

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

[Docket No. PRM-50-109; NRC-2014-0257]

Improved Identification Techniques Against Alkali-Silica Reaction (ASR) Concrete Degradation at Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Petition for rulemaking; denial.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is denying a petition for rulemaking (PRM), PRM-50-109, dated September 25, 2014, submitted by the C-10 Research and Education Foundation (C-10 or the petitioner). The petitioner requests that the NRC amend its regulations to provide improved identification techniques for better protection against concrete degradation due to alkali-silica reaction (ASR) at U.S. nuclear power plants. The petitioner asserts that reliance on visual inspection will not adequately identify ASR, confirm ASR, or provide the current state of ASR damage without petrographic examination. The NRC is denying the petition because existing NRC regulations and NRC oversight activities provide reasonable assurance of adequate protection. Specifically, existing NRC regulations are sufficient to ensure that concrete degradation due to ASR will not result in unacceptable reductions in the structural capacity of safety-related structures at nuclear power plants.

DATES: The docket for the petition for rulemaking PRM-50-109 is closed on **[INSERT DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**.

ADDRESSES: Please refer to Docket ID NRC-2014-0257 when contacting the NRC about the availability of information regarding this petition. You can obtain publicly-available documents related to the petition using any of the following methods:

- **Federal Rulemaking Web site:** Go to <http://www.regulations.gov> and search on the petition Docket ID NRC-2014-0257. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions, contact the individual listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at <http://www.nrc.gov/reading-rm/adams.html>. To begin the search, select "[ADAMS Public Documents](#)" and then select "[Begin Web-based ADAMS Search](#)." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in the Supplementary Information section. For the convenience of the reader, instructions about obtaining materials referenced in this document are provided in Section V, Availability of Documents.

- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

FOR FURTHER INFORMATION CONTACT: Meena Khanna, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-2150, e-mail: Meena.Khanna@nrc.gov, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001.

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I. The Petition

On September 25, 2014, C-10, with assistance from the Union of Concerned Scientists (UCS), submitted a petition for rulemaking to the NRC (ADAMS Accession No. ML14281A124). The NRC docketed the petition on October 8, 2014, and assigned Docket No. PRM-50-109 to the petition. The petitioner requests that the NRC amend its applicable regulations to provide improved identification techniques for better protection against concrete degradation due to ASR at U.S. nuclear power plants. Specifically, the petitioner requests that the NRC require that all licensees comply with American Concrete Institute (ACI) Committee Report 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" (ACI 349.3R), and American Society for Testing and Materials (ASTM) Standard C856-11, "Standard Practice for Petrographic Examination of Hardened Concrete" (ASTM C856-11).

The petitioner previously submitted a request for enforcement action in accordance with § 2.206 of title 10 of the *Code of Federal Regulations* (10 CFR), “Requests for action under this subpart,” specific to Seabrook Station (ADAMS Accession No. ML16006A002). That petition was rejected by the NRC in a letter dated July 6, 2016 (ADAMS Accession No. ML16169A172), because the request addressed deficiencies within existing NRC rules, similar to those raised in PRM-50-109. While discussion of Seabrook Station (the only nuclear power plant with a documented occurrence of ASR) is included below in response to the petitioner’s comments, the NRC’s focus in this denial is on the generic request that the NRC require that all licensees comply with ACI 349.3R and ASTM C856-11. The NRC is conducting its licensing and oversight responsibilities for Seabrook Station, including performing routine inspections at the operating facility and reviewing applications for license renewal and an ASR-related license amendment. These responsibilities are carried out through processes separate from the review of PRM-50-109.

The petitioner raises the following three specific issues in PRM-50-109.

Issue 1: Visual inspections are not adequate to detect and confirm ASR.

The petitioner asserts that visual inspections are not capable of adequately identifying and confirming ASR or providing accurate information on the state of ASR damage (i.e., its effect on structural capacity). The petitioner also asserts that only petrographic examinations (the use of microscopes to examine samples of rock or concrete to determine their mineralogical and chemical characteristics) in accordance with ASTM C856-11 are capable of determining or confirming whether ASR is present and determining the state of ASR damage. The petitioner offers additional information in five areas related to this issue.

A. At an NRC public meeting at Seabrook Station on June 24, 2014, when C-10

asked if the NRC was investigating U.S. nuclear power plants for ASR concrete degradation, the NRC staff responded that ASR concrete degradation could be adequately identified through visual examination.

B. When structural degradation is occurring, the petitioner asserts that it is critical to determine the root cause and confirm the form of degradation. The NRC has repeatedly stated that ASR is confirmed only through petrographic examination, and that this is confirmed in a letter from the licensee for Seabrook Station, NextEra Energy Seabrook, LLC (NextEra) to the NRC, May 1, 2013 (ADAMS Accession No. ML13151A328).

C. Commentaries by materials science expert Dr. Paul Brown, provided by C-10 and UCS, challenge the central hypothesis in the report submitted by NextEra, “Seabrook Station: Impact of Alkali-Silica Reaction on Concrete Structures and Attachments” (ADAMS Accession No. ML12151A397). As summarized in the petition, Dr. Brown challenges the conclusion in the report that “confinement reduces cracking, and taking a core bore test would no longer represent the context of the structure once removed from the structure.”

D. The petitioner also asserts that the NRC memorandum titled, “Position Paper: In Situ Monitoring of Alkali-Silica Reaction (ASR) Affected Concrete: A Study on Crack Indexing and Damage Rating Index to Assess the Severity of ASR and to Monitor ASR Progression” (ADAMS Accession No. ML13108A047), supports the assertion that visual examination is insufficient to reliably identify ASR or evaluate its state (including contribution to rebar stress). The petitioner cites portions of text from the paper, which states that ASR can exist without indications of pattern cracking, visible surface cracking may be suppressed by heavy reinforcement while internal damage exists through the depth of the section, and crack mapping alone to determine ASR effects on the structure does not allow for the consideration of rebar stresses.

E. Finally, the petitioner asserts that visual inspections are of limited scope and cannot identify areas of degradation in many portions of concrete structures, such as below-grade portions that cannot be visually examined but are most likely to be exposed to groundwater and be more vulnerable to ASR. Cracking in the concrete wall of the shield building of the Davis-Besse Nuclear Power Station, discovered in 2011, when a hole was cut through the building's wall to replace the reactor vessel head, remained undetected by visual inspections for a long period.

Issue 2: ACI and ASTM codes and standards address the detection and evaluation of ASR damage.

The petitioner asserts that ACI 349.3R provides an acceptable means of protecting against excessive ASR concrete degradation and is endorsed by the NRC in Information Notice (IN) 2011-20, "Concrete Degradation by Alkali-Silica Reaction" (ADAMS Accession No. ML112241029). Quantitative criteria in ACI 349.3R can be used to evaluate inspection results. The petitioner also states that ASTM C856-11 is an acceptable means of conducting petrographic examination. The staff notes that although ACI 349.3R provides useful, general guidance for the development and implementation of a monitoring plan for concrete structures, it is neither formally endorsed nor approved for use by the NRC. Instead, IN 2011-20 mentions ACI 349.3R as a resource where additional information may be found regarding visual inspections (ADAMS Accession No. ML112241029).

