ISI OF DRY STORAGE CASKS FOR SPENT NUCLEAR FUEL: LEVERAGING KNOWLEDGE FROM REACTORS

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The need for clear guidance on inservice inspection (ISI) requirements for dry cask storage systems (DCSSs) is motivated by several systems reaching the end of their initial license term. Although there are significant differences between systems for dry storage of spent nuclear fuel and operating reactors, knowledge can be leveraged from ISI of reactors to the extent that it is appropriate. This paper provides an overview of characteristics and examination requirements for DCSSs as well as an overview of the ASME Boiler and Pressure Vessel Code Section XI ISI requirements for reactor components and NDE performance data based on inspection of reactor components. The analysis provides some insights with regard to suitable ISI requirements for DCSS components but also indicates that there are gaps with respect to application of these requirements directly to DCSS components.

I. INTRODUCTION

Systems for dry storage of spent nuclear fuel may be initially licensed up to 40 years, after which the license may be renewed for periods up to 40 years, according to Title 10 of the Code of Federal Regulations (10 CFR), Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste." The need for clear guidance on inservice inspection (ISI) requirements for these systems is motivated by several systems reaching the end of their initial license term. Although there are significant differences between dry cask storage systems (DCSSs) for spent fuel and operating reactors, knowledge can be leveraged from ISI of reactors to the extent that it is appropriate. Examples of knowledge that can be leveraged towards dry storage systems include the concept of performance demonstrations and information gained on the performance of non-destructive examination (NDE) technologies and systems. However, there is also a need to modify or tailor the framework developed for reactors to meet the specific requirements of DCSSs. In addition, detailed assessments of DCSSs reveal potential challenges to the deployment of NDE technologies for inspection. Cognizance of these specific requirements and challenges and their influence on the effectiveness of NDE are necessary for the development of effective ISI requirements for DCSSs. This paper provides an overview of requirements for examination of DCSS canisters as well as an overview of ISI requirements for reactors. This is followed by a summary of performance information for NDE technologies commonly deployed for ISI of reactor components and an overview of the approach to qualification of NDE systems for reactors. A brief discussion of ISI considerations for dry storage system canisters is then presented. Finally, the paper ends with some concluding remarks.

II. REQUIREMENTS FOR EXAMINATIONS OF DRY STORAGE SYSTEMS

Several requirements related to instrumentation used to perform examinations of DCSS canisters are outlined by Meyer et al.[1] In particular, the report documents constraints related to accessibility and to the environment for vertical and horizontal systems. In addition to target aging degradation mechanisms, these constraints are likely to influence ISI and NDE strategies.

II.A. Canister Degradation Mechanism

Atmospheric stress corrosion cracking (SCC) is considered a potential form of degradation in DCSS canisters located in coastal areas,[2] where they may be exposed to chloride salts, and also inland areas where canisters may be exposed to corrosive species from industrial, agricultural, and commercial activities.[3] Although no instances of atmospheric SCC in DCSS canisters have been reported, instances of similar degradation in other components in the nuclear power industry have been observed, where it is referred to as outer diameter SCC (ODSCC).[4] Additionally, experimental investigations to date have not been able to rule out the possibility of atmospheric SCC occurring in DCSS canisters located in coastal environments.[5, 6]

Like other forms of SCC, atmospheric SCC can occur when the conditions of a susceptible material, corrosive environment, and presence of tensile stress are satisfied. Dry storage canisters are typically fabricated from stainless steel (i.e., 304, 304L, 316, or 316L). The susceptibility of these materials to SCC caused by chromium depletion in the heat-affected zones (HAZ) of welds is well known. A typical canister includes welds to the base plate and lid, as well as a circumferential weld at its mid-section, and longitudinal welds. Crevice conditions may also contribute significantly to the occurrence of atmospheric SCC as crevices can provide a mechanism for the accumulation and concentration of corrosive species on the canister surface. Crevice conditions caused by pipe clamps have been observed with instances of ODSCC discovered at power plants.[4] In dry storage systems, crevice conditions will occur where the canister is in contact with supporting structures.

