, Nuclear All-purpose STORage (MAGNASTOR) System: New Generation Multipurpose Spent Fuel Storage for Global Application, Paper No. 108, PATRAM 2004: International Symposium on the Packaging and Transportation of Radioactive Materials, Berlin, Germany, 20-24 September 2004.	Multipurpose canister systems for storage and transportation used by US utilities. MEGAVAC drying system: can use vacuum or pressurized helium depending on heat loads.
340	Perfettini, J., Neutron Moisture Meter Measurements in Waste Packages, Nuclear Technology, Vol. 115, August 1996, pp. 153-161.	Compares thermal neutron adsorption characteristics for actual material to chemically similar dry material to assess free water content.
341	Pescatore, C. and M. Cowgill, Temperature Limit Determination for the Inert Dry Storage of Spent Nuclear Fuel, Brookhaven National Laboratory, EPRI TR-103949, Project 3290-03, Electric Power Research Institute, Palo Alto, California, May 1994.	Identifies cladding creep from fill gas overpressure as the rate-determining degradation and failure mechanism for setting maximum allowable temperature limits. Because of their much lower internal pressure, BWR fuel rods are much less susceptible to cladding creep and may tolerate temperatures of 400°C or higher. Gross rupture is discounted (even for PWR) with pinhole leaks assessed as the most likely fracture failure in dry storage. Finds U.S. data in good correlation with German creep model.

342	Petrik, N. G., A. B. Alexandrov, and A. I. Vall, Interfacial Energy Transfer during Gamma Radiolysis of Water on the Surface of ZrO_2 and Some Other Oxides, Journal of Physical Chemistry B, Vol. 105, No. 25, 2001, pp. 5935-5944.	Examines radiolytic hydrogen yield and how it varies in the presence of oxides. Some lower the yield (MnO_2, Co_3O_4, CuO, and Fe_2O_3); some have little effect compared to control experiments (MgO, CaO, SrO, BaO, ZnO, CdO, Cu_2O, NiO, Cr_2O_3, Al_2O_3, CeO_2, SiO_2, TiO_2, Nb_2O_5 and WO_3); and others increase the yield (Ga_2O_3, Y_2O_3, La_2O_3, Nd_2O_3, Sm_2O_3, Eu_2O_3, Gd_2O_3, Yb_2O_3, Er_2O_3, HfO_2, and ZrO_2).
343	Pickman, D. O., "Properties of Zircaloy Cladding," Nuclear Engineering Design, Vol. 21, 1972, pp. 212-236.	Discusses the mechanical aspects of cladding and zircaloy performance requirements under BWR and PWR conditions, including irradiation effects and hydride alignment correlation with strength.
344	Piepho, M. G., and R. D. Crowe, Thermal Analysis of Cold Vacuum Drying of Spent Nuclear Fuel, HNF-SD-SNF-CN-023, Rev. 1, July 1998.	MCO thermal modeling evaluates normal and off-normal operating cases. Includes some UH3 pyrophoricity references.
345	Planell, J., M. Mason, and G. Guerra, "Extended Fuel Burnup Demonstration Program Topical Report," Transnuclear, Inc., DOE/ET/34014-11, 1983.	Evaluates heat load, leak rate and minimum cooling time for extended burnup of fuels.
346	Plys, M. G., and B. Malinovic, *IWTSMetal-Water Reaction Rate Evaluation,* FAII99-26, Fauske & Associates, Inc., Burr Ridge, Illinois. (Released by the SNF Project as SNF-4266, Rev. 0, *IWTS Metal- Water Reaction Rate Evaluation,* EDT/ECN: 626804, July 1999.)	The report presents a thermal stability analysis of partially metallic particulate in two IWTS components, the knock out pot and settlers. Examines mechanisms by which a particulate bed could ignite.
347	Plys, M. G., and D. R. Duncan, Uranium Pyrophoricity Phenomena and Prediction (FAI/00-39), SNF-6781, Rev. 0, EDT 629783, October 2000.	Ignition theory and documented incidents of pyrophoric reactions.
348	Porsch, G., J. Fleisch, and B. Heits . 1986. "Accelerated High-Temperature Tests With Spent PWR and BWR Fuel Rods Under Dry Storage Conditions." Nucl. Tech. 74:287-298.	No cladding breaches after 16 months of dry storage at 400, 430, and 450°C in helium atmosphere. Creep deformation remained below 1% as predicted. Concludes that for zircaloy clad fuel dry storage at up to 450°C in inert gas is acceptable.
349	Procedures and techniques for the management of experimental fuels from research and test reactors, Proceedings of an Advisory Group meeting, 1-4, International Atomic Energy Agency, Vienna, Austria, December 1997.	Illustrates the dawning understanding that a strategy of interim dry storage will necessarily preclude permanent fuel disposition. Full text available at http://www-pub.iaea.org/MTCD/publications/PDF/te_1080_prn.pdf

350	Puigh, R. J., II, H. Toffer, F. J. Heard, J. J. Irwin, T. D. Cooper, M. Plys, M. Beary, A. Johnson, Jr., and F. F. Huang, Technology Status in Support of Refined Technical Document Title: Baseline for the Spent Nuclear Fuel Project, ECN 615132, WHC-SD-SNF-TI-014, Rev. 1, October 20, 1995.	Uranium hydride, potential for ignition, conditioning and related issues for N Reactor fuel.
351	Purcell, P., Experimental testing on a simulated flask lid seal system to confirm containment measuring procedures, RAMTRANS, Vol. 13, No. 3-4, pp. 251-255, 2002.	Effect of moisture on leak detection and the need to dry sealing surface to ensure valid leak test results. The addition of a vacuum drying step provided improved results over exposure to pressurized dry air (alone).
352	R. L. Sindelar, G. R. Caskey, H. B. Peacock, Jr., and P. S. Lam, "Evaluation of Degradation During Interim Dry Storage of Aluminum-clad Fuels" in proceedings of International High Level Radioactive Waste Conference at Las Vegas, Nevada, 05/01/95 - 05/01/95, pp. 652-654.	In the presence of moisture and/or air, corrosion of the aluminum clad and the fuel matrix is th limiting degradation mechanism at low temperatures (below 200°C).
353	R.E. Lords, “Five Years of INEEL Spent Nuclear Fuel Drying Experience,” INEEL/CON-98-00074, American Nuclear Society Topical Meeting, Charleston, SC, 1998.	Vacuum drying cold damaged fuel for vented storage.
354	Rashid, J. Y. R., and Machiels, A. J., A methodology for the evaluation of fuel rod failures under transportation accidents, Paper # 118, 14th International Symposium on the Packaging and Transportation of Radioactive Materials (PATRAM 2004), Berlin, Germany, September 20-24, 2004.	Discussed failure modes. Figures show model predictions for temperature, hoop stress, hydrogen and radial hydride precipitation with time.
355	Rashid, J.Y.R. and Machiels, A.J., “Examination of the Creep Rupture Phenomenon and the Development of an Acceptance Criterion for Spent Fuel Dry Storage”, IAEA-CN-102/68, International Conference on Storage of Spent Fuel from Power Reactors, Vienna, 2-6 June 2003, pp. 114-116.	The study shows that the 1% strain criterion has no valid basis for indicating cladding integrity. A stress-based criterion is recommended instead.
356	Rashid, Y. R, D. L. Sunderland, and R. O. Montgomery, “Creep as the Limiting Mechanism for Spent Fuel Dry Storage,” EPRI TR-1001207, Electric Power Research Institute, December 2000.	Stress corrosion cracking, delayed hydride cracking, creep, and pin-hole-equivalent failure in cladding under dry storage conditions.

