Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Davis-Besse Nuclear Power Station, Unit No. 1, dated October 1975.

Agencies and Persons Consulted

In accordance with its stated policy, on April 18, 1997, the staff consulted with the Ohio State official, Carol O'Claire, of the Ohio Emergency Management Agency, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the Toledo Edison Company and Centerior Service Company submittal dated December 13, 1996, supplemented by letter dated February 14, 1997, which are available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606.

Dated at Rockville, Maryland, this 7th day of May 1997.

For the Nuclear Regulatory Commission.

Allen G. Hansen,

Project Manager, Project Directorate III–3, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

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NUCLEAR REGULATORY COMMISSION

Proposed Generic Letter; Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in the Containment (TAC N0. M97146)

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of opportunity for public comment.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to issue a generic letter to licensees of operating nuclear power reactors regarding the potential for degradation of the emergency core cooling system (ECCS) and the containment spray system (CSS) after a loss-of-coolant accident (LOCA) because of construction and protective coating deficiencies and foreign material that may be present in the containment. The NRC is issuing this generic letter to alert licensees to the fact that foreign material continues to be found inside operating nuclear power plant containments. During a design basis LOCA, this foreign material could block the ECCS or safety-related CSS flow path or damage ECCS or safety-related CSS equipment. In addition, construction deficiencies and problems with the material condition of ECCS systems, structures, and components (SSCs) inside the containment continue to be found. Design deficiencies also have been found which could potentially degrade the ECCS or safety-related CSS. No actions or information are requested regarding these issues. The NRC has issued many previous generic communications on this subject and expects licensees to have considered possible actions at their facilities to address these concerns.

The NRC is also issuing this generic letter to alert licensees to the problems associated with the material condition of protective coatings inside the containment and to request information under 10 CFR 50.54(f) to evaluate their programs for ensuring that protective coatings do not detach from their substrate during a design basis LOCA and interfere with the operation of the ECCS and the safety-related CSS. The NRC intends to use this information to assess whether current regulatory requirements are being correctly implemented and whether they should be revised.

The NRC expects addressees to ensure that the ECCS and the safety-related CSS remain capable of performing their intended safety functions. The NRC will conduct inspections to ensure compliance with existing licensing bases and respond to discovered inadequacies with aggressive enforcement consistent with its enforcement policy.

The NRC is seeking comment from interested parties regarding both the technical and regulatory aspects of the proposed generic letter presented under the SUPPLEMENTARY INFORMATION heading.

The proposed generic letter was endorsed by the Committee to Review Generic Requirements (CRGR) on May 5, 1997. The relevant information that was sent to the CRGR will be placed in the Public Document Room. The NRC will consider comments received from interested parties in the final evaluation of the proposed generic letter. The final evaluation by the NRC will include a review of the technical position and, as appropriate, an analysis of the value/impact on licensees. Should this generic letter be issued by the NRC, it will become available for public inspection in the Public Document Room.

DATES: Comment period expires June 27, 1997. Comments submitted after this date will be considered if it is practical to do so; assurance of consideration can only be given for those comments received on or before this date.

ADDRESSES: Submit written comments to Chief, Rules Review and Directives Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Written comments may also be delivered to 11545 Rockville Pike, Rockville, Maryland, from 7:30 am to 4:15 pm, Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, 2120 L Street, NW, (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Richard M. Lobel (301) 415–2865 or James A. Davis (301) 415–2713.

SUPPLEMENTARY INFORMATION:

NRC Generic Letter 97–XX: Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in the Containment

Addressees

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter for several reasons. It alerts addressees that foreign material continues to be found inside operating nuclear power plant containments. During a design basis loss-of-coolant accident (DB LOCA), this foreign material could block an emergency core cooling system (ECCS) or safety-related containment spray system (CSS) flow path or damage ECCS or safety-related

CSS equipment. In addition, construction deficiencies and problems with the material condition of ECCS systems, structures, and components (SSCs) inside the containment continue to be found. Design deficiencies also have been found which could potentially degrade the ECCS or safety-related CSS. No actions or information are requested regarding these issues. The NRC has issued many previous generic communications on this subject, as discussed later in this generic letter, and expects the addressees to have considered possible actions at their facilities to address these concerns.

The NRC is also issuing this generic letter to alert the addressees to the problems associated with the material condition of protective coatings inside the containment and to request information under 10 CFR 50.54(f) to evaluate the addressees' programs for ensuring that protective coatings do not detach from their substrate during a DB LOCA and interfere with the operation of the ECCS and the safety-related CSS. The NRC intends to use this information to assess whether current regulatory requirements are being correctly implemented and whether they should be revised.