NRC Information Notice 2012-20[2] documents previous cases of atmospheric SCC of welded stainless steel piping systems and tanks at operating reactor locations. Atmospheric SCC growth rates determined from operational experience are observed up to 2.9×10^{-11} m/s for components at ambient temperatures. In addition to atmospheric SCC, degradation mechanisms such as pitting corrosion, crevice corrosion, and micro-biologically induced corrosion (MIC) are considered plausible, although the perception is that the greatest risk of canister penetration due to aging mechanisms is through atmospheric SCC.[7] However, pitting corrosion can have additional consequence to canister integrity by providing potential initiating sites for atmospheric SCC.

II.B. Accessibility Constraints

Accessibility to the surface of dry storage system canisters is physically hampered by their positioning inside of thick concrete over-packs. In vertical systems, this can result in a small annulus around the circumference of the canister through which the canister surface may be accessed (Fig. 1). The size of this annulus may vary nominally up to a few inches. In some cases, if the canister is tilted or not aligned at perfect center, there may be contact between the canister surface and the liner of the inner wall of the over-pack. Access to this annulus could be achieved by maneuvering through the outlet ventilation ports of the over-pack (requiring one or more 90° turns) or by removal of the over-pack lid.

Horizontal systems are slightly different in that the round canister is placed inside of a rectangular-shaped over-pack cavity (Fig. 2). Access to the canister surface still requires maneuvering through ventilation ports of the over-pack (requiring one or more 90° turns) or by removal of the front shield door, which provides an annular gap (typically < 1 in.) for access around the circumference of the canister.

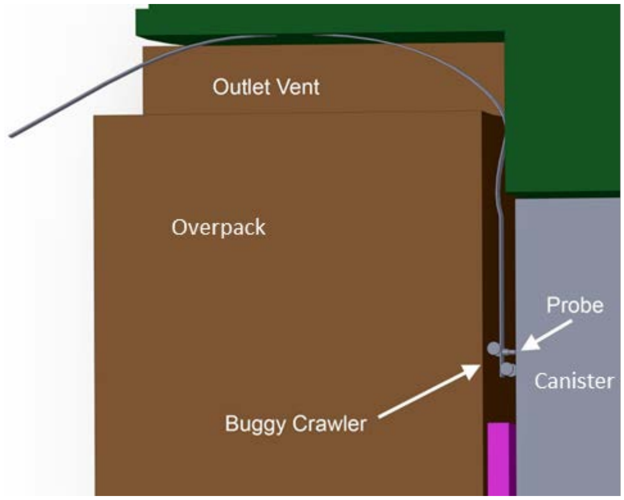

Fig. 1. Illustration of NDE accessibility to canister surface through outlet vent in vertical dry storage system.[1]

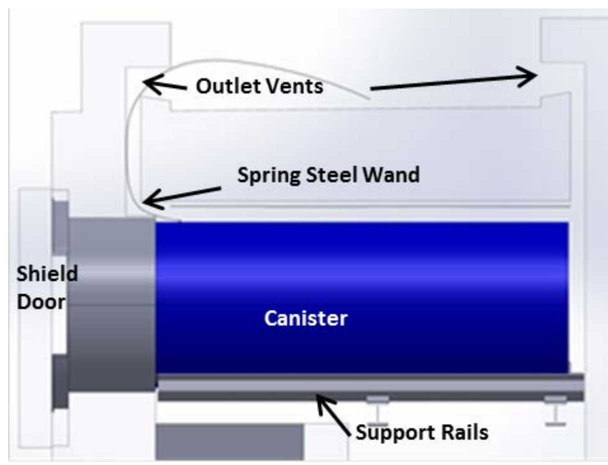

Fig. 2. Illustration of NDE accessibility to canister surface through outlet vent in horizontal dry storage system.[1]

II.C. Environmental Constraints

The environmental conditions (i.e., temperature and radiation exposure) near dry storage system canisters can influence the application of technologies for NDE. Thermal modeling of both vertical and horizontal systems predict bounding temperatures of approximately 225°C at the time of initial loading and a decrease to approximately 150°C after 30 years (Figs. 3 and 4).[1] In addition, conservative MCNPX (Monte Carlo N-Particle eXtended) simulations predict a gamma radiation exposure of 2.7×10^{4} R/hr at 5 years of storage and decreasing to 1.2×10^{4} R/hr after 15 years of storage.[1] The horizontal system scenario in Fig. 3 considers a canister containing 32 PWR fuel assemblies with an initial heat load of 41 kW. The vertical system scenario in Fig. 4 assumes 32 PWR fuel assemblies at an initial heat load of 30 kW.