The petitioner also provided information specific to activities at Seabrook Station related to the implementation of ACI 349.3R and the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section XI, Subsection IWL. The petitioner states that ACI 349.3R requires the formation of a "composite team," consisting of qualified civil or structural engineers, concrete

inspectors, and technicians familiar with concrete degradation mechanisms and long-term performance issues, to effectively identify and evaluate concrete degradation, including degradation due to ASR.

The petitioner claims that NextEra did not have a composite team as specified in ACI 349.3R, and since it became the owner of Seabrook Station, NextEra has not had a trained and dedicated “responsible engineer” conducting the inspections to accurately record the results or take further action as required. The petitioner asserts that NextEra failed to test the concrete despite the extent of cracking visibly increasing, and that NextEra never had a code-certified “responsible engineer” doing the visual inspections of the Seabrook containment in accordance with ASME BPV Code, Section XI, Subsection IWL.

The petitioner’s claim related to the implementation of ACI 349.3R at Seabrook Station, including the formation of a composite team, is outside the scope of the NRC’s consideration of the generic rulemaking action in response to PRM-50-109; however, this apparent claim of licensee wrongdoing was considered by the NRC’s allegations staff in Region I. After discussions with the petitioner, it was confirmed that the petitioner cited the issues with NextEra as examples of its concerns with regulations and did not intend the issues to be considered as allegations. Furthermore, the NRC found no violation of this ASME BPV Code requirement in its inspections, as discussed in Section III, “Reasons for Denial,” of this document.

Issue 3: Regulations should require compliance with ACI 349.3R and ASTM C856-11.

The petitioner states that, although both ACI 349.3R and ASTM C856-11 are endorsed by the NRC as acceptable, the NRC does not require nuclear power plant licensees to implement either of these standards. However, although ACI 349.3R and

ASTM C856-11 provide useful general guidance for the development and implementation of a monitoring plan for concrete structures, the NRC has neither formally endorsed nor approved their use. IN 2011-20 mentions ACI 349.3R as a resource where additional information may be found regarding visual inspections (ADAMS Accession No. ML112241029).

To support its position that use of the standards should be required, the petitioner asserts that Seabrook Station's ASR concrete degradation would have been identified before it caused moderate to severe degradation in seismic Category I structures if the NRC had required compliance, instead of merely encouraging compliance, with these existing standards. The petitioner claims that when NextEra determined 131 locations with "assumed" ASR visual signs within multiple power-block structures during 2012, further engineering evaluations were not done. The petitioner also claims that, since discovering the situation, the NRC has not required Seabrook Station to: 1) test a core bore taken from the containment; 2) use certified laboratory testing of key material properties to determine the extent of condition; or 3) obtain the data necessary to monitor the rate of progression.

The petitioner's claims related to this subject are outside the scope of the NRC's consideration of the generic rulemaking action in response to PRM-50-109; however, these apparent claims of NRC wrongdoing were forwarded to the NRC's Office of the Inspector General and subsequently to the NRC's allegations staff in Region I. After discussions with the petitioner, the NRC confirmed that the petitioner cited the issues as examples of their concerns with the regulations and did not intend them to be considered as allegations or claims of wrongdoing. Furthermore, as noted in Section III of this document, NextEra commented in response to PRM-50-109 that all 131 locations were included in the structural evaluation.

II. Public Comments on the Petition

The NRC published a notice of docketing of PRM-50-109 on January 12, 2015 (80 FR 1476). The public comment period closed on March 30, 2015. Comment submissions on this petition are available electronically via <https://www.regulations.gov> using docket number NRC-2014-0257.

Overview of Public Comments

The NRC received 10 different comment submissions on the PRM. A *comment submission* means a communication or document submitted to the NRC by an individual or entity, with one or more individual *comments* addressing a subject or issue. Eight of the comment submissions were received during the comment period and two were filed late. The NRC determined that it was practical to consider comment submissions received after the end of the public comment period and therefore considered all 10 comment submissions. Key information for each comment submission is provided in the following table.

| Submission # | ADAMS Accession Number | Commenter | Affiliation |
|--------------|------------------------|-------------------|--|
| 1 | ML15026A339 | Josephine Donovan | Private Citizen |
| 2 | ML15026A338 | Lynne Mason | Private Citizen |
| 3 | ML15027A178 | Katherine Mendez | Private Citizen |
| 4 | ML15076A457 | David Lochbaum | Union of Concerned Scientists |
| 5 | ML15076A459 | Garry Morgan | Blue Ridge Environmental Defense League – Bellefonte Efficiency and Sustainability Team / Mothers Against Tennessee River Radiation (BREDL/BEST/MATRR) |
| 6 | ML15076A460 | G. Dudley Shepard | Private Citizen |
| 7 | ML15085A523 | Jason Remer | Nuclear Energy Institute (NEI) |

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|-----|-------------|---------------------|----------------|
| 8 | ML15089A284 | James M. Petro, Jr. | NextEra Energy |
| 9* | ML15097A337 | Anonymous | Anonymous |
| 10* | ML15112A265 | Scott Bauer | STARS Alliance |

*Comments submitted after March 30, 2015.

Seven commenters expressed support for the PRM and proposed identification techniques, while the three remaining commenters (numbers 7, 8, and 10) opposed the PRM in part or in whole. Based on similarity of content, the public comments were binned into six bins. The NRC reviewed and considered the comments in its decision to deny the PRM. Summaries of each bin and the NRC's responses are provided in the following discussion in an order that provides appropriate context for the response to each of the comment bins.

NRC Responses to Comments on PRM-50-109

Comment Bin 1: Existing inspection techniques will not adequately detect concrete degradation due to ASR, and C-10's proposed solutions (i.e., requiring compliance with ACI 349.3R and ASTM C856-11 via regulation) are appropriate to adequately detect ASR degradation. (Submission 4, Submission 5, Submission 6)

NRC Response: The NRC disagrees with the comments that visual inspections do not adequately identify ASR and that ACI 349.3R and ASTM C856-11 should be regulatory requirements. The current ASR literature and case history, as described in Section III and referenced in Section V, "Availability of Documents," of this document, provides no evidence that ASR would degrade the safety function of a structure or component before it expands to a degree that would cause visible symptoms (e.g., cracking). Existing regulations, such as those listed in the response to Comment Bin 4, require inspection methods that can detect applicable degradation mechanisms (including ASR), and require that significant degradation (regardless of cause) be appropriately addressed through additional plant-specific inspections or structural evaluations. Furthermore, the

documents (ACI 349.3R and ASTM C856-11) that are being proposed for inclusion in the regulations do not provide specific guidance for identifying ASR degradation in structures and, therefore, requiring their use via regulation would not provide improved techniques for identifying ASR degradation. Additional details on the NRC's position can be found in Section III, "Reasons for Denial," of this document.