357	Rashid, Y. R. and R. S. Dunham, “Creep Modeling and Analysis Methodology for Spent Fuel in Dry Storage,” EPRI TR-1003135, Electric Power Research Institute, November 2001.	Introduces irradiation-damage annealing and hydrogen effect model modifications to creep model that already addressed stress and temperature. Includes localized damage from oxidation-induced metal loss, hydrides and other local defects. Use of a strain value as an extention from fuel design to limit in storage creep is contradicted by creep rupture data. Use of a stress-based criterion is reinforced by a cumulative damage model.
358	Reed, D. T., and R. A. Van Konynenburg, Effect of Ionizing Radiation on Moist Air Systems, Argonne National Laboratory and Lawrence Livermore National Laboratory, Vol. 112, Materials Resarch Society Symposium Proceedings, 1988, pp. 393-404.	Nitric acid changes the chemistry by comparision to dry air systems. Summarizes past work in this area and evaluates it in the context of a tuff repository environment.
359	Reese, S. and J. C. Crepeau, Drying Behavior of Three Mock Spent Fuel Types, Progress Report, September 1998.	Internal Report. Like [249], only test materials represented stainless steel plate, graphite, and particulate fuels
360	Reid, C. R., and E. R. Gilbert, Methodology for Determining Criteria for Storing Spent Fuel in Air. PNL-6018, UC-85, Pacific Northwest Laboratory, Richland, Washington, November 1986.	Develops linear cumulatve damage and remaining life methods for modeling fuel oxidation in dry storage with non-constant storage temperature.
361	Reilly, M. A., Spent Nuclear Fuel Project Technical Databook, HNF-SD-SNF-TI-015, Rev. 6, ECN 68624, October 1998.	The authoritative source for fuel related information for the Hanford SNF Project.
362	Rigby, D. B., Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel, United States Nuclear Waste Technical Review Board, December 2010.	General discussion of drying and storage referencing regulatory guidance.
363	Ritchie, A. G., "A Review of the Rates of Reaction of Uranium of Oxygen and Water Vapor at Temperatures up to 300°C," Journal of Nuclear Materials, 102, pp. 170-182, 1981.	Reviews literature and assesses dry and moist air oxidation rates for uranium. Discusses variations in relative rates with temperature and humidity.
364	Ritchie, A. G., R. C. Greenwood, and S. J. Randles, The Kinetics of the Uranium-Oxygen-Water Vapour Reaction Between 40 and 100°C, Journal of Nuclear Material, Vol. 139, 1986, pp. 121-136.	The reaction rate is tested between 40-100°C and 11 to 75% humidity. Rate equation is given as $k = 7.6 \times 10^{13} \exp(-26.4 \text{ kcal}/RT)$ mg weight gain/cm^2h. Kinetics appeared to be linear and in good agreement with data from an earlier review.
365	Ritter, G. A., S. C. Marschman, P. J. MacFarlan, and D. A. King, System Design Description for the Whole Element Furnace Testing System, PNNL-11807, UC-602,	Documentation to support drying and MCO loading operations.

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366	Roberts, J. P., and F. C. Sturz, "Progress in Dry Spent Fuel Storage Licensing and Rulemaking," Third International Spent Fuel Storage Technology Symposium/Workshop, CONF-860417, Seattle, Washington, April 8-10, 1986, pp. S-204 to S-212.	Identifies cracks in welds of the borated stainless steel cask basket of the unlicensed demonstration of the CASTOR V/21 cask storing Surry fuel at INEL.
367	Roberts, R. J., D. Tulberga, and C. Carter, The Idaho spent fuel project, An update - January 2003, IAEA-CN-102/79, Storage of Spent Fuel from Power Reactors, 2003 Conference Proceedings, organized by the International Atomic Energy Agency in co-operation with the OECD Nuclear Energy Agency, Vienna, Austria, October 2003, pp. 218-226.	This privatized dry storage project was designed and intended to accommodate Shippingport, TRIGA, and PeachBottom fuels. The project was abandoned before the start of construction.
368	Rodrigo, L., J. Therrien, and B. Surette, Low-temperature, vacuum-assisted drying of degraded spent nuclear fuel, Canadian Nuclear Society conference on waste management, Ottawa, Ontario, May 2005.	Vacuum drying overpacked tile hole container at below 100C (similar to Hanford's MCOs). Drying tests consider issues of freezing (water at low pressure due to poor heat transfer), melting bitumen (present in some tile hole containers), and cover gas in storage after drying (2% oxygen in Argon with a provision to purge and repressurize to 1-2 psig when pressure exceeds 5 psig).
369	Roland, V., Y. Solignac, M. Chiguer, and Y. Guenon, Dry Storage Technologies: Keys to Choosing Among Metal Casks, Concrete Shielded Steel Canister Modules and Vaults, Radioactive Waste and Spent Fuel Management International Conference, Bulgaria,6-8 November 2003.	TN-24 cask family and NUHOMS modular dry storage system discussed.
370	Rosen, R. S., and W. J. O'Connell, Creep Strains Predicted from Constitutive Equations for Zircaloy-Clad Spent Fuel Rods, Lawrence Livermore National Laboratory, Proceedings of the Sixth Annual International Conference on High-Level Radioactive Waste Management, Las Vegas, NV, ASCE and ANS, 1995, pp. 621-624.	The gas volume increases as the strain increases, resulting in a lowering of the stress and subsequent strain rate. Diffusion-controlled cavity growth (DCCG) is expected to be the rate-controlling failure mechanism. The volume-stress effect is important for assessing the design upper-limit temperatures for storage.
371	Rothman, A. J., Potential Corrosion and Degradation Mechanisms of Zircaloy Cladding on Spent Nuclear Fuel in a Tuff Repository, UCID-20172, Lawrence Livermore National Laboratory. ACC: NNA.19870903.0039, Livermore,	Identifies the primary potential for cladding degradation to come from delayed hydride cracking and stress corrosion cracking.