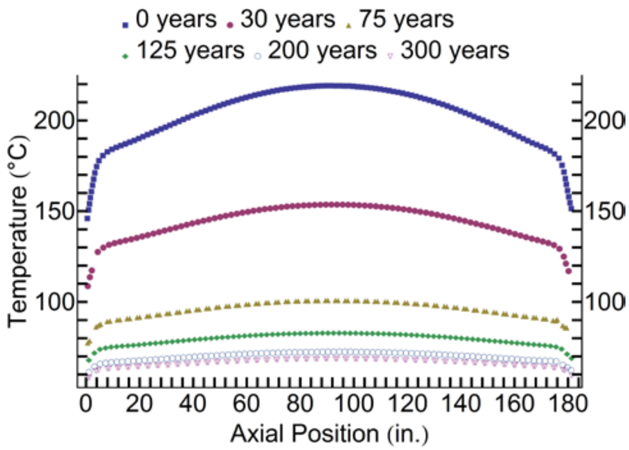

Fig. 3. Temperature profiles along the length of a horizontal canister from loading to 300 years of storage assuming 32 PWR fuel assemblies and initial heat load of 41 kW.[1]

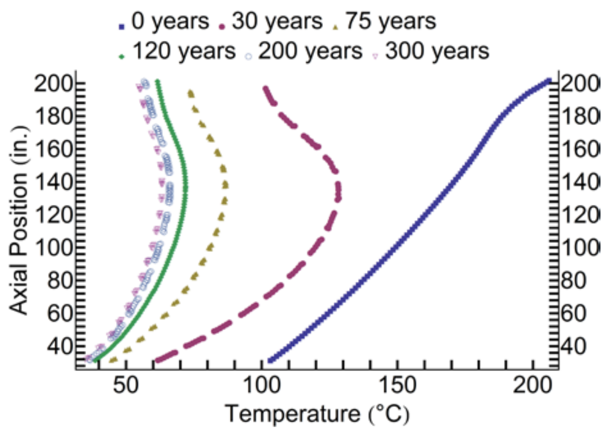

Fig. 4. Temperatures profiles along the length of a vertical canister from loading to 300 years of storage assuming 32 PWR fuel assemblies and initial heat load of 30 kW.[1]

Temperature not only impacts the ability to deploy examination technologies, but also influences susceptibility to chloride-induced degradation. Deliquescence of chloride particles on the DCSS canister surfaces can promote atmospheric SCC below a threshold temperature (~85°C).[5] The temperature profiles in Figs. 3 and **4** provide bounding estimates for design basis heat loads. Thus, surfaces of canisters will generally be expected to drop below the threshold temperature for deliquescence faster than what these figures indicate, with portions of some canisters remaining below this threshold, even at loading. Figs. 3 and **4** do indicate that ends of the canisters are likely to drop below the threshold before the middle of the canisters.

III. ISI REQUIREMENTS FOR REACTORS

Rules for the ISI of nuclear power plant components are provided by Division 1 of Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code). The rules and regulations outlined by the Code are mandatory and are enforceable in the USA through Title 10 of the Federal Code of Regulations, Part 50 (10 CFR 50). The requirements of the Code can be generic and more specific guidelines can be issued by industry or owner's groups (e.g., Nuclear Energy Institute [NEI] and Electric Power Research Institute [EPRI] through the Boiling Water Reactor Vessels and Internals Project [BWRVIP] and Pressurized Water Reactor [PWR] Materials Reliability Program [MRP]).