Comment Bin 2: The NRC should grant the C-10 petition for rulemaking because visual inspection of ASR concrete degradation is insufficient. (Submission 1, Submission 2)

NRC Response: The NRC disagrees. As noted in the response to Comment Bin 1, the staff finds visual inspection sufficient to detect ASR concrete degradation before the safety function of a structure or component would be degraded. The commenters did not provide a basis for their position that visual inspection of concrete degradation is insufficient to identify ASR that would lead to unacceptable changes in concrete structural properties.

Comment Bin 3: The NRC should investigate the concrete cracks at Seabrook Station because the concrete degradation poses serious safety concerns. (Submission 3)

NRC Response: The NRC views this comment as a request for regulatory action outside the scope of PRM-50-109. As discussed in Section 1 of this document, the NRC has referred this comment to its Region I allegations staff, and has advised the commenter of this request. Further details are discussed in Section I of this document. The NRC continues to use its ongoing oversight and licensing processes to ensure the safe operation of Seabrook Station.

Comment Bin 4: The nuclear industry does not believe that rulemaking is necessary to resolve issues related to inspecting concrete for ASR degradation. Following the

issuance of NRC IN 2011-20, licensees took appropriate actions by: a) recording the issue in the Institute for Nuclear Power Operations (INPO) Operating Experience system; and b) updating their Structures Monitoring Program, improving procedures, and informing responsible individuals concerning examination for conditions that could potentially indicate the presence of ASR. In addition, there already exist ample regulatory requirements to ensure appropriate attention is given to potentially degraded concrete, including due to ASR. (Submission 7, Submission 10)

NRC Response: The NRC agrees with the comment. By issuing IN 2011-20, the NRC has made the U.S. nuclear power industry aware of the operating experience related to ASR concrete degradation at Seabrook Station. Licensees are expected to evaluate INs in their operating experience programs and incorporate, as appropriate and applicable, the information into their monitoring programs and procedures. For example, NextEra has conducted prompt operability evaluations for Seabrook Station and is taking ongoing actions in its corrective action program for long-term resolution of the issue. The NRC documented the licensee's commitments in a Confirmatory Action Letter (CAL; ADAMS Accession No. ML12125A172) and has overseen the completion of these actions (see, for example, inspection report (IR) 05000443/2012010 (ADAMS Accession No. ML13221A172)). In addition, multiple license renewal applications (LRA) submitted after the issuance of IN 2011-20 have included information that demonstrates their monitoring programs have been updated to inspect for ASR degradation, regardless of the aggregate reactivity test results from construction (see, for example, Section 3.5.2.2.2.1.2 of LaSalle County Station LRA (ADAMS Accession No. ML14343A849), Waterford Steam Electric Station LRA (ADAMS Accession No. ML16088A324), or River Bend Station LRA (ADAMS Accession No. ML17153A282)).

Existing regulations such as § 50.55a, "Codes and Standards"; § 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants";

10 CFR part 50, appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”; 10 CFR part 50, appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors”; and 10 CFR part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” require licensees to monitor the performance or condition of structures and take corrective action to address degraded or nonconforming conditions in a manner commensurate with the safety significance of the structures. Compliance with these regulations provides reasonable assurance that affected structures remain capable of performing their intended functions. Further, the NRC confirms the acceptability of licensees’ approaches through processes such as the reactor oversight process, license renewal, and generic communications (e.g., bulletins, generic letters, and INs to address significant industry events, operating experience, and degradation-specific issues that may have generic applicability). The existing regulatory requirements and processes provide reasonable assurance of adequate protection of public health and safety; therefore, it is not necessary to amend the regulations on a degradation-specific basis to require better protection against the degradation of concrete structures.

The technical comments and clarifications made by the commenters related to ACI 349.3R and the role of visual inspections are addressed in Section III of this document.

Comment Bin 5: New rulemaking is not necessary to resolve issues related to inspecting concrete for ASR. The ACI 349.3R and ASTM C856-11 have been used for investigation of ASR conditions at Seabrook Station; however, neither standard provides inspectors with new or improved means to identify, monitor, or assess ASR-impacted structures, as implied by the petition. The commenter questions the basis of the petition, including misconceptions and factual errors made in the petition concerning NextEra

activities at Seabrook Station. (Submission 8)

NRC Response: The NRC agrees with the comment that new rulemaking is not needed. The guidance in ACI 349.3R is primarily based on visual inspection; addresses only commonly occurring degradation conditions in nuclear structures; and provides very limited guidance with regard to ASR identification, monitoring, and evaluation.

Therefore, it is not considered an authoritative document for ASR. ASTM C856-11 is a consensus standard that provides an established method for conducting petrography that can be used to confirm the diagnosis of ASR. Neither ACI 349.3R nor ASTM C856-11, however, provides a method for monitoring progression, or evaluating and quantifying observed ASR effects on structural capacity or performance. These documents have been in existence since 1996 (for ACI 349.3R) and 1977 (for ASTM C856-11) and do not provide any new or improved methods beyond what is already standard practice in the concrete industry.

The portions of the comment concerning NextEra activities at Seabrook Station and possible factual errors in the petition are addressed in Section III of this document.

Comment Bin 6: Current ASME testing protocols should be followed. Ultrasonic testing should be conducted for reactor pressure vessels to test for defects and radiation filters should be installed on pressure vessels as a post-Fukushima precaution. (Submission 9)

NRC Response: This comment is not related to ASR degradation and is outside the scope of PRM-50-109.

III. Reasons for Denial

The NRC has determined that rulemaking, as requested in the petition, is not needed for reasonable assurance of adequate protection at nuclear power plants with respect to ASR. The NRC's evaluation of the three issues raised in PRM-50-109 (summarized by the NRC in Section I, "The Petition," of this document and constituting the petition's basis for the requested rulemaking) are set forth below.

Issue 1: Visual Inspections are not adequate to detect and confirm ASR.

The NRC agrees with the petitioner that visual inspections are not enough to positively confirm ASR. However, given the slow progression of ASR, visual inspections are sufficient to identify manifestations of potentially damaging ASR before there would be significant structural impacts. Such results would be sufficient to inform whether further actions should be taken. Therefore, the NRC's position is that visual examination is acceptable for routinely monitoring concrete structures to identify areas of potential structural distress or degradation, including degradation due to ASR. This position is supported by the current ASR literature and case history, as referenced in Section V of this document. The occurrence of ASR expansion results in one or more common visual indications (e.g., expansion causing deformation, movement, or displacement; cracking; surface staining; gel exudations; pop-outs) prior to causing significant structural degradation (as shown in FHWA-HIF-09-004 and Canadian Standards Association (CSA) A864-00, referenced in Section V of this document). However, the presence of one or more of these visual symptoms is not necessarily an indication that ASR is the main factor responsible for the observed symptoms. If suspected, the presence or absence of ASR should be confirmed by an acceptable method (e.g., petrographic examination).