	California, September 1984.	
372	Ryskamp, J. M., J. P. Adams, E. M. Faw, P. A. Anderson, Corrosion Experiments on Stainless Steels Used in Dry Storage Canisters of Spent Nuclear Fuel, INEL-96/0317, September 1996.	Microbes / biofilms identified as possible drying impediment and possible storage problem with Microbially Induced Corrosion (MIC). Initial test plans established. Appendix A includes an overview of possible microbe influenced corrosion.
373	S. Cohen & Associates, "Effectiveness of Fuel Rod Cladding as an Engineered Barrier in the Yucca Mountain Repository," Contract No. 68D70073 prepared for the U.S. Environmental Protection Agency, 1999.	Discusses modes of cladding failure (manufacturing defects, in-service failure, handling mishaps, failure in wet or dry storage, and projection to repository disposal). Summarizes fuel temperature and burnup characteristics and cladding failures for PWR and BWR dry storage demonstration fuels.
374	Sanders, T. L., C. A. Ottinger, J. L. Brimhall, J. M. Creer, E. R. Gilbert, R. H. Jones, and P. E. McConnell, Considerations Applicable to the Transportability of a Transportable Storage Cask at the End of the Storage Period, SAND88-2481, Sandia National Laboratory, Albuquerque, New Mexico, November 1991.	Discusses waterlogged fuel and estimates residual water at 4 grams per failed rod. Considers all manner of failure modes and errors that could affect safety functions during transportation. Recommends design basis limits to internal environment to maintain fuel and material integrities. Aging properties of metals are not well known for long periods under temperature and other conditions in the storage environment.
375	Sanders, T. L., K. D. Seager, Y. R. Rashid, P. R. Barrett, A. P. Malinauskas, R. E. Einziger, H. Jordan, T. A. Duffey, S. H. Sutherland, P. C. Reardon, A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements, Sandia Report, SAND-90-2406, November 1992.	Brief, well-referenced discussion of waterlogged fuel; the rest of the document is not really germain to drying.
376	Sandoval, R. P., R. E. Einziger, H. Jordan, A. P. Malinauskas, and W. J. Mings, Estimate of CRUD Contribution to Shipping Cask Containment Requirements, SAND88-1358, Sandia National Laboratories, January 1991.	Emphasis is on crud as a source of radioactive material deposited on the surface of fuel. Analysis indicates constituents, particle size distribution and range of crud depth observed. Similar considerations facilitate water adsorption estimates.

377	Sasahara, A. and T. Matsumura, "Post-irradiation Examinations Focused on Fuel Integrity of Spent BWR-MOX and PWR-UO2 Fuels Stored for 20 years," Nuclear Engineering and Design, Vol. 238, 2008, pp. 1250-59.	Post irradiation examination of BWR and PWR fuel after 20 years in storage. Gas analysis, cladding metallography and strength were evaluated as indicators of fuel integrity. Hydrogen migration had little effect on the fuel integrity during dry storage. Experimental heat of transport data was used to project hydrogen redistribution behavior over 40 years indicating little effect on fuel integrity in storage.
378	Satterthwaite, B. C., and G. C. Thurston, ICPP-603 Fuel Canning Project Fuel Handling Operations Functional Test Report, Project File No. 015635, September 1994.	Basic process description, function, and mockup performance.
379	Schneider, K. J., and S. J. Mitchell, Foreign Experience on Effects of Extended Dry Storage on the Integrity of Spent Nuclear Fuel, PNL-8072, Pacific Northwest Laboratory, April 1992.	Status of international dry storage usage in 1992. Identifies corrosion mechanisms of concern and compares use of oxidizing and inert environments.
380	Schnizlein, J. G., L. Baker, Jr., and J. D. Bingle, The Ignition of Binary Alloys of Uranium, Journal of Nuclear Materials, Vol. 20, 1966, pp. 39-47.	Addition of aluminum (and also e, Bi, C, Pb, Mo, Nb, Pd, Pt, Ru, Si, Ti, and V) lowers ignition temperature. Copper (and also Bi, Pb, Pd, Pt, Ru and V) enhances the protectiveness of the oxide. Addition of Ce, Cr, H, Fe, Ni, Rh, Ag, Ta, Th, and Zr had no significant effect. Complex but reproducible temperature-time curves were obtained where both effects occured with the same alloy. Binary alloys of uranium with 0.5, 1, and 2% other element were tested.
381	Schnizlein, J. G., P. J. Pizzolato, J. A. Porte, J. D. Bingle, D. F. Fischer, L. W. Mishler, and R. C. Vogel, *Ignition Behavior and Kinetics of Oxidation of the Reactor Metals Uranium, Zirconium, Plutonium and Thorium, and Binary Alloys of Each,* A Status Report, ANL-5974, Reactors-General, TID-4500, 14th Ed., Argonne National Laboratory, Lemont, IL, April 1959.	Foundational work for [380]. Highlights the importance of specific surface area. Includes surveys of the literature on the oxidation of plutonium and thorium.
382	Schreiner, R., Clad Degradation - FEPs Screening Arguments, ANL-WIS-MD-000008, Rev. 2, October 2004.	Features, events and processes associated with cladding degradation. Discusses implications associated with waterlogged fuel.
383	Schwartz, M. W., Standard Review Plan for Reviewing Safety Analysis Reports for Dry Metallic Spent Fuel Storage Casks, OCRL-21137 DE89 004708, 1988.	Degradation and related drying issues framing license requirements.