Division 1 of Section XI is divided into several subsections including:

- Subsection IWA – General Requirements
- Subsection IWB – Class 1 Components
- Subsection IWC – Class 2 Components
- Subsection IWD – Class 3 Components
- Subsection IWE – Class MC and Metallic Liners of Class CC Components
- Subsection IWL – Class CC Concrete Components

A summary of Subsection IWB, IWC, IWD, IWE, and IWL requirements is included in Table 1. Classes 1, 2, and 3 are pressure vessel components distinguished by safety significance, while MC and CC refer to metal containment and concrete containment components, respectively. The general requirements in Subsection IWA are organized into six articles including IWA-2000, "Examination and Inspection." Article IWA-2000 defines examination methods as surface, volumetric, and visual. Visual methods are further classified as VT-1, VT-2, and VT-3. VT-1 examinations are to be performed to detect discontinuities on the surface of components such as cracking and corrosion. VT-2 examinations are performed to detect evidence of component leakage, while VT-3 examinations are performed to observe the general mechanical and structural condition of a component. Surface examinations are conducted to detect the presence of surface discontinuities and can be performed using magnetic particle testing (MT), liquid penetrant testing (PT), eddy current testing (ET), or ultrasonic testing (UT). Finally, volumetric examinations are conducted to detect the presence of discontinuities through the volume of a component and can be performed using radiographic testing (RT), UT, ET, and acoustic emission testing (AET). Rules for conducting an examination with the various methods are provided in Section V of the ASME Code. In addition, alternative examination methods may be used provided they are demonstrated to be equivalent or superior with respect to performance.

TABLE I. Tabulation ASME B&PV Code Section XI Division 1 Requirements for ISI

ASME B&PV Code Section XI Division 1 Subsection	Component Type	Type of Examination	Frequency of Examinations	Acceptance Criteria	Extent of Examinations
IWB	Pressure Vessel (Class 1)	Volumetric and Surface Visual for pressure retaining bolting and core support structures	Every 10 years	Allowable planar flaws based on structural integrity analysis	Category B-J (25%) Category B-D (100%)
IWC	Pressure Vessel (Class 2)	Surface w/ some Volumetric	Every 10 years		Category C-F-1 (7.5%)
IWD	Pressure Vessel (Class 3)	Visual w/ some Surface	Every 10 years		---
IWE	Metal Containment (MC)	General Visual, Detailed Visual, Ultrasonic	Every 10 years	No crack indications, 10% wall thickness reduction	Category E-A (100% of accessible surfaces)
IWL	Concrete Containment (CC)	General Visual, Detailed Visual	Every 5 years	Judgment of Responsible Engineer	All accessible surfaces Inaccessible sur-faces (e.g., covered by liners, obstructed by backfill, etc.) are generally exempt

A schedule is applied to examinations, with the specification that all required examinations for a given category be distributed over three inspection intervals within a 10-year period. The Code establishes a sampling strategy for the inspection of components based on their importance to safety. Systems in direct contact with reactor coolant (Class 1) receive the most stringent and populated inspections because it is important to maintain the coolant inventory and contain radioactivity. In this case, volumetric and surface examinations are performed on a relatively large percentage of components. Class 2 systems, such as secondary systems that remove primary heat or are necessary to actuate in case of emergency, also must be examined, but smaller populations of components are generally inspected than on Class 1 systems. In addition, most Class 2 components require only surface examinations with some selected volumetric examinations. Class 3 systems, such as the component cooling water supply, service water, and steam conversion systems, which provide support functions, are inspected primarily with visual (VT) examinations, selected surface examinations, and no volumetric examinations.

Inspection requirements of concrete sections of the containment, metallic reinforcement systems, and post-tensioned tendon systems are covered by Article IWL-2000. IWL-2000 specifies two types of visual inspections for these components referred to as a general visual inspection and a detailed visual inspection. The purpose of detailed visual inspections is to assess the severity of degradation identified during the general visual inspection. Article IWL-2000 specifies that components must be inspected at 1, 3, and 5 years following containment construction and every 5 years thereafter. Rules for the inspection of metal containments and metallic liners are outlined in Subsection IWE of the ASME Code, Section XI. The Code specifies general visual inspections for all surfaces accessible to inspection. Paragraph IWE-1241 specifies that surface regions subject to accelerated corrosion require additional augmented inspections by either more detailed visual or ultrasonic means. Components covered by IWE are also subject to the 10-year period for examinations that applies to IWB, IWC, and IWD components. Generally, all accessible surfaces require examination for IWE and IWL components.