Based on this information, the NRC maintains that visual examination is an acceptable method for detecting indications of ASR degradation. Once ASR is

suspected based on visual indications, the licensee would conduct additional inspections, testing (non-destructive or invasive), petrographic analysis, and structural evaluations, as appropriate to the specific case, to evaluate the effects of ASR on structural performance under design loads. This general approach is similar to and consistent with the approach recommended in literature related to ASR (e.g., FHWA-HIF-09-004 and guidance by the Institution of Structural Engineers, referenced in Section V of this document).

The NRC evaluated the five areas in which the petitioner provided additional information related to this issue.

A. The NRC notes that the petition does not provide the context of the statements made by the NRC staff regarding the acceptable use of visual examination during the June 24, 2014, public meeting. The NRC staff stated that it finds the use of visual examination acceptable for routine periodic monitoring, in implementing a structures monitoring program under the maintenance rule and the containment inservice inspection program pursuant to § 50.55a, and in identifying the general condition of concrete structures and areas that are suspected to have deterioration or distress due to any degradation mechanism, including ASR. If the licensee identifies visual indications of ASR, the licensee's next step would be to confirm ASR by petrographic examination or other methods, and conduct further assessments, as necessary, to determine the impact on the structure's intended functions and the need for corrective actions. While visual inspections alone would not confirm the presence or absence of ASR, a petrographic examination of concrete is not necessary prior to manifestation of visual symptoms of ASR, given the minimal impact ASR has on structural performance of reinforced concrete structures at this stage, as discussed above. The NRC maintains its position that visual examination is an acceptable and

adequate approach to assess the concrete's general condition and identify areas of potential structural distress or deterioration, including areas where ASR is suspected.

B. Specific to the petitioner's statement related to the need to determine the root cause of degradation, Criterion XVI, "Corrective Action," of 10 CFR part 50, appendix B requires that conditions adverse to quality and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Therefore, existing NRC regulations require the identification of the root cause of significant conditions adverse to quality. The NRC agrees that, while other techniques may emerge, petrographic examination of the concrete sample under a microscope is a well-established technique to confirm the presence or absence of ASR at any stage.

Once ASR is confirmed at a site by petrographic examination (conducted after manifestation of characteristic visual symptoms), it is conservative to assume that other structures exhibiting visible symptoms are also affected, based on similarity of materials and environmental exposure conditions. Therefore, it is not necessary to take cores from all potentially affected structures for confirmatory petrographic examination. Once identified visually, ASR can be verified via petrography, or it can be conservatively assumed based on confirmation in similar areas, and the degradation can be addressed accordingly.

The NRC considered the experience at Seabrook Station as an example in evaluating this generic request for regulatory action. The licensee for Seabrook Station classified ASR in safety-related concrete structures as a significant condition adverse to quality. The licensee first identified visual symptoms of ASR, and then confirmed the presence of ASR by conducting petrographic examinations of a sample of cores taken from safety-related structures that exhibited the worst visual ASR symptoms. The

licensee then conservatively assumed that other structures with visual symptoms are also affected by ASR, given similar concrete mix and exposure conditions. Based on this assumption, the licensee conducted a root cause analysis of the degradation, made prompt operability determinations of affected structures, and continued monitoring to demonstrate a reasonable expectation that the affected structures were, and remain, operable and capable of performing their intended safety functions. The licensee also initiated ongoing research and testing to develop technical bases for a long-term resolution of the issue at the site, while continuing to monitor the degradation.

The NRC concluded that root cause analyses and appropriate follow-up actions are already required under 10 CFR part 50, appendix B and have been implemented for affected structures at Seabrook Station. Therefore, the NRC staff has concluded that a revision to the NRC's regulations is not necessary based upon Seabrook Station as a generic example.

C. The NRC has previously responded to the statements referenced by the petitioner from Dr. Paul Brown, which were included in a letter from UCS to the NRC dated November 4, 2013 (ADAMS Accession No. ML13309B606). The NRC responded to the UCS letter on December 6, 2013 (ADAMS Accession No. ML13340A405). In the response, the NRC noted that information from drilled cores may be valuable for assessing the impact of ASR on concrete; however, the use of test data from cores alone may not be an appropriate, realistic indicator of overall structural performance.

Additionally, the NRC notes that ASR literature and case history indicate that ASR has a much more detrimental effect on the mechanical properties of concrete cores and cylinders than on the structural behavior of reinforced concrete structural components and systems (as described in TXDOT Technical Report No. 12-8XXIA006 and the *ACI Structural Journal* article referenced in Section V of this *document*). These documents indicate that the empirical relationships in the ACI codes between concrete-

cylinder compressive strength and other mechanical properties, including structural capacity, may not necessarily remain valid for ASR-affected structures. Reinforced concrete structures and components respond to load as part of a composite structural system in which there are external restraints, internal confinement, and interaction between the steel reinforcement and the concrete. Therefore, an evaluation of the impact of ASR on structural performance of affected reinforced concrete structural components and systems should consider the structural context to obtain a realistic assessment of the impact on structural capacity. The use of core test data in the traditional manner, alone, may not be appropriate or realistic to assess structural performance of ASR-affected structures.

D. Regarding the petitioner's reference to the NRC position paper (ADAMS Accession No. ML13108A047), the NRC's current position on the role of visual inspections in identifying ASR is set forth above. The referenced position paper is not an official NRC position on the topic, but rather was prepared by an individual staff member to facilitate internal technical discourse and inform staff review of an issue. The referenced position paper does not state that visual examination is insufficient to identify indications of ASR; however, it does note that surface cracking or crack mapping, alone, may not indicate the severity of ASR degradation and is not adequate to determine structural effects of ASR. The NRC agrees that surface crack mapping alone is not adequate to monitor ASR progression and address its structural effects. In addition, petrographic examination provides very limited information to evaluate the structural effects of ASR.

Addressing visual indications of a potential concrete-degradation issue does not end with the visual inspection. If indications of distress or deterioration are visually identified, under existing NRC regulations, licensees are required to address the effects of the observed degradation and demonstrate that the structure remains capable of

performing its safety functions. Depending on the observed conditions, this is accomplished through additional inspections, testing, and structural evaluations.

Specific to the example of Seabrook Station referenced by the petitioner, the licensee has submitted a license amendment request (LAR) proposing a method of evaluation and supporting technical bases to address the impact of ASR on structural performance of affected structures, and to ensure appropriate monitoring programs are in place to adequately monitor its progression, such that intended functions are maintained. This LAR is currently under review by the NRC staff. The NRC's ongoing oversight and licensing processes will ensure that the licensee takes appropriate actions for long-term resolution of the ASR issue at Seabrook Station. Consideration of these site-specific aspects is outside the scope of the generic requests made in PRM-50-109.