384	Seiffert, S. L., and A. H. Novick, Destructive Examination of Defective Fuel Plates From ATR Fuel Elements A1 97F and XA456N, PR-T-79-0 16, EG&G Idaho, Inc., Idaho Fall, ID, August 1979.	Nondestructive and destructive evaluation of ATR fuel plates, one with blister, another with comet defect (possibly from pitting corrosion). Assemby with apparent manufacturing defect (thin cladding) failed near the end of normal service.
385	Seiffert, S. L., Destructive Examination of ATR Fuel Element XA029HK Which Failed During Operation, PR-T-79-020, EG&G Idaho, Inc., September 1979.	Nondestructive and destructive evaluation of ATR fuel element with two observable pinholes and other significant pits. Element failed near the end of normal service. The deepest pits were in an area of scuffs and shallow scratch marks allow the plate edge. Localized loss of fuel was identified in the area around the pinholes. Corrosion product streaming associated with the defects extended to the bottom of the element. Upward streaming near other significant pits suggest a possible link with canal storage corrosion where natural convective coolant flow conditions exist.
386	Sexton, R. A., MCO Monitoring Activity Description, HNF-3312, Rev. 1, ECN 651401, November 1998.	Calls for limited observation and sampling and high pressure detection capability.
387	Shelton-Davis, C., Fuel Canister Stress Corrosion Cracking Susceptibility Experimental Results, DOE/SNF/REP-082, Rev. 0, March 2003.	Concludes that austenetic stainless steels are not susceptible to cesium or rubidium from the SNF. Liquid metal embrittlement (LME) is not a concern. Also, 316L and 304L appear immune to stress corrosion cracking in the presence of cesium or rubidium hydroxide.
388	Shoesmith, D. W., S Sunder, L. H. Johnson, and M. G. Bailey, Oxidation of CANDU UO_2 Fuel by the Alpha-Radiolysis Products of Water, Atomic Energy of Canda Limited, Vol. 50, Materials Research Society Symposium Proceedings, 1985, pp. 309-316.	Oxidation of UO_2 by H_2O_2 (hydrogen peroxide being an alpha-radiolysis product) appears to be about 200 times faster than by dissolved oxygen. Both show strong pH dependence.
389	Siegmann, E. Clad Degradation - Summary and Abstraction, ANL-WIS-MD-000007 REV 00 ICN 01, acc: ML003722043, Las Vegas, Nevada, CRWMS M&O, April 2000.	Summarizes cladding degradation mechanisms.
390	Sindelar, R. L., D. J. Pak, P. J. French, R. F. Eakle, W. S. Large, and K. Chen, Instrumented, Shielded Test Canister System for Evaluation of Spent Nuclear Fuel in Dry Storage, WSRC-TR-97-00269 (U), September 1997.	Describes instrumented storage canister and drying process to be used for aluminum clad fuel dry storage demonstration. Recommends pressure response as an indication of dryness.

391	Sindelar, R. L., G. S. Bumgamer, G. R. Caskey, Jr., G. T. Chandler, J. R. Chandler, M. J. Dalmaso, D. L. Fisher, P. J. French, H. N. Guerrero, J. P. Howell, M. L. Hyder, N. C. Iyer, P. S. Lam. W. S. Large, D. R. Leader, S. Y. Lee, M. R. Louthan, Jr., J. I. Mickalonis, J. R. Murphy, D. J. Pak, H. B. Peacock, Jr., T. E. Skidmore, B. J. Wiersma, J. F. Zino, Alternative Aluminum Spent Nuclear Fuel Treatment Technology Development Status Report (U), WSRC-TR-97-0084, Westinghouse Savannah River Company, April 1997.	Aluminum corrosion and fuel performance in the context of direct disposal. Drying specification for conditioning prior to storage and disposal. Aluminum clad fuel is limited to less than 200°C. Free water limited to prevent excessive corrosion and to keep hydrogen below 4% by volume. Recommends slightly elevated temperature for vacuum drying.
392	Sindelar, R. L., H. B. Peacock, Jr., P. S. Lam, N. C. Iyer, M. R. Louthan, Jr., and J. R. Murphy, Scientific Basis for Storage Criteria for Interim Dry Storage of Aluminum-Clad Fuels, Mat. Res. Soc. Symp. Proc. Vol. 412, Materials Research Society, 1996, pp. 99-106.	Aluminum clad aluminum-uranium alloy fuel, FRR and DRR, preparation for sealed and non-sealed dry storage, limited to temps <200C (due to creep and diffusion mechanisms), low humidity to limit corrosion, R.H. <20%, mechanistic power law corrosion rate model (oxide film growth), also max pit depth with time to the 1/3 power (not accounting for through penetration galvanic effects), kinetic equation for uranium oxidation (oxide film growth, no passivating fim formation), Hydrogen Blistering at and above 275°C, mentions conditions for U-metal or UH_3 pyrophoricity.
393	Sindelar, R. L., S. D. Burke, and J. P. Howell, Evaluation of Radionuclide Release from Aluminum-Based SNF in Basin Storage, proceedings, 3rd Topical Meeting on DOE Spent Nuclear Fuel and Fissile Material Management, Charleston, SC, September 8-11, 1998, pp. 259-263.	Concludes postulated release is within authorized limits, allowing for receipt of some breached fuel to the basin without canning.
394	Sindelar, R.L., Iyer, N.C., Peacock, H.B., and Louthan, Jr., M.R. Acceptance Criteria for Interim Dry Storage of Aluminum-Clad Fuels, Westinghouse Savannah River Company Report WSRC-MS-94-0618, CONF-950570-10, 1994.	Overview of drying issues.
395	Sindelar, R.L., Peacock, H.B., Lam, P-S., Iyer, N.C., and Louthan, Jr., M.R. Acceptance Criteria for Interim Dry Storage of Aluminum-Alloy Clad Spent Nuclear Fuels (U) Fuels, Westinghouse Savannah River Company Report WSRC-TR-95-0347, March 1996.	Establishes limits to degradation for aluminum clad fuel so that after drying and dry storage post-storage handleability, a full range of ultimate disposal options, criticality safety, and radionuclide confinement by the cladding would be retained over a nominal 50 year storage period. Allows for up to 3 mils depth of corrosion to cladding and exposed fuel. No creep rupture or rupture to cladding due to severe embrittlement. Anticipates but does not establish a limit to fission product release.