ISI requirements for reactor internals are also covered by Subsection IWB, generally requiring VT-1 exams for components in the beltline region and VT-3 exams for components outside of the beltline region. Industry provides additional guidance for performing ISI of reactor internals through BWRVIP and MRP documents.

Section XI of the ASME B&PV Code also provides acceptance criteria for NDE examinations. Acceptance criteria are used to define whether or not a component is observed to be in acceptable condition for continued operation. If not, then action must be taken to restore the component to operable condition, for example, through repair or replacement for continued operation to be allowed. As Table I shows, the acceptance criteria for many pressure vessel components (Classes 1, 2, and 3) allow for the existence of planar flaws (i.e., cracks) below some critical depth. Acceptance criteria for Class MC (subsection IWE) components, which are examined primarily by VT, do not allow for any crack indications and allow for 10% wall thickness reduction due to corrosion. Finally, determination of the acceptability of NDE responses observed for class CC components is left to the discretion of the responsible engineer.

IV. NDE PERFORMANCE

Information has been collected on the performance of NDE methods for nuclear power plant components over several decades by the USA and the larger international community. The performance of ultrasonic testing (UT) has been studied for various applications through round robin studies including the European Programme for Inspection of Steel Components III Austenitic Stainless Steel Testing (PISC III AST) studies, the Piping Inspection Round Robin (PIRR),[8] the Mini Round Robin (MRR),[9] Program for Inspection of Nickel Alloy Components (PINC),[10] and the ongoing Program for Assessing the Reliability of Emerging Nondestructive Technologies (PARENT).[11]

Information has been collected on the performance of UT for inspection of stainless steel piping with wall thickness ranging from 12.7 mm to 15 mm through several of these studies. A comprehensive analysis and comparison of results from the PIRR, MRR, and PISC III AST studies indicates good reliability (~ 90% detection rate) for detection of flaws that are 10-mm deep.[9] It is important to note that these studies did not include some of the more advanced UT techniques, such as phased-array UT, that are in common use today.

The performance of ET has also been assessed in some of these studies, including PINC and PARENT. ET results from PINC on dissimilar metal weld components indicate significantly better performance than UT for detection of shallow flaws. Good reliability was observed for detection of flaws with 5-mm depth (88% detection rate).[10]

VT has not been assessed in the above studies, which all focused on Class 1 components. More recently, studies of VT have been initiated to assess performance for boiling water reactor (BWR) internal component inspection applications. VT performance is gauged with respect to crack opening displacement (COD) because of its significant influence on VT performance. An initial assessment of visual testing indicates that VT performance is reliable for detecting cracks with COD ≥ 100 μm while cracks with COD ≤ 20 μm can be difficult to detect, even under favorable conditions.[12, 13] Performance within this range is dependent on several different factors, such as lighting conditions, surface conditions, scan rate, etc. A systematic evaluation of service-induced crack characteristics has been performed by review of failure analysis reports.[14] This analysis indicates that the mean COD for SCC cracks range from 16–30 μm. Cooperative research on the performance of VT involving Pacific Northwest National Laboratory (PNNL), the Nuclear Regulatory Commission (NRC), and EPRI is ongoing.

V. QUALIFICATION

Nondestructive examinations performed during ISI are required to be performed with a qualified system where the system refers to the personnel, procedure, and equipment. Appendix VIII of Section XI of the ASME Code outlines requirements for performance demonstration of systems for UT examinations. The performance demonstration process subjects a system to a practical test, in which acceptable performance must be observed as a condition to obtaining qualified status. The practical tests are conducted on mock-ups of field components with realistic and challenging flaws. The concept for qualification by performance demonstration arose, in part, because of the results of the early round-robin studies (i.e., PISC, PIRR, and MRR), which showed that NDE was highly skill-dependent. Performance demonstration is implemented in industry as the Performance Demonstration Initiative (PDI), which is administered by EPRI. As part of the qualification process, essential parameters of the system must be identified where essential parameters refer to those system variables that have a significant influence on performance. If an essential parameter is changed to a value beyond the qualified range, requalification is necessary.