E. Specific to the petitioner's comment on the limited scope of visual inspections, the NRC agrees that visual inspections cannot directly identify degradation in inaccessible portions of concrete structures. However, many below-grade structures in nuclear power plants are accessible for visual inspection on the interior face of the concrete. For example, visual symptoms of ASR were first discovered on the interior surfaces of exterior below-grade walls at Seabrook Station. Furthermore, ASR degradation or expansion in inaccessible areas would manifest visually in accessible areas, in the form of cracking, displacements, or deformations, before causing a significant structural impact. As noted previously, current ASR literature and case history show that visual inspections are sufficient to identify manifestations of potentially damaging ASR before there would be significant structural impacts. For concrete containment structures, existing regulations in § 50.55a(b)(2)(viii) require evaluation of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or could result in, degradation to such inaccessible areas. Therefore, existing regulations, regulatory guidance, and licensee programs have

provisions to adequately address degradation in inaccessible areas.

The issue of laminar cracking in the shield building at Davis-Besse, referenced by the petitioner, was a unique situation resulting from a combination of extreme environmental conditions and the design configuration of the shield building. The licensee evaluated the issue, including operability determinations and root cause analysis in its corrective action program; and the NRC's continued oversight of the issue has been documented in a series of NRC inspection reports, the latest of which is IR 05000346/2014008, dated May 28, 2015 (ADAMS Accession No. ML15148A489). This issue has no connection to ASR degradation or ASR detection.

Issue 2: Codes and standards exist for detecting and evaluating ASR damage.

The NRC disagrees that there are consensus codes or standards sufficient to provide guidance for detecting and evaluating ASR damage. The scopes of ACI 349.3R and ASTM C856-11 are discussed separately below.

A. The ACI 349.3R is an ACI committee technical report intended to provide recommended guidance for developing and implementing a procedure for inspection and evaluation of many common concrete degradation mechanisms in nuclear concrete structures. It contains only very limited general information regarding ASR. ASR is not a common condition in nuclear power plants, and the quantitative evaluation criteria provided in the document have little or no specific applicability to ASR degradation; therefore, ACI 349.3R is not an authoritative document to address and evaluate the impact of ASR on intended functions of affected structures.

The discussion of evaluation techniques in ACI 349.3R recommends visual inspection as the initial technique used for any evaluation, and states that visual inspection can provide significant quantitative and qualitative data regarding structural performance and the extent of any degradation. The recommended approach places

emphasis on the use of general condition survey practices (visual inspection) in the evaluation, supplemented by additional testing or analysis as needed, based on the results of the general survey. Chapter 5, "Evaluation Criteria," of ACI 349.3R states: "these guidelines focus on common conditions that have a higher probability of occurrence and are not meant to be all-inclusive. These criteria primarily address the classification and treatment of visual inspection findings because this technique will have the greatest usage."

Although ACI 349.3R provides useful general guidance for the development and implementation of a monitoring plan for concrete structures, it is neither formally endorsed nor approved for use by the NRC. Instead, IN 2011-20 mentions ACI 349.3R as a resource where additional information may be found regarding visual inspections (ADAMS Accession No. ML112241029). Since ASR degradation would need to be addressed on a degradation-specific and plant-specific basis, requiring the use of ACI 349.3R would not provide better protection against ASR concrete degradation than the current NRC requirements.

Related to the petitioner's comments on "composite teams," the NRC agrees that qualified personnel should be used to conduct activities related to safety-related functions of structures, systems, and components (SSCs); this is covered in existing regulations by the quality assurance program requirements in 10 CFR part 50, appendix B. This appendix requires applicants and licensees to establish and implement a quality assurance program that applies to all activities affecting the safety-related functions of SSCs. This program ensures that the activities are controlled and correctly performed to provide adequate confidence that SSCs will perform satisfactorily in service, including appropriate qualification and training of personnel performing activities affecting quality to assure suitable proficiency. This adequate confidence is a basis for concluding that reasonable assurance of adequate protection is provided. The ASME BPV Code,

Section XI, Subsection IWL, defines specific qualifications and responsibilities of the “responsible engineer,” who evaluates the examination results and the condition of the structural concrete related to the containment. Section 50.55a(g)(4) requires compliance with the ASME BPV Code, Section XI. In addition to § 50.55a requirements for containments, safety-related structures are monitored under § 50.65 (the maintenance rule), and the associated qualification requirements are typically provided in the licensee’s implementing procedures, based on their 10 CFR part 50, appendix B program.

As noted above, the petitioner’s claims on the subject of personnel qualification are outside the scope of the NRC’s consideration of the generic rulemaking action in response to PRM-50-109. The NRC understands the above assertions are based on excerpts from the NRC License Renewal Inspection Report 05000443/2011007 (ADAMS Accession No. ML111360432). While this report noted that the licensee’s aging management program procedures should include a more explicit definition of “responsible engineer,” acceptance criteria, and the qualification requirements of the inspectors, the NRC found no violation of the requirement to have a qualified “responsible engineer” to direct inspections, as required by the ASME BPV Code and § 50.55a. In a letter to the NRC dated December 17, 2010 (ADAMS Accession No. ML103540534), the licensee noted that the acceptance criteria had been updated, and committed to also update the procedures to include the definition of “responsible engineer,” prior to the period of extended operation. The licensee also commented on the petition assertions in Submission 8 (ADAMS Accession No. ML15089A284), and stated that each of the IWL inspections had a certified responsible engineer, as required by ASME Section XI.

B. The NRC agrees that ASTM C856-11 is a consensus standard that details how to conduct petrographic analysis of concrete bores, and provides an acceptable

method to positively confirm the diagnosis of ASR. However, it does not provide any guidance on when cores should be taken, from where cores should be taken, how many cores should be taken, or how frequently cores should be taken. Also, it does not provide a method to evaluate ASR damage for impact on structural performance.

ASTM C856-11 outlines procedures for the petrographic examination of samples of hardened concrete for a variety of purposes. One of the purposes of this consensus standard is identifying visual evidence to establish whether ASR has taken place, what aggregate constituents were affected, and what evidence of the reaction exists. Petrographic examination provides an assessment of the extent of ASR gel development and its intrusion into the pores of the concrete sample; however, petrographic examination does not indicate the impact of the ASR reaction on the structural performance under design loads. Furthermore, ASTM C856-11 does not provide any guidance on monitoring or evaluating a concrete structure, such as when to take cores, or which portion of a structure should be evaluated via core bores.

Materials laboratories that perform petrographic examination of hardened concrete samples typically follow the current ASTM C856 standard practice for the application; unless another specific procedure is specified in the request. The standard to which a plant-specific petrographic examination is performed is specified by the licensee and not addressed in the regulations; however, 10 CFR part 50, appendix B requires licensees to ensure that activities affecting safety-related functions are correctly performed. Also, 10 CFR part 50, appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion 1, "Quality standards and records," requires, in part, that "where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function." Therefore, the licensee must ensure the analysis will

adequately identify ASR.

In summary, both ACI 349.3R and ASTM C856-11 provide useful guidance and methods licensees may adopt, as applicable, to meet requirements in existing NRC regulations, such as § 50.55a, § 50.65, and 10 CFR part 54; however, neither of the documents provide methods to comprehensively address the long-term structural impact and management of ASR degradation.