396	Singh, K.P. “Forced Gas Flow Canister Dehydration.” U.S. Patent No. 7,210,247 B2. May 2007.	Defines dryness in terms of a desired maximum water vapor pressure. Condenses water and reheats/recycles non-reactive gas. Illustrates vapor pressure expected based on number of turnovers or exchanges of cavity free volume.
397	Smith (Lords), R. E., Drying characteristics of thorium fuel corrosion products, Journal of Nuclear Materials, Vol. 328, 2004, pp. 215-224.	Abridged version of [251].
398	Smith, M. L., H. S. McKay, and D. P. Batalo, "Spent Fuel Storage Activities at the Surry Power Station," CONF-860417, Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986, pp. S-175 to S-184.	Summary of dry storage demonstration program planned under cooperatiive agreement with INEL, Virginia Power (Surry and North Anna), PNL, GNSI (CASTOR V/21 cask), Transnuclear (TN-24P cask), and Westinghouse (MC-10 cask).
399	Smith, R. B., "Pyrophoricity - A Technical Mystery Under Vigorous Attack," Nucleonics, Vol. 14 No. 12, December 1956, pp. 28-33.	Abridged version of TID-8011[400].
400	Smith, R. B., "The Fire Properties of Metallic Uranium," Monograph, TID-8011, The Industrial Atom, April 12, 1956.	Discusses experience with uranium, zirconium, and thorium fires and related incidents in the early years of the U.S. nuclear industry, referring to specific events and conditions. Observes that powders and moist conditions can be particularly problematic.
401	Solignac, Y., A. Gevorgyan, and G. Markosyan, The Spent Fuel Storage Solution in Armenia, Management of Spent Fuel from Nuclear Power Reactors, IAEA proceedings, Vienna, Austria, 19-22 June 2006, pp. 295-305.	Uses NUHOMS storage system.
402	Spedding, F. H., A. S. Newton, J. C. Warf, O. Johnson, R. W. Nottorf, I. B. Johns, and A. H. Daane, Uranium Hydride—I, Preparation, Composition and Physical Properties, Nucleonics, January 1949, pp. 4-15.	Description of uranium hydride preparation from hydrogen exposure to uranium at elevated temperature. Decomposition pressures shown as a function of temperature.

403	Spent Fuel Performance Assessment and Research, IAEA-TECDOC-1343, International Atomic Energy Agency, Vienna, Austria, 2003.	Summarizes degradation mechanisms for wet and dry storage for a wide range of SNF. Discounts any concern for pressure increase due to (decay product) helium generation. Identifies this caveat for diffusion-controlled cavity growth (DCCG) peak temperature clad failure prediction: However, cladding rupture from this mechanism [creep rupture] is a sudden, non-ductile type of failure with no early manifestation of the damage, and does not accurately identify a creep type failure. By definition, the latter is a macroscopic phenomenon that does not progress without external evidence of damage (creep strain).
404	Spent fuel storage and transport cask decontamination and modification, An overview of management requirements and applications based on practical experience, OSTI ID: 341653, IAEA-TECDOC-1081, International Atomic Energy Agency, Vienna, Austria, April 1, 1999.	German experience with thermal response drying in the TN-1300 cask.
405	Spilker, H., M. Peehs, H. Dyck, G. Kaspar, and K. Nissen, "Spent LWR Fuel Dry Storage in Large Transport and Storage Casks After Extended Burnup," Journal of Nuclear Materials, Vol. 250, 1997, pp. 63-74.	Tests zircaloy cladding for hoop stresses ranging from 80 to 150 N/mm^2 and test temperatures from 250 to 400°C. The tests lasted up to 10,000 hours. Results were incorporated into a mathematical approach to describing the creep in relation, to stress, temperature and time.
406	Spitzberg, D. B., to P. D. Hinnenkamp, U.S. Nuclear Regulatory Commission Inspection Report 050-00458/05-016; 072-00049/05-005, Docket Nos.: 50-458, 72-049, License No.: NPF-47, January 19, 2006.	An NRC inspection was conducted at River Bend Station on December 11-20, 2005 and December 28-29, 2005. No violations were identified. Discussion of fuel loading and procedure/ time to boil calculation avoiding water boiling in transfer cask. Procedure review included the Holtec forced helium drying protocol.
407	Stahl, D., and E. Siegmann, Clad Degradation – Summary and Abstraction for LA, ANL-WIS-MD-000021 REV 01 Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20070614.0002, February 2005.	Summarizes cladding degradation mechanisms. Update from [389].
408	Štefanic I. and LaVerne, J.A. 2002. "Temperature Dependence of the Hydrogen Peroxide Production in the y-Radiolysis of Water." Journal of Physical Chemistry, 106, (2), 447-452. [Washington, D.C.]: American Chemical Society. 166303. TIC: 255323.	Yield of hydrogen peroxide decreases with increasing temperature. OH radicals appear to be the sole source of H_2O_2 at short times in the gamma radiolysis of water.

409	Suikki, M., M. Warinowski, and J. Nieminen, A Drying System for Spent Fuel Assemblies, translated to Englis by P. Suominen, June 2007.	Design description and cost estimate for ambient temperature vacuum drying. For fuel with decay heat of 1370-1830 Watts. Drying ascertained by low final pressure of ~100 Pa with holding time (not specified).
410	Sumar, R. N., and S. Jonjev, Dry Storage of Irradiated CANDU Fuel at Pickering NGS, INIS-mf-14987(v.1,2), CONF-940631, annual conference of the Canadian Nuclear Association, Montreal, Canada, 5-8 June 1994.	Vacuum dried. Helium backfilled. Continous monitoring of loaded dry storage containers over 4 years for demonstration.
411	Sunder, S., and N. H. Miller, Oxidation of CANDU uranium oxide fuel by air in gamma radiation at 150°C, Journal of Nuclear Materials, Vol. 231, No. , 1996, pp. 121-131.	Oxidation of UO_2 at 150°C by oxygen, in a gamma field can lead to the formation of U_30_8 on the UO_2 surface. The rate of formation is very low. Oxidation to U_3O_8(and $UO_3 \bullet xH_20$) is strongly enhanced by the presence of water vapor. However, water vapor radiolysis, in the absence ofoxygen (or other oxidizing agents), does not cause UO_2 oxidation at 150°C.
412	Sunder, S., D.W. Shoesmith, H. Christensen, and N.H. Miller, Oxidation of UO_2 fuel by the products of gamma radiolysis of water, Atomic Energy of Canada Limited, AECL-10669, Journal of Nuclear Materials, Vol. 190, 1992, pp. 78-95.	Examines the two stage oxidation of UO_2 in radiolysis products as a function of gamma dose. First a layer of U_3O_7 forms. Subsequent dissolution of the initial layer produces soluble U^{VI} species and secondary phases such as hydrated schoepite.
413	Sunder, S., D.W. Shoesmith, R. J. Lemire, M. G. Bailey, and G. J. Wallace, The effect of pH on the corrosion of nuclear fuel (UO_2) in oxidative solutions, Atomic Energy of Canada Limited, AECL-10254, Corrosion Science, Vol. 32, No. 4, 1991, pp. 373-386.	First a thin film of U_3O_7 forms. This film achieves a steady-state thickness (-6 nm) in 5 to 10 hours; the thickness increases with an increase in pH. Over the next 10 to 100 hours hydrated phases form after which steady state dissolution occurs. In acidic solutions, the UO_2 proceeds directly to the soluble U^{VI} state without the thin film formation.
414	Supko, E., "Industry Spent Fuel Storage Handbook," EPRI Final Report 1021048, Electric Power Research Institute, July 2010.	Summarizes U.S. storage experience, demonstration projects, casks, and regulations. Discusses damaged fuel and acceptable strategies for its storage and transportation. Section on dry storage issues generically addresses protocol for moisture removal and helium backfill.
415	Survey of wet and dry spent fuel storage, IAEA-TECDOC-1100, Nuclear Fuel Cycle and Materials Section, International Atomic Energy Agency, Vienna, Austria, July 1999.	Describes storage systems with little information on drying per se. Does describe Wisconsin Power, May 28, 1996, hydrogen ignition event in VSC-24 cask during lid welding after loading and indicates no apparent fuel damage.