VI. DISCUSSION

This section attempts to assess the appropriateness of many of the ISI requirements summarized in Table I for reactors for application to DCSS components. This includes a discussion of acceptance criteria, acceptable examination methods, and examination frequencies. In addition, deployment issues are mentioned with respect to implications for qualification testing and additional parallels are drawn with reactor applications. Finally, ISI requirements for DCSS over-pack components are briefly discussed.

TABLE II. Comparison of ASME B&PV Section XI Subsections with DCSS Components.

		Materials		
		Austenitic Stainless Steel	Carbon Steel	Concrete
Functions	Containment of air-born radioactive particles	DCSS canister	IWE	IWL
	Maintaining reactor core coolant inventory	IWB, IWC, IWD	---	---
	Physical protection from weather and external events	---	---	IWL, DCSS over-pack

The function of DCSS canisters may bear more similarity to class MC components and metallic liners of class CC components than to Class 1, 2, and 3 pressure vessel components. In addition, MC components and metallic liners of class CC components can be susceptible to pitting and crevice corrosion as well as MIC.[15] However, MC components and metallic liners of class CC components are usually made of carbon steel, whereas DCSS canisters are fabricated from 300 series stainless steel so that different assumptions are appropriate regarding degradation growth rates and mechanical properties of the materials. In general, acceptance criteria for Class 1, 2, and 3 components include bounds for tolerable flaw sizes that have been determined based on rigorous structural integrity analysis. For MC components and metallic liners of CC components, acceptance criteria are defined for material loss by corrosion. In addition, acceptance criteria for these components indicate that corrective actions or additional evaluation is triggered by any direct observation of cracking, regardless of extent observed. A comparison of reactor component class function and materials with DCSS components is provided in Table II.

Studies indicate that the type of visual inspections called for in IWE would not be adequate for direct observation of atmospheric SCC in DCSS canisters. A visual examination may still be useful for detection of atmospheric SCC assuming significant crack formation is preceded by more obvious signs such as corrosion products or pitting. The extents to which such signs will accompany crack formation are not fully understood. Further, the extents to which such signs must be present for reliable detection by visual detection are not well defined. Even so, such signs will not be visible in crevice areas, where chlorides have a chance to collect and become concentrated. Performance data indicates that ET and UT may be effective at detecting atmospheric SCC before canister penetration. In this case, acceptance criteria could be defined based on documented performance data, anticipated atmospheric SCC growth rates, and frequency of required examinations. However, ET would also not be suitable for detecting degradation in crevice conditions due to inaccessibility.

Components covered under Subsections IWB, IWC, IWD, and IWE are inspected once every 10 years. This inspection period is not directly tied to growth rates for relevant degradation mechanisms in these components. Thus, a potentially more rational inspection period can be determined for DCSS canisters based on knowledge of possible growth rates for atmospheric SCC and other potentially applicable degradation mechanisms such as pitting corrosion, crevice corrosion, and MIC.

Accessibility for inspection of canister surfaces is challenged by the geometric and environmental constraints discussed in Sections II.B and II.C. Environmental conditions will vary depending on the time after loading that the canister is inspected and the initial load conditions. In many cases, environment may not pose a very significant obstacle to deployment of instrumentation for examinations. Nevertheless, awareness of the environmental conditions near the canister surface in relation to environmental ratings for instrumentation proposed for examination represents good ISI planning practice. Qualification of systems for inspecting DCSS canisters will require mock-ups that replicate field conditions (i.e., geometric constraints and temperature) to an appropriate extent and which include realistic and challenging flaws. Environmental conditions and access constraints near the canister make it likely that remote systems will be deployed for examination.

The geometric challenges bear some resemblance to examination challenges encountered with reactors such as examination of BWR core shrouds behind jet pumps in which only a few inches of clearance is available for access by a probe. In addition, examinations of control rod drive mechanism (CRDM) penetration tubes behind thermal sleeves have motivated development of “blade” probes, which have very low profiles and can contain several UT and ET sensors (see Fig. 5).[16] Deploying such instruments on DCSS canisters with minimum modifications would likely require removal of over-pack lids or shield doors.