Issue 3: Regulations should require compliance with ACI 349.3R and ASTM C856-11.

The NRC disagrees that its regulations should be revised to require compliance with ACI 349.3R and ASTM C856-11. As discussed previously, the NRC's existing regulations are sufficient to provide reasonable assurance of adequate protection of public health and safety due to concrete degradation, including ASR.

The petition does not take into account the NRC's existing regulatory requirements that each nuclear power reactor licensee must meet to demonstrate the ongoing capability of structures to perform their intended safety functions. The NRC's regulatory requirements are programmatic and generic in nature, applicable to all operating reactors, and focused on overall structure and component performance requirements to maintain intended safety functions. The NRC's regulations do not generally prescribe how licensees must meet the requirements, nor do the regulations normally address degradation-specific issues. The following discussion identifies and briefly summarizes the relevant NRC regulatory requirements and processes and explains how they require licensees to address ASR before it becomes a safety issue.

- Section 50.65 requires licensees to monitor the performance or condition of SSCs under its scope, considering industry-wide operating experience, in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their

intended functions. For structures, this requirement is normally met by periodically monitoring the condition of structures on a frequency that is commensurate with the safety significance of the structure and its condition. If the basic assessments identify degradation, additional degradation-specific condition monitoring is required, along with more frequent assessments until the degradation is addressed. Regulatory Guide (RG) 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," provides guidance on methods acceptable to the NRC staff for implementation of the maintenance rule and includes the attributes of an acceptable structural monitoring program. The NRC will consider taking a formal regulatory position on the use of ACI 349.3R guidelines in the next revision of RG 1.160; however, the existing regulation already requires structural assessments that are adequate to detect visual indications of ASR before it would pose a significant structural concern. Accordingly, § 50.65 is a regulatory requirement that forms part of the NRC's regulatory infrastructure for the identification and further technical evaluation of ASR, before there would be significant degradation of structural integrity of safety-related concrete structures at nuclear power plants.

- Criterion XVI, "Corrective Action," of 10 CFR part 50, appendix B requires licensees to implement a corrective action program to assure that conditions adverse to quality and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined, and corrective action is taken to preclude repetition. The 10 CFR part 50, appendix B requirement, in this regard, applies to all degradation mechanisms, including ASR. In the case of ASR, a licensee would have to identify the root cause of the degradation and address the degradation, such that intended safety functions are not impacted. Accordingly, Criterion XVI is an NRC regulatory requirement that is part of the NRC's regulatory infrastructure for the identification and further

technical evaluation of ASR, before there would be significant degradation of structural integrity of safety-related concrete structures at nuclear power plants.

- Section 50.55a(g)(4) requires licensees to inspect concrete containments in accordance with the ASME BPV Code, Section XI, Subsection IWL, as incorporated by reference and subject to conditions. Subsection IWL requires that a general visual examination of all accessible containment concrete surfaces be conducted every 5 years by qualified personnel under the direction of the “responsible engineer.” Further, Subsection IWL requires a detailed visual examination to determine the magnitude and extent of deterioration and distress of suspect containment concrete surfaces initially detected by general visual examinations. Subsection IWL specifies acceptance standards based on acceptance by examination, acceptance by engineering evaluation (requires preparation of an engineering evaluation report including cause of the condition), or acceptance by repair/replacement. In accordance with the condition on use of Section XI in § 50.55a(b)(2)(viii)(E), licensees must evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. These requirements ensure that visual indications of ASR will be detected prior to causing significant structural degradation that could impact the intended safety function of the containment. Accordingly, § 50.55a is a regulatory requirement that is part of the NRC’s regulatory infrastructure for the identification and further technical evaluation of ASR, before there would be significant degradation of structural integrity of concrete containment structures at nuclear power plants.

- Appendix J to 10 CFR part 50, “Primary Reactor Containment Leakage Testing Requirements for Water Cooled Reactors,” requires that primary reactor containments periodically meet the leakage-rate test requirements to ensure that a) leakage does not exceed allowable leakage rates specified in the technical

specifications; and b) integrity of the containment structure is maintained during its service life. This regulation requires periodic performance monitoring of the containment to demonstrate that the containment can perform its intended safety function, regardless of identified degradation. If the containment were unable to meet the requirements of 10 CFR part 50, appendix J, it would be declared inoperable and the plant could not return to operation until the issue was addressed. Accordingly, appendix J of 10 CFR part 50 is a regulatory requirement that is part of the NRC's regulatory infrastructure for the identification and technical evaluation of ASR, before there would be significant degradation of structural integrity of concrete containment structures at nuclear power plants.

- Section 54.21(a)(3) requires applicants for license renewal to demonstrate that the effects of aging will be adequately managed, such that the intended functions of structures and components subject to aging management are maintained, consistent with the current licensing basis for the period of extended operation. Regulatory guidance for developing aging management programs, including for ASR aging effects on concrete structures, is provided in NUREG-1801, "Generic Aging Lessons Learned Report" (GALL Report). Any licensee applying for license renewal must have a structural aging management program in place that can identify indications of concrete degradation, including degradation due to ASR, before it becomes an issue that could impact an intended safety function. Accordingly, § 54.21(a)(3) is a regulatory requirement that is part of the NRC's regulatory infrastructure for the identification and further technical evaluation of ASR, before there is significant degradation to structural integrity of safety-related concrete structures at nuclear power plants.

- The Reactor Oversight Process (ROP) is the process that the NRC uses to verify that power reactors are operating in accordance with NRC rules and

regulations. Following the ROP, the NRC conducts routine baseline inspections, problem identification and resolution inspections, reactive inspections, and other assessments of plant performance. If licensees are not properly meeting the regulations, the NRC can take actions to protect public health and safety. Inspection Manual Chapter (IMC) 0326, "Operability Determinations and Functionality Assessments for Conditions Adverse to Quality or Safety" (ADAMS Accession No. ML13274A578), provides inspectors with guidance to assist their review, under the ROP, of operability and resolution of degraded or nonconforming conditions, such as degradation due to ASR. Section 07.04, "Final Corrective Action," of IMC 0326 states that a licensee's range of corrective action may involve: 1) full restoration to the Updated Final Safety Analysis Report (UFSAR) described condition; 2) a change to the current licensing basis (CLB) to accept the as-found condition as is; or 3) some modification of the facility or CLB other than restoration to the condition as described in the UFSAR. In items 2 and 3, the final corrective action requires a review in accordance with § 50.59, "Changes, tests, and experiments," to determine if it requires NRC approval via a license amendment pursuant to § 50.90, "Application for amendment of license, construction permit, or early site permit."

- The generic communications process is used to address potential generic issues that are safety significant and may necessitate action by licensees to resolve. Generic communications are used to communicate safety significant issues and operating experience, including degradation-specific issues, include bulletins, generic letters, and INs. The NRC has issued a generic communication (IN 2011-20) to inform the industry of the generic impacts of ASR. Information about the NRC's Generic Communications Program is available at <http://www.nrc.gov/about-nrc/regulatory/gencomms.html>.