The NRC expects addressees to ensure that the ECCS and the safety-related CSS remain capable of performing their intended safety functions. The NRC will conduct inspections to ensure compliance with existing licensing bases and respond to discovered inadequacies with aggressive enforcement consistent with its enforcement policy.

Background

Foreign Material Exclusion, Construction Deficiencies and Design Deficiencies

In some recent events, foreign material, which could have affected the operation of the ECCS, was discovered inside the containment. As part of its review of these events, the NRC staff reviewed the history of such events and identified several related problems.

These events are discussed in Appendix A to this generic letter. A more complete list of the previous events is provided in Appendix B. As discussed in Appendix A, almost all of these events have been the subject of previous NRC generic communications and licensee event reports (LERs). The following types of problems continue to occur.

(1) Foreign material has been found in areas of the containment where it could be transported to the sump(s) or the suppression pool and potentially affect the operation of the ECCS or safety-related CSS. Such material has also been found in PWR sumps, in BWR suppression pools and downcomers, and in safety-related pumps and piping.

(2) Deficiencies have been found in the construction of the ECCS sumps or strainers. These deficiencies, which could have impaired the operation of the ECCS or the safety-related CSS, include missing screens, unintended openings in screens, and screens that are incorrectly sized.

(3) Problems have also been found with the material condition of sumps or suction strainers, potentially impairing the operation of the ECCS or safety-related CSS. These problems include deformed suction strainers and unintentional flow paths created by missing grout.

(4) Design deficiencies have been found, including valves in flow lines with clearances smaller than the sump screen mesh size and strainers with a flow area smaller than required.

(5) There have been two incidents, described in LERs, in which doors to emergency sump structures were left open when ECCS and safety-related CSS operability was required by the technical specifications.

The Discussion section of this generic letter discusses the regulatory and safety basis for these concerns.

It is evident that past NRC generic communications have not been completely effective in achieving an acceptable level of control of these problems. Nevertheless, the NRC expects that licensees will ensure that the ECCS and safety-related CSS remain capable of performing their intended safety functions.

The NRC plans to further emphasize this issue by conducting inspections to ensure compliance with the existing plant licensing basis and to respond to discovered inadequacies with aggressive enforcement consistent with the NRC enforcement policy.

Protective Coatings

Protective coatings inside nuclear power plant containments serve three general purposes. Protective coatings are applied to steel, aluminum, and galvanized surfaces to control corrosion. Protective coatings are applied to surfaces to control radioactive contamination levels. Protective coatings are also applied to protect surfaces from erosion and wear.

Protective coatings inside the containment and the regulatory requirements and guidance for their use are discussed in Appendix C.

Qualified protective coatings are capable of adhering to their substrate during a DB LOCA in order to minimize the amount of material which can reach the emergency sump screens or suction strainers and clog them. Not all coatings inside the containment are qualified. The amount of unqualified coatings must be limited since the unqualified coatings are assumed to detach from their substrates during a DB LOCA or steam line break and may be transported to the emergency sump screens or suction strainers.

In some cases, coatings which should have been qualified failed during normal operation. Some of these events are discussed in Appendix D.

Discussion

NRC regulations in 10 CFR 50.46 require that licensees design their ECCS to provide long-term cooling capability so that the core temperature can be maintained at an acceptably low value and decay heat can be removed for the extended period required by the long-lived radioactivity remaining in the core. This criterion must be demonstrated while assuming the most conservative single failure. Some addressees may credit CSSs for pressure and radioactive source term reduction as part of the licensing basis. These CSSs may also take suction from the suppression pools or emergency sumps.

Foreign materials, degraded coatings inside the containment that detach from their substrate, and ECCS components not consistent with their design basis, along with LOCA-generated debris, are potential common-cause failure mechanisms which may clog suction strainers, sump screens, filters, nozzles, and small-clearance flow paths in the ECCS and safety-related CSS and thereby interfere with the long-term cooling function.

Qualified coatings used inside containment must be demonstrated to be capable of withstanding the environmental conditions of a postulated DB LOCA without detaching from their substrates (detached coatings may then be transported to the sumps or strainers and cause or contribute to flow blockage). The LERs and NRC inspection reports described in Appendix D of this generic letter provide evidence of weaknesses in addressee programs with regard to applications of protective coatings for Class I service. These weaknesses include deficiencies in addressee programs to (1) Control the preparation and cleanliness of the substrate before the coatings are applied, (2) control the preparation of paint before its application, (3) control the dry film thickness of coatings applied to the substrate, (4) monitor for and control the use of excessive amounts of unqualified coatings inside the containment, (5) monitor the status of ''qualified'' coatings already applied to the surfaces of the containment structure and to other equipment inside the containment, and (6) assess the safety significance of coatings inside containment that have been determined to detach from their substrate and to repair these coatings, if necessary.