In addition to DCSS canisters, aging of DCSS concrete over-pack components must be considered as systems begin to operate beyond the initial licensing period. For reactor concrete containments, ISI requirements are covered by Subsection IWL. Over-pack systems share several functional characteristics with reactor concrete containments such as physical protection from external events and radiation shielding. However, over-pack systems do not serve a containment function. Other similarities may include the presence of thick sections, metal liners, and below-grade components.

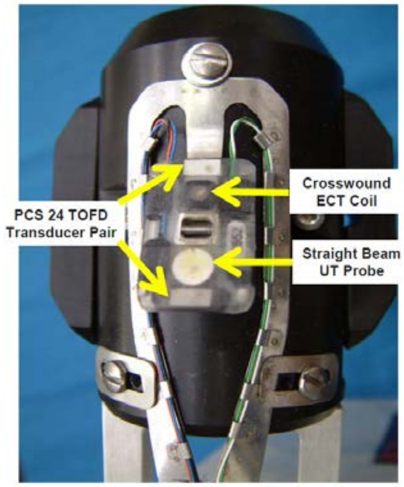

Fig. 5. Photograph of low-profile "blade" probe for examination of CRDM penetration tubes behind thermal sleeves.[16]

Concrete is subject to several forms of degradation initiating on the surface and internally. The nature of these degradation effects includes cracks, voids, delamination, as well as changes to continuously distributed properties such as strength and porosity. Thus, inspections of over-pack components will require the application of technologies to examine the surface and internal volume, including embedded steel. Shah and Hookham[15] review potential aging degradation mechanisms for class CC components in reactors and applicable inspection and testing techniques are reviewed by Norris et al.[17] Additional guidance for evaluating nuclear safety-related structures is included in American Concrete Institute document ACI 349.3R, which is also referenced in Subsection IWL. ACI 349.3R includes information about examination technologies, possible forms of degradation, and evaluation criteria.

Existing literature indicates that while there are several examination technologies to apply to assess the extent of degradation of concrete and embedded steel components, below grade or buried components still pose a significant challenge due to accessibility difficulties. In addition, the need for better guidance to interpret NDE results for concrete components has been cited by Naus.[18] The vagueness of acceptance criteria for IWL components (class CC) supports this conclusion.

VII. CONCLUSIONS

The development of ISI programs for dry cask storage systems can benefit from the ISI experience from reactors. An overview of ISI requirements for reactors and NDE performance data based on inspection of reactor components provides insight with regard to developing suitable requirements for DCSS components. This overview indicates that although existing Section XI requirements cover components of similar materials and performing similar functions to DCSS components, there are gaps with respect to application of these requirements directly to DCSS components. For canisters, gaps exist because none of the component classes fully address the specific combination of materials, function, and environmental conditions relevant to DCSS canisters. DCSS over-packs draw significant parallels with class CC reactor components (IWL). While existing codes and standards could be applied to DCSS over-packs, there may be opportunities to gain efficiency by developing more detailed and specific guidance for interpreting the results of an examination performed on DCSS over-packs.

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REFERENCES

1. R. MEYER, A. PARDINI, J. CUTA, H. ADKINS, A. CASELLA, A. QIAO, M. R. LARCHE, A. DIAZ, and S. R. DOCTOR, *NDE to Manage Atmospheric SCC in Canisters for Dry Storage of Spent Fuel: An Assessment*, PNNL-22495, Pacific Northwest National Laboratory, Richland, Washington (2013).
2. NRC, *Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters*, NRC Information Notice 2012-20, U.S. Nuclear Regulatory Commission (NRC), Washington, D.C. (2012).
3. X. HE, R. PABALAN, T. MINTZ, G. OBERSON, D. DUNN, and T. AHN, "Scoping Study of Effect of Salts on Non-Coastal Particulate Matter on Stress Corrosion Cracking of Type 304 Stainless Steel," *CORROSION 2013*, Orlando, FL, March 17-21, 2013, NACE (2013). Paper No. 2547.
4. NRC, *Contaminants and Stagnant Conditions Affecting Stress Corrosion Cracking in Stainless Steel Piping in Pressurized Water Reactors*, NRC Information Notice 2011-04, U.S. Nuclear Regulatory Commission (NRC), Washington, D.C. (2011).
5. L. CASERES and T. S. MINTZ, *Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments*, NUREG/CR-7030, U.S. Nuclear Regulatory Commission, Washington, D.C. (2010).