- The enforcement process may be used if licensees fail to adequately

address safety-significant issues, consistent with the regulatory requirements as outlined above. The NRC may use enforcement actions, including issuing orders pursuant to § 2.202, "Orders," to modify, suspend, or revoke a license if ASR becomes a safety-significant issue that a licensee is not adequately addressing.

In addition to these generic requirements and processes, the GALL Report (NUREG-1801) makes specific reference to ACI 349.3R in its guidance for aging management programs (AMPs). AMP XI.S6, "Structures Monitoring," recommends that visual inspection be used to identify structural distress or deterioration of concrete, such as that described in ACI 201.1R and ACI 349.3R. In addition, the GALL Report notes that the personnel qualifications in Chapter 7 and the evaluation criteria in Chapter 5 of ACI 349.3R are acceptable for concrete structures. However, the GALL Report also notes that use of plant-specific criteria may also be justified. Although ACI 349.3R is one acceptable method to monitor concrete structures for degradation, it is not the only method, and so there is no basis for the NRC to require its exclusive use via regulation.

With respect to ASTM C856-11, the NRC agrees that it is an acceptable and established consensus testing standard for conducting petrographic examination of hardened concrete that can be used to confirm the diagnosis of ASR. However, as discussed previously, the NRC's existing regulations in 10 CFR part 50, appendix A and appendix B, ensure appropriate methods or standards are used when conducting tests associated with safety-related structures. Therefore, there is no need to require the use of ASTM C856-11 through regulation.

Specific to the petitioner's claims regarding Seabrook Station, the licensee for Seabrook Station addressed, in its ASR root cause analysis, the performance and organizational factors that contributed to inadequacies in its Structures Monitoring Program and the failure of licensee staff to have identified ASR degradation sooner. The licensee has taken corrective action to preclude repetition (see licensee response to

the NRC's CAL, ADAMS Accession No. ML13151A328). The licensee also determined that there would have been no significant increase in safety if the licensee had identified the degradation earlier. In Comment Submission 8 (ADAMS Accession No. ML15089A284), the licensee directly responded to the petitioner's assertion regarding structural evaluation and stated that all of the 131 locations with visual indications of ASR were included in the structural evaluation.

Since ASR has been identified at Seabrook, the NRC has inspected the licensee's corrective actions multiple times, including two CAL follow-up inspections, which are summarized in IR 05000443/2012009 and IR 05000443/2012010 (ADAMS Accession Nos. ML12338A283 and ML13221A172, respectively). The reports detail the actions taken by the licensee after the discovery of ASR, including the interim structural assessment, and note that the NRC verified that the licensee had evaluated all locations with visual indications of ASR and determined that all ASR-affected structures, including containment, remain operable. The staff also noted that the licensee would not exclude any reinforced concrete structure from ASR monitoring until a satisfactory petrographic examination confirmed the absence of ASR; therefore, all structures, including the containment, would be assumed to experience ASR until petrographic examination determined otherwise. Although cores have not been taken from the containment, the licensee has assumed that local portions of the containment with visual indications are affected by ASR, and evaluated the structures, accordingly.

The NRC's ongoing oversight and licensing processes will ensure that the licensee takes appropriate actions for long-term resolution of the ASR issue at Seabrook Station. Disposition of these site-specific aspects is outside the scope of the generic requests made in PRM-50-109.

The NRC also considered whether ASR concrete degradation raises new safety concerns that would justify additional regulatory requirements for all licensees beyond

those already included in NRC regulations, as described above. While it is possible there could be plants that used a potentially reactive aggregate in their concrete, the NRC is not aware of any U.S. nuclear power plants, other than Seabrook Station, that have a documented occurrence of ASR. The NRC notes that the use of a potentially reactive aggregate does not necessarily result in the occurrence of ASR. In addition to reactive aggregates, relatively high alkali content in the cement, and high relative humidity levels are necessary for ASR to occur. Through the issuance of IN 2011-20, the NRC has informed licensees of the occurrence of ASR-induced concrete degradation at Seabrook Station, with the expectation that the operating experience would be evaluated by licensees and considered for appropriate action. Thus, the nuclear power industry is aware of the potential for ASR to occur, even if aggregates were screened out based on reactivity tests or other tests conducted at the time of construction. For the reasons outlined above, the NRC has determined that a regulation is not needed to require industry-wide compliance with ACI 349.3R and ASTM C856-11.

IV. Conclusion

For the reasons cited in Section III of this document, the NRC is denying PRM-50-109 under § 2.803. Existing NRC regulations establish programmatic and design basis requirements that are adequate to address the effects of concrete degradation mechanisms, including ASR, in safety-related structures. Compliance with these regulations, verified through NRC licensing and oversight processes, provide reasonable assurance of adequate protection. Specifically, existing NRC regulations ensure that concrete degradation due to ASR will not result in unacceptable reductions in structural capacity of safety-related structures at nuclear power plants. Therefore, new or amended regulations to require the use of the documents identified in the PRM (ACI 349.3R and ASTM C856-11) to provide better protection against concrete

degradation due to ASR are not needed in order to provide reasonable assurance of adequate protection at U.S. nuclear power plants.

V. Availability of Documents

The documents identified in the following table are available to interested persons through one or more of the following methods, as indicated. For more information on accessing ADAMS, see the ADDRESSES section of this document.

| Document | ADAMS Accession Number / Federal Register Citation / Report Number and Date | Link to Publication |
|--|---|--|
| PRM Documents | | |
| PRM from the C-10 Research and Education Foundation | ADAMS Accession No. ML14281A124 September 25, 2014 | http://pbadupws.nrc.gov/docs/ML1428/ML14281A124.pdf |
| <i>Federal Register</i> notice for PRM, notice of docketing, and request for comment | Federal Register / Vol. 80, No. 7 / Monday, January 12, 2015 / Proposed Rules | http://www.gpo.gov/dsys/pkg/FR-2015-01-12/html/2015-00199.htm |
| SECY-18-XXXX, “Denial of Petition for Rulemaking Submitted by the C-10 Foundation (PRM-50-109) | ADAMS Accession No. ML15301A084 [date] | http://pbadupws.nrc.gov/docs/ML1530/ML15301A084.pdf |
| Public Comments on PRM (see table under the heading, I. Public Comments on the Petition) | | |
| ASR-Related Technical Materials | | |
| “Standard Practice for Petrographic Examination of Hardened Concrete” ASTM International | ASTM C856-11 2011 | Available for purchase: http://www.astm.org/Standards/C856.htm |
| “Standard Practice for Petrographic Examination of Hardened Concrete” ASTM International | ASTM C856-14 2014 | Available for purchase: http://www.astm.org/DATABASE.CART/HISTORICAL/C856-11.htm |
| “Evaluation of Existing Nuclear Safety Related Concrete Structures” American Concrete Institute | ACI 349.3R-02 June 2002 | Available for purchase: https://www.concrete.org/store/productdetail.aspx?ItemID=349302&Format=DOWNLOAD |
| “Effect of Alkali Silica Reaction Expansion and Cracking on Structural Behavior of Reinforced Concrete Beams” <i>ACI Structural Journal</i> | Title no. 95-S44 September-October 1998 | Available for purchase: https://www.concrete.org/publications/internationalconcreteabstractsportal?m=details&i=564 |