416	T. L. Welsh, R. B. Baker, E. W. Hoppe, A. J. Schmidt, J. Abrefah, J. M. Tingey, P. R. Bredt, and G. R. Golcar, (B. J. Makenas, ed.), Analysis of Sludge from Hanford K East Basin Canisters, HNF-SP-1201, UC-2070, September 1997.	Detailed analysis of sludge with consideration for effect on fuel handling.
417	T.C. Totemeier, R.G. Pahl, S.L. Hayes, and S.M. Frank, Metallic Uranium ZPPR Fuel: Corrosion Characteristics and Corrosion Product Oxidation Kinetics, Report ANL-98/11, UC-504 Materials, Argonne National Laboratory - West, October 1997.	Similar to [436] and [437]. Report is in two parts. History of corrosion of ZPPR plates and analytical confirmation of uranium hydride contents. Discussion of oxidation kinetics and passivation to prevent room temperature ignition.
418	Tada, E., and G. S. Frankel, Localized Corrosion Behavior of Type 304SS with a Silica Layer under Atmospheric Corrosion Environments, DT #47393, ACC: MOL.20060705.0068, 2006.	Localized corrosion of waste packages under postulated Yucca Mountain repository conditions. Silica does not affect pit initiation, but does seem to retard pit propagation compared to uncoated 304 stainless steel.
419	Tanabe, T., S. Miura, and S. Imoto, Isotope Effect in Dissociation of Uranium Hydride, Journal of Nuclear Science and Technology, Vol. 16, No. 9, Sept. 1979, pp. 690-696.	Illustrates hydride dissociation temperature and pressure relation.
420	Tasooji, A.; Einziger, R.E.; and Miller, A.K., "Modeling of Zircaloy Stress-Corrosion Cracking: Texture Effects and Dry Storage Spent Fuel Behavior," Zirconium in the Nuclear Industry, Sixth International Symposium, Vancouver, British Columbia, D. G. Franklin, and R. B. Adamson, Eds., ASTM STP 824, American Society for Testing and Materials, 1984, pp. 595-626.	Develops a model for iodine-induced stress corrosion cracking in zircaloy cladding and predicts limiting temperature for specific fuel rod conditions.
421	Taylor, P, R. J. Lemire, and D. D. Wood, The Influence of Moisture on Air Oxidation of UO_2: Calculations and Observations, Nuclear Technology, Vol. 104, December 1993, pp. 164-170.	Phase relations among solids in the UO_2-O_2-H_2O system are examined for temperatures of 25-200°C.
422	Technical Report Series No. 443, Understanding and managing ageing of material in spent fuel storage facilities, ISSN 0074–1914, STI/DOC/010/443, ISBN 92–0–105205–7, International Atomic Energy Agency, Vienna, Austria, May 2006.	Summarizes international experience with wet and dry storage of spent nuclear fuel and associated degradation issues.
423	Test Plan for High-Burnup SNF Cladding Integrity during Cask Transport Accidents following Drying, Transfer and Long-Term Storage, IPS-498-00-01, acc: ML073360044, Argonne National Laboratory, November 26, 2007.	Considers impact resistence (embrittlement) of cladding of high burnup (>45 GWd/MTU) fuel.

424	Tetenbaum, M., L. Mishler, and G. Schnizlein, Uranium Powder Ignition Studies, Nucl. Sci. Eng., Vol. 14, 1962, pp. 230-238.	Surface area dependence of uranium ignition temperature and an assessment of the limiting ignition temperature for uranium powder using the Frank-Kamenetskii theory of thermal explosions.
425	Thiel, E. C., and P. W. Fuhrman, Nuclear Materials Focus Area Fiscal Year 2002 Mid Year Review, INEEL/EXT-02-00755, May 2002.	Brief status report on radiolysis of residual water in dry storage with spent fuel.
426	Thomas, E., S. Arthur, P. Brady, P. Domski, S. LeStrange, C. Lum, and P. Mariner, In-Package Chemistry Abstraction. ANL-EBS-MD-000037, Rev. 04. Las Vegas, Nevada: Bechtel SAIC Company, ACC: DOC.20070816.0004, July 2005.	Models fuel oxidation and leach rates in the repository.
427	Thomas, L. E., and R. E. Einziger, Grain Boundary Oxidation of Pressurized-Water Reactor Spent Fuel in Air, Materials Characterization, Vol. 28, 1992, pp. 149-156.	Reiterates findings from [428] and observes that oxidation begins simultaneously from the UO_2 grain corners without indication of enhancement at fragment surfaces. UO_2 and U_3O7 can be distinguished by backscatter electron imaging to indicate rate.
428	Thomas, L. E., O. D. Slagle, and R. E. Einziger, Nonuniform oxidation of LWR spent fuel in air, Journal of Nuclear Materials, Vol. 184, 1991, pp. 117-126.	Fragments of used UO_2 fuel (H. B. Robinson - PWR) were exposed to air at 175°C and assessed by ceramography. The resultant reaction lead to the formation of U_4O_9 along the grain boundaries. Concludes that unlike unirradiated UO_2, irradiated LWR fuel does not readily form U_3O_7 or U_3O_8 at lower temperatures.
429	Thomas, L. E., R. E. Einziger, and H. C. Buchanan, Effect of fission products on air-oxidation of LWR spent fuel, Journal of Nuclear Materials, Vol. 201, 1993, pp. 310-319.	Continues work from [427]. Oxidation to U_3O_8 only occurred with exposure to substantially higher temperatures. This work also shows addition of 4 to 8 wt % Gd_2O, in unirradiated UO_2 stabilizes U_4O_9 and delays U_3O_8 formation.
430	TMI-2 Technical Bulletin, "Dewatering Canisters in Preparation for Offsite Shipment", TB 86-33 Rev. 2 December 23, 1986.	Internal project document. Loading weight and void space data for TMI-2 Canisters before shipment to Idaho.
431	Torrero, M. E., E. Baraj, J. De Pablo, J. Giménez, I. Casas, Kinetics of Corrosion and Dissolution of Uranium Dioxide as a Function of pH, Chemical Kinetics, Vol. 29, 1997, pp. 261-267.	Similar results to [413]. Suggests that the experimental methodology used can give reliable rates of uranium dioxide alteration under oxidizing conditions and at acidic conditions.
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Charleston, SC, 1998, pp. 271-278.