6. X. HE, T. MINTZ, R. PABALAN, L. MILLER, and G. OBERSON, *Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts*, NUREG/CR-7170, U.S. Nuclear Regulatory Commission, Washington, D.C. (2014).
7. EPRI, *Failure Modes and Effects Analysis (FMEA) of Welded Stainless Steel Canisters for Dry Cask Storage Systems*, EPRI Report 3002000815, Electric Power Research Institute (EPRI), Palo Alto, California (2013).
8. P. G. HEASLER and S. R. DOCTOR, *Piping Inspection Round Robin*, NUREG/CR-5068, PNL-10475, U.S. Nuclear Regulatory Commission, Washington, D.C. (1996).
9. P. G. HEASLER and S. R. DOCTOR, *A Comparison of Three Round Robin Studies on ISI Reliability of Wrought Stainless Steel Piping*, NUREG/CR-6795, PNNL-13873, U.S. Nuclear Regulatory Commission, Washington, D.C. (2003).
10. S. E. CUMBLIDGE, S. R. DOCTOR, P. G. HEASLER, and T. T. TAYLOR, *Results of the Program for the Inspection of Nickel Alloy Components*, NUREG/CR-7019; PNNL-18713, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C. (2010).
11. B. G. BRAATZ, S. E. CUMBLIDGE, S. R. DOCTOR, and I. PROKOFIEV, "Primary Water Stress Corrosion Cracks in Nickel Alloy Dissimilar Metal Welds: Detection and Sizing Using Established and Emerging Nondestructive Examination Techniques," *Third International Conference on NPP Life Management (PLiM) for Long Term Operations (LTO)*, Salt Lake City, Utah, May 14-18, 2012, International Atomic Energy Agency, Vienna, Austria (2012). IAEA-CN-194-025.
12. S. E. CUMBLIDGE, M. T. ANDERSON, and S. R. DOCTOR, *An Assessment of Visual Testing*, NUREG/CR-6860, PNNL-14635, U.S. Nuclear Regulatory Commission, Washington, D.C. (2004).
13. S. E. CUMBLIDGE, M. T. ANDERSON, S. R. DOCTOR, F. A. SIMONEN, and A. J. ELLIOT, *A Study of Remote Visual Methods to Detect Cracking in Reactor Components*, NUREG/CR-6943, PNNL-16472, U.S. Nuclear Regulatory Commission, Washington, D.C. (2007).
14. J. WÅLE, *Crack Characterisation for In-service Inspection Planning - An Update*, SKI Report 2006:24, Swedish Nuclear Power Inspectorate, Stockholm, Sweden (2006).
15. V. N. SHAH and C. J. HOOKHAM, "Long-term Aging of Light Water Reactor Concrete Containments," *Nucl. Eng. Des.*, 185, (1998).
16. J. P. LAREAU, D. C. ADAMONIS, S. C. MCKINNEY, R. P. VESTOVICH, M. W. KIRBY, R. S. DEVLIN, M. MELBI, and F. D'ANNUCCI, "Reactor Vessel Head Penetration Inspection Technology Past, Present and Future," in *Proceedings of the Conference on Vessel Penetration Inspection, Crack Growth and Repair*, p. 79, T. S. MINTZ and W. H. CULLEN, JR., Eds., U.S. Nuclear Regulatory Commission, Washington, D.C. (2005). NUREG/CR-0190, Vol. 1.
17. W. E. NORRIS, D. J. NAUS, and H. L. GRAVES III, "Inspection of Nuclear Power Plant Containment Structures," *Nucl. Eng. Des.*, 192, 303 (1999).
18. D. J. NAUS, "The Management of Aging in Nuclear Power Plant Concrete Structures," *Journal of the Minerals, Metals and Materials Society*, 61, (2009).