| Document | ADAMS Accession Number / Federal Register Citation / Report Number and Date | Link to Publication |
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| <p>“Guide to the Evaluation and Management of Concrete Structures Affected by Alkali-Aggregate Reaction”</p> <p>CSA Group</p> | <p>CSA A864-00 Reaffirmed 2005</p> | <p>Available for purchase: http://shop.csa.ca/en/canada/concrete/a864-00-r2005/inv/27010172000</p> |
| <p>“Report on Alkali-Aggregate Reactivity”</p> <p>American Concrete Institute</p> | <p>ACI 221.1R-98 Reaffirmed 2008</p> | <p>Available for purchase: https://www.concrete.org/store/productdetail.aspx?ItemID=221198&Format=DOWNLOAD</p> |
| <p>“ASR/DEF Damaged Bent Caps: Shear Tests and Field Implications”</p> <p>Texas Department of Transportation</p> | <p>Technical Report No. 12-8XXIA006 August 2009</p> | <p>http://library.ctr.utexas.edu/digitized/IACreports/IAC-12-8XXIA006.pdf</p> |
| <p>“Report on the Diagnosis, Prognosis, and Mitigation of Alkali-Silica Reaction (ASR) in Transportation Structures”</p> <p>Federal Highway Administration</p> | <p>FHWA-HIF-09-004 January 2010</p> | <p>https://www.fhwa.dot.gov/pavement/concrete/pubs/hif09004/hif09004.pdf</p> |
| <p>“Structural effects of alkali-silica reaction – technical guidance for the appraisal of existing structures”</p> <p>The Institution of Structural Engineers</p> | <p>April 2010</p> | <p>Available for purchase: http://shop.istructe.org/structural-effects-of-alkali-silica-reaction.html</p> |
| <p>NRC Information Notice 2011-20: Concrete Degradation by Alkali-Silica Reaction</p> <p>NRC</p> | <p>ADAMS Accession No. ML112241029 November 18, 2011</p> | <p>http://www.nrc.gov/docs/ML1122/ML112241029.pdf</p> |
| <p>“Position Paper: In Situ Monitoring of Alkali-Silica Reaction (ASR) Affected Concrete: A Study on Crack Indexing and Damage Rating Index to Assess the Severity of ASR and to Monitor ASR Progression”</p> <p>NRC</p> | <p>ADAMS Accession No. ML13108A047 April 30, 2013</p> | <p>http://www.nrc.gov/docs/ML1310/ML13108A047.pdf</p> |

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| Referenced Documents Specific to Seabrook Station | | |
| <p>“Seabrook Station – Response to Request for Additional Information – NextEra Energy Seabrook License Renewal Application – Aging Management Programs”</p> <p>NextEra</p> | <p>ADAMS Accession No. ML103540534 December 17, 2010</p> | <p>http://www.nrc.gov/docs/ML1035/ML103540534.pdf</p> |
| <p>“Confirmatory Action Letter, Seabrook Station, Unit 1 – Information Related to Concrete Degradation Issues”</p> <p>NRC</p> | <p>ADAMS Accession No. ML12125A172 May 16, 2012</p> | <p>http://www.nrc.gov/docs/ML1212/ML12125A172.pdf</p> |
| <p>“Seabrook Station Response to Confirmatory Action Letter 1-2012-002 and Information Related to Concrete Degradation Issues”</p> <p>NextEra</p> | <p>ADAMS Accession No. ML12151A396 May 24, 2012</p> | <p>http://www.nrc.gov/docs/ML1215/ML12151A396.pdf</p> |
| <p>“Seabrook Station: Impact of Alkali-Silica Reaction on Concrete Structures and Attachments”</p> <p>MPR Associates Inc.</p> | <p>ADAMS Accession No. ML12151A397 May 2012</p> | <p>http://www.nrc.gov/docs/ML1215/ML12151A397.pdf</p> |
| <p>“Seabrook Station Response to Confirmatory Action Letter”</p> <p>NextEra</p> | <p>ADAMS Accession No. ML13151A328 May 1, 2013</p> | <p>http://www.nrc.gov/docs/ML1315/ML13151A328.pdf</p> |
| <p>“Seabrook Station, Unit No. 1 – Confirmatory Action Letter Follow-Up Inspection – NRC Inspection Report 05000443/2012010”</p> <p>NRC</p> | <p>ADAMS Accession No. ML13221A172 August 9, 2013</p> | <p>http://www.nrc.gov/docs/ML1322/ML13221A172.pdf</p> |
| <p>Letter from David Wright, UCS, to NRC Commissioners</p> <p>UCS</p> | <p>ADAMS Accession No. ML13309B606 November 4, 2013</p> | <p>http://www.nrc.gov/docs/ML1330/ML13309B606.pdf</p> |
| <p>Letter from William M. Dean, NRC, to David Wright, UCS</p> <p>NRC</p> | <p>ADAMS Accession No. ML13340A405 December 6, 2013</p> | <p>http://www.nrc.gov/docs/ML1334/ML13340A405.pdf</p> |

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| Letter from Robert M. Taylor, NRC, to Sandra Gavutis, C-10 NRC | ADAMS Accession No. ML16169A172 July 6, 2016 | http://www.nrc.gov/docs/ML1616/ML16169A172.pdf |
| Additional Referenced Documents | | |
| NUREG-1801, "Generic Aging Lessons Learned Report," Revision 2 | December 2010 | http://www.nrc.gov/reacting-rm/doc-collections/nuregs/staff/sr1801/ |
| RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3 | ADAMS Accession No. ML113610098 May 2012 | http://www.nrc.gov/docs/ML1136/ML113610098.pdf |
| IMC 0326, "Operability Determinations and Functionality Assessments for Conditions Adverse to Quality or Safety" NRC | ADAMS Accession No. ML13274A578 January 31, 2014 | http://www.nrc.gov/docs/ML1327/ML13274A578.pdf |
| "Davis-Besse Nuclear Power Station – Inspection of Apparent Cause Evaluation Efforts for Propagation of Laminar Cracking in Reinforced Concrete Shield Building and Closure of Unresolved Item Involving Shield Building Laminar Cracking Licensing Basis – Inspection Report 05000346/2014008" NRC | ADAMS Accession No. ML15148A489 May 28, 2015 | http://www.nrc.gov/docs/ML1514/ML15148A489.pdf |

Dated at Rockville, Maryland, this day of , 2018.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,
Secretary of the Commission.

SUBJECT: PRM-50-109 - IMPROVED IDENTIFICATION TECHNIQUES AGAINST ALKALI-SILICA REACTION (ASR) CONCRETE DEGRADATION AT NUCLEAR POWER PLANTS
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