The NRC has issued a number of generic communications on various aspects of the potential for the loss of the ECCS and safety-related CSS as a result of strainer clogging and debris blockage. These generic communications are listed in Appendix E. The basic safety concern applies to both PWRs and BWRs. These events, discussed in these generic communications, as well as similar events described in LERs and NRC inspection reports, demonstrate the need for a strong foreign material exclusion (FME) program in all areas of PWRs and BWRs that may contain materials that could interfere with the successful operation of the ECCS. Other events demonstrate the need to ensure the correct design and to maintain the material condition of emergency core cooling system and safety-related containment spray system SSCs, including the suppression pools, ECCS strainers and sumps, and the protective coatings inside containment.

The requirements of 10 CFR Part 50, Appendix B, are germane to this issue.

The maintenance rule, 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' includes in its scope all safety-related SSCs, and those non-safety-related SSCs that fall into the following categories: (1) Those that are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures; (2) those whose failure could prevent safety-related SSCs from fulfilling their safety-related function; and (3) those whose failure could cause a reactor scram or an actuation of a safety-related system.

The PWR sumps and BWR strainers are included within the scope of the maintenance rule.

To the extent that protective coatings meet these scoping criteria, they are within the scope of the maintenance rule.

The maintenance rule requires that licensees monitor the effectiveness of maintenance for these protective coatings (as discrete systems or components or as part of any SSC) in accordance with paragraph (a)(1) or (a)(2) of 10 CFR 50.65, as appropriate.

The NRC expects all addressees to have programs and procedures in place to ensure that the ECCS and the safety-related CSS are not degraded by foreign material in the containment, that the ECCS and the safety-related CSS are consistent with their design and licensing bases, and that sumps, strainers, and coatings are in good material condition. The staff may evaluate the condition of sumps, strainers and protective coatings as a part of maintenance rule inspections.

The NRC has conducted numerous inspections in the areas addressed by this generic letter; for example, the NRC issued Technical Instruction 2515/125, ''Foreign Material Exclusion Controls,'' on August 25, 1994. Violations have been identified and appropriate enforcement action has been taken in accordance with the NRC's Enforcement Policy (NUREG–1600, ''General Statement of Policy and Procedures for NRC Enforcement Actions: Enforcement Policy''). A list of significant enforcement actions is provided in Appendix F of this generic letter. The NRC intends to continue to conduct inspections in order to ensure compliance with the existing licensing basis and to respond to discovered inadequacies with aggressive enforcement consistent with the NRC Enforcement Policy.

The NRC will consider violations in this area as significant regulatory failures and will, accordingly, consider categorizing inadequacies at least as Severity Level III violations. The NRC will also consider the long history of generic communications on this issue as prior notice to licensees when the agency assesses civil penalties in accordance with Section VI.B.2 of the Enforcement Policy. Finally, notwithstanding the normal civil penalty assessment, the NRC will consider whether the circumstances of the case warrant escalation of enforcement sanctions in accordance with Section VII.A.1 of the Enforcement Policy.

If in the course of assessing the effectiveness of the plant-specifc FME program or preparing a response to the requested information it is determined that a facility is not in compliance with the Commission's rules or regulations, the addressees are expected to take whatever actions are deemed appropriate in accordance with requirements stated in Appendix B to 10 CFR 50 and as required by the plant technical specifications to restore the facility to compliance.

Required Information

Within 75 days of the date of this generic letter, addressees are required to submit a written response that includes the following information:

(1) A summary description of the plant-specific program implemented to ensure that Class I protective coatings used inside the containment are procured, applied, and maintained in compliance with applicable regulatory requirements and the plant-specific licensing basis for the facility. Include a discussion of how the plant-specific program meets the applicable criteria of 10 CFR Part 50, Appendix B, as well as information regarding any applicable standards, plant-specific procedures or other guidance used for (a) Controlling the procurement of coatings and paints used at the facility; (b) the qualification testing of protective coatings; and (c) surface preparation, application, surveillance, and maintenance activities for protective coatings.

(2) Information demonstrating compliance with your plant-specific licensing basis related to tracking the amount of unqualified coatings inside the containment and for assessing the impact of potential coating debris on the operation of safety-related SSCs during a postulated DB LOCA.