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437 Totemeier, T. C., R. G. Paul, S. L. Hayes, and S. M. Frank, Characterization of corroded metallic uranium fuel plates, Jounal of Nuclear Materials, Vol. 256, 1998, pp. 87-95. — History of corrosion of ZPPR plates and analytical confirmation of uranium hydride contents.

438 Totemeier, T. C., S. L. Hayes, R. G. Pahl, and D. C. Crawford, Corrosion and Pyrophoricity of ZPPR Fuel Plates: Implications for Basin Storage, ANL/ED/CP--90712, CONF-970321--6, AIChE, Houston, Texas, March 1997. — Summary only.

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440 Tözsér, S., Spent fuel management: Semi-dry storage, Scientific forum on nuclear fuel cycle issues and challenges, September 2004. — Presentation slides. Marks the successful completion of fuel drying campaign (Phase 1) identified in [441].

441	Tözsér, S., T. Hargitai, and I. Vidovszky, Encapsulation of nuclear spent fuel for semi-dry storage at the Budapest research reactor, IAEA-CN-100/81, International conference on research reactor utilization, July 2003, pp. 659-675.	Pre-heat added (130C for 40 min.), inert backfilled (2 evacuation-nitrogen purge cycles), sealed (welded), and returned to underwater storage (or sent to dry storage). Check package weight to confirm seal integrity (suggested every 5-10 years).
442	Transnuclear, Inc., for Oak Ridge National Laboratory, A cask fleet operations study, ORNL/Sub-87-06604/1, DE88 007844, March 1988.	Detailed description of the original Transnuclear PWR & BWR drying process with analysis.
443	Troutner, V. H., "Observations on the Mechanisms and Kinetics of Aqueous Aluminum Corrosion. Part 1 - Role of the Corrosion Product Film in the Uniform Aqueous Corrosion of Aluminum," Corrosion, Vol. 15, January 1959, pp. 9-12.	Compact amorphous corrosion film barrier on aluminum inhibits further corrosion. Rate varies with environment, but barrier layer thickness is independent. A second crystalline bulk corrosion layer is more permeable. The effect of alloying is on the barrier layer not the bulk.
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447	Tyfield, S. P., "Corrosion of Reactor Grade Uranium in Aqueous Solutions Relevant to Storage and Transport", Nuclear Energy, Vol. 27 No. 2, April 1988, pp. 91-98.	U metal fuel corrosion in water, hydride formation, oxidation kinetics and two constant hydrogen generation regimes.
448	U. S. Nuclear Regulatory Commission, Division of Spent Fuel Storage and Transportation, Interim Staff Guidance - 2, Rev. 1, Fuel Retrievability, February 22, 2010.	Discusses fuel packaging and handling considerations. Defines ready retrieval and normal means for retrieving fuel.

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450	U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Information Notice No. 84-72, SSINS No. 6835, Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation, September 10, 1984.	Specifies that over a period of twice the shipping time hydrogen generated must be limited to a molar quantityno more than 5% by volume (or equivalent limit for other inflammable gases) or the secondary container and cask cavity must be inerted with a diluent to keep oxygen below 5% by volume in the portion of the package with more than 5% hydrogen.
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454	Unterzuber, R., R. D. Milnes, B. A. Marinkovich, and G. M. Kubancsek, Spent Fuel Dry Storage Testing at E-MAD (March 1978 Through March 1982), PNL-4533, AESD-TME-3162, UC-85, Advanced Energy Systems Division, Westinghouse Electric Corporation, September 1982.	Test report on drywell testing, concrete silo testing, fuel assembly internal temperature testing, and air-cooled vault testing. Thermal model validation and gas sampling were done; Turkey Point fuel assemblies were stored.
455	Van Swam, L. F., A. A. Strasser, J. D. Cook, and J. M. Burger, "Behavior of Zircaloy-4 and Zirconium Liner Zircaloy-4 Cladding at High Burnup," American Nuclear Society, Proceedings of the International Topical Meeting on LWR Fuel Performance, Portland Oregon, March 1997, pp. 421-431.	Examines R. E. Ginna PWR fuel rods at 57 MWd/KgU burnup. Although hydrogen pickup rates were similar, the liner Zircaloy-4 cladding was less susceptible to corrosion (at least initially).

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458	Vinson, D. W., R.W. Deible, and R.L. Sindelar, "Impact of Degraded Al-SNF on Shipping and Basin Storage," ANS Topical Conference, Charleston, SC, 2002.	Evaluates breached fuel from Ezeiza Central Storage Facility in Argentina and its impact on shipping and storage in the basin at Savannah River. Concludes this RA-3 fuel can be shipped in a standard cask without canning.
459	Waber, J. T., 'The Corrosion Behavior of Plutonium and Uranium', Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Vol. 6, Basic Metallurgy and Fabrication of Fuels, United Nations, Geneva, 1958, pp 204-214.	Metal and alloy oxidation, humidity and hydride formation. Updates mechnanism as presented by Draley. Corrosion of Pu in dry air is 65 fold less rapid than air at 50% relative humidity (at 75°C).
460	Wasywich, K.M., J. Freire-Canosa, and S. J. Naqvi, "Canadian Spent Fuel Storage Experience:' Safety and Engineering Aspects of Spent Fuel Storage, IAEA-SM-335/4, Proceedings of IAEA Symposia, Vienna, 1994, IAEA, pp. 41-55.	Cursory overview of storage experience and capacity. Brief description of UO_2 oxidation studies.
461	Wasywich, K.M., W. H. Hocking, D. W. Shoesmith, and P. Taylor, Differences in Oxidation Behavior of Used CANDU Fuel During Prolonged Storage in Moisture-Saturated Air and Dry Air at 150°C, Nuclear Technology, Vol. 104, December 1993, pp. 309-329.	Assesses the progression of oxidation in intentionally defected CANDU fuel after 99.5 and 69 months in dry and saturated conditions respectively. Moisture promotes a more general distribution of oxidation at the UO2 grain boundaries. Dry air appears to produce a "rind" of U3O7 in the vacinity of the defect.
462	Weakley, E. A., *Interim Report* on *Concreted Uranium Fines and Chips Billet Curing Tests - A Basis for Resuming Shipment of Concreted Uranium Scrap Billets,* UNI-1454, United Nuclear Industries, Inc., May 1980.	Uranium scrap handling tests. Develops process for producing concrete billets of scrap to avoid ignition events.
463	Wertsching, A. K., Material Interactions on Canister Integrity During Storage and Transport, DOE/SNF/REP-104, Rev. 0, December 2007.	An update on concerns and degradation mechanisms flagged in [237] with emphasis on potential for canister damage. More recent work on water content, corrosion, radiolysis, fission gas release, and metal embrittlement discussed.