Include the following information in the discussion to the extent it is available:

(a) The date and findings of the last assessment of coatings, and the planned date of the next assessment of coatings

(b) The limit for the amount of unqualified protective coatings allowed in the containment and how this limit is determined. Discuss any conservatisms in the method used to determine this limit.

(c) If a commercial-grade dedication program is being used at your facility for dedicating commercial-grade coatings for Class I applications inside the containment, describe why the program is sufficient to qualify such a coating for Class I service. Identify what standards or other guidance are currently being used to dedicate containment coatings at your facility.

(d) If a commercial-grade dedication program is not being used at your facility for qualifying and dedicating commercial-grade coatings for use inside containment for Class I applications, provide the regulatory and safety basis for not controlling these coatings in accordance with such a program. Additionally, explain why the facility's licensing basis does not require such a program.

Address the required written information to the U.S. Nuclear

Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). This information will enable the Commission to determine whether the license should be modified, suspended, or revoked. In addition, submit a copy of the written information to the appropriate regional administrator.

Backfit Discussion

This generic letter requires information from the addressees under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR Part 50.54(f). This generic letter does not constitute a backfit as defined in 10 CFR 50.109(a)(1) since it does not impose modifications of or additions to systems, structures, and components or to design or operation of an addressee's facility. It also does not impose an interpretation of the Commission's rules that is either new or different from a previous staff position. The staff has, therefore, not performed a backfit analysis.

Reasons for Information Request

This generic letter transmits an information request pursuant to the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f) for the purpose of verifying compliance with applicable regulatory requirements. Specifically, the requested information will enable the NRC staff to determine whether the addressees' protective coatings inside the containment comply and conform with the current licensing basis for their respective facilities and whether the regulatory requirements pursuant to 10 CFR 50.46 are met.

Protective coatings are necessary inside containment to control radioactive contamination and to protect surfaces from erosion and corrosion. Detachment of the coatings from the substrate may make the ECCS unable to satisfy the requirement of 10 CFR 50.46(b)(5) to provide long-term cooling and make the safety-related CSS unable to satisfy the plant-specific licensing basis by controlling containment pressure and radioactivity following a LOCA.

Appendix A—Discussion of Events Related to ECCS Sumps and Strainers Including Foreign Material Inside the Containment and Construction and Design Deficiencies

On November 16, 1988, the NRC issued Information Notice (IN) 88–87, ''Pump Wear and Foreign Objects in Plant Piping Systems,'' concerning several incidents in which the potential existed for a flow reduction as a result of pump wear and foreign objects in plant piping systems. In one of these incidents, the licensee found foreign objects in a temporary pump discharge cone strainer. The licensee investigated further and found foreign objects, dating to early construction modifications, in the sump. In addition, various deficiencies were found in the sump screens.

On November 21, 1989, the NRC issued IN 89–77, ''Debris in Containment Emergency Sumps and Incorrect Screen Configurations,'' which discussed loose parts and debris in the containment sumps of three pressurized-water reactors (PWRs), Surry Units 1 and 2 and Trojan. At Surry Units 1 and 2, some of the debris was large enough to cause pump damage or flow degradation. In addition, some of the screens had gaps large enough to allow additional loose material to enter the sump. The licensee found that screens that separate the redundant trains of the inside recirculation spray system were missing at both units. At Trojan, the licensee discovered debris in the sump. Some debris was found after containment closeout. In addition, still later, before startup, the NRC identified missing portions of the sump top screen and inner screen. IN 89–77 also reported that in 1980 the Trojan licensee found a welding rod jammed between the impeller and the casing ring of a residual heat removal pump.

On December 23, 1992, the NRC issued IN 92–85, ''Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage,'' which alerted licensees to events at two PWRs. In these events, foreign material blocked flow paths within the ECCS safety injection and containment spray pumps so that the pumps could not produce adequate flow.

On April 26, 1993, and May 6, 1993, the NRC issued IN 93–34, ''Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment,'' and its supplement. In these information notices, the NRC described several instances of clogged ECCS pump strainers, including two events at the Perry Nuclear Power Plant, a domestic boiling-water reactor (BWR). In the first Perry event, residual heat removal (RHR) strainers were clogged by operational debris consisting of ''general maintenance-type material and a coating of fine dirt.'' After cleaning the strainers in January 1993, the licensee discovered that RHR A and B strainers were deformed. The strainers were replaced. The second Perry event involved an RHR pump test which was run after a plant transient in March 1993. Pump suction pressure dropped to 0 KPa (0 psig). No change in pump flow rate was observed. Material found on the strainer screen was analyzed and found to consist of glass fibers from temporary drywell cooling filters that had been inadvertently dropped into the suppression pool and corrosion products that had been filtered from the pool by the glass fibers adhering to the surface of the strainer. This significantly increased the pressure drop across the strainer.

In response to these two events, the licensee for Perry increased the suction strainer area, provided suction strainer backflush capability, and improved measures to keep the suppression pool clean.

On May 11, 1993, the NRC issued Bulletin 93–02, ''Debris Plugging of Emergency Core Cooling Suction Strainers,'' which requested that both PWR and BWR addressees (1) identify fibrous air filters and other temporary sources of fibrous material in containment not designed to withstand a loss-of-coolant accident (LOCA) and (2) take prompt action to remove the foreign matter and ensure the functional capability of the ECCS. All addressees have responded to the bulletin, and the NRC staff has completed its review of their responses.

The licensee for Arkansas Nuclear One, Unit 2, reported by Licensee Event Report (LER) 93–002–00, dated November 22, 1993, that the containment sump integrity was inadequate to keep foreign material out. Holes in the masonry grout below the sump screen assembly would have let water into the sump without being screened. The licensee attributed this condition to failure to implement design basis requirements for the sump during initial plant construction. The holes were difficult to detect. The holes appeared to be part of the design because of their uniform spacing and because they were ''somewhat recessed * * * such that to see the holes they must be viewed from near the floor or from a significant distance away from the sump.''

On August 12, 1994, the NRC issued IN 94–57, ''Debris in Containment and the Residual Heat Removal System,'' which alerted operating reactor licensees to additional instances of degradation of ECCS components because of debris. At River Bend Station, the licensee found a plastic bag on an RHR suction strainer. At Quad Cities Station, Unit 1, on July 14, 1994, the remains of a plastic bag were found shredded and caught within the anti-cavitation trim of an RHR test return valve. Subsequent to that event at Quad Cities, Unit 1, the licensee observed reduced flow from the ''C'' RHR pump and, upon further investigation, found a 10-cm (4-in.) diameter wire brush wheel and a piece of metal wrapped around a vane of the pump.

On January 25, 1995, the NRC issued IN 95–06, ''Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment,'' which discussed a concern that plastic or fibrous material, brought inside the containment to reduce the spread of loose contamination, to identify equipment, or for cleaning purposes, may collect on screens and strainers and block core cooling systems. Several examples were cited.

On October 4, 1995, the NRC issued IN 95–47, ''Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage,'' which discussed an event on September 11, 1995, at the Limerick Generating Station, Unit 1, during which a safety/relief valve discharged to the suppression pool. The operators started an RHR pump in the suppression pool cooling mode. After 30 minutes, fluctuating motor current and flow were observed. Subsequent inspection of the strainers found them covered with a ''mat'' of fibrous material and sludge (corrosion products) from the suppression pool. The licensee removed approximately 635 kg (1400 lb) of debris from the Unit 1 pool. A similar amount of debris had been removed earlier from the Unit 2 pool. A supplement to IN 95–47 was issued on November 30, 1995.

On October 17, 1995, the NRC issued NRC Bulletin 95–02, ''Potential Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode,'' which discussed the Limerick Unit 1 event and requested that BWR addressees review the operability of their ECCS and other pumps that draw suction from the suppression pool while performing their safety function. The addressees' evaluations were to take into consideration suppression pool cleanliness, suction strainer cleanliness, and the effectiveness of the addressees' foreign material exclusion (FME) practices. In addition, BWR addressees were requested to implement appropriate procedural modifications and other actions (e.g., suppression pool cleaning), as necessary, in order to minimize the amounts of foreign material in the suppression pool, drywell, and containment. BWR addressees were also requested to verify their operability evaluation through appropriate testing and inspection.

On February 10, 1996, the NRC issued IN 96–10, ''Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillances,'' which discussed debris blockage in ECCS lines taking suction from the containment sumps at a PWR in Spain. In one of the two partially blocked lines, almost half the flow area of the pipe was blocked off; the other line was less blocked. Upon further investigation, Spanish regulators found that many sections of piping in both PWRs and BWRs are only called upon to function during accident conditions and are not used during normal operation or tested during functional surveillance tests. The licensee in this case concluded that the safety significance was low because the partial blockage of the lines would not have prevented the ECCS from providing sufficient core cooling. However, it was also noted that some of the debris could have been entrained in the water flow and could have detrimental effects on other parts of the system (e.g., pump and valve components and heat exchangers).