464	Wheeler, D., Choosing the right storage medium, Fuel Cycle, Nuclear Engineering International, April 1984, pp. 26-31.	Missing pp. 27-28. An early survey of the considerations between dry storage in air or inert gas.
465	Wiborg, J. Hanford Spent Nuclear Fuel Project Evaluation of Multi-Canister Overpack Venting and Monitoring Options During Staging of K Basins Fuel, WHC-SD-SNF-ES-017, Rev. 0, EDT/ECN 612988, December 1995.	Documentation to support drying and MCO loading operations.
466	Wilt, T., Letter Report Survey and Recommendations for Potential Drying and Sealing Issues with TAD Designs, U.S. Nuclear Regulatory Commission, Contract NRC–02–07–006, acc: ML090410523, Center For Nuclear Waste Regulatory Analyses, San Antonio, Texas, February 2009.	Repository transport, aging, and disposal in the context of existing fuel drying practices.
467	Winer, K., C. A. Colmenares, R. L. Smith, and F. Wooten, Interaction of Water Vapor with Clean and Oxygen-Covered Uranium Surfaces, Surface Science, Vol. 183, 1987, pp. 67-99.	Examines oxidation of U metal at and below room temperature. Suggests that oxygen inhibits the water vapor reaction, possibly due to preferential adsorbtion of the oxygen over the water vapor.
468	Wolfram, J.H., R. E. Mizia, R.E., and W. J. Dirk, Microbial Sampling of Aluminum Clad Spent Nuclear Fuel, Idaho National Engineering Laboratory, NACE International, Corrosion 98, Paper #162, 1998.	All samples taken from CPP-603 basin tested positive for microbes, including SNF cladding. Microenvironments exist in fuel storage supporting anaerobic (sulfate reducing) bacteria. These conditions may favor microbially influenced corrosion. In the transfer of aluminum clad SNF to dry storage, biofilms and the microbial cells will accompany the fuel. Their metabolic activity should be taken into consideration.
469	Wolfram, J.H.; Mizia, R.E.; Jex, R.; Nelson, L.; and Garcia, K.M. 1996. The Impact of Microbially Influenced Corrosion on Spent Nuclear Fuel and Storage Life. INEL-96/0335. Idaho Falls, Idaho: Idaho National Engineering Laboratory, Lockheed Martin Idaho Technologies Company. 165268. ACC: MOL.20030925.0039.	Literature search summary on types of microbes and mechanisms for their influence on corrosion. No evidence that zirc alloys are susceptible, but aluminum and steels can be. Extent of damage due to MIC in dry storage is unknown and but expected to be dependent upon the presence, form and quantity of moisture.

470	Wood, D. H., S. A. Snowden, H. J. Howe, L. L. Thomas, D. W. Moon, H. R. Gregg, and P. E. Miller, Letter to the Editors Regarding the chemistry of metallic uranium stored in drums, Journal of Nuclear Material, Vol. 209, 1994, pp. 113-115.	Investigation prompted by a mild explosion of a drum of iron clad depleted uranium rods opened after a period of 5 years. After the event the drum was re-sealed, overpackaged, and transported to a safer location. Gas analysis (83 days later) revealed an atmosphere of 75% nitrogen and 25% hydrogen in the drum. Silica gel dessicant was not useful in the prevention of the uranium-water reaction. Pine wood brace, plywood, and cardboard tubes in the package may have been the moisture source.
471	Woodley, R. E., "Considerations to the Dry Storage of LWR Fuel Rods Containing Water," NUREG/CR-3658, TI84 015220, HEDL-TME-84-14, Hanford Engineering Development Laboratory, June 1984.	Fuel rod and storage canister pressurization was considered, as well as cladding and fuel oxidation to assess radiological containment for "worst-case" conditions.
472	Wronkiewicz, D. J., J. K. Bates, S. F. Wolf, and E. C. Buck, Ten-year results from unsaturated drip tests with UO_2 at 90°C: implications for the corrosion of spent nuclear fuel, Journal of Nuclear Materials, Vol. 238, 1996, pp. 78-95.	UO_2 oxidation rate is rapid over the first 2 years and then slows considerably over the 2 to 10 year period. The rapid oxidation is characterized by grain boundary oxidation and spallation of grains. The development of a "dense mat of alteration phases" lead to less spallation and a lower reaction rate. The behavior is similar to that of geologic uranium mineral systems.
473	WWER-440 Fuel Rod Experiments under Simulated Dry Storage Conditions, IAEA-TECDOC-1385, International Atomic Energy Agency, Vienna, Austria, April 2004.	Data are gathered and used to calculate the pressure in the test rods and the stresses in the cladding. From the results, it can be concluded that: After thermal testing in argon environment at 350°C for two months and then at 390°C for another two months, the technical state of the fuel rods did not show any detectable changes. For near constant temperature, cumulative strain does not exceed 1% in 50 years (secondary creep rate). Cumulative cladding hoop strain (approximately 0.02%, too low for concern) under a decreasing temperature condition indicates that the cladding creep is saturated after 10 years of dry storage.
474	Zalkind, S., R. Eshkenazy, S. Harush, D. Halperin, D. Moreno, E. Abramov, and A. Venkert, Stress corrosion cracking of U-0.1% Cr in humid helium atmosphere, Journal of Nuclear Material, Vol. 209, 1994, pp. 169-173.	Oxidation in humid helium over three years. Hydride phase was observed, mainly beneath the oxide layer and penetrated into the metal matrix as needle-like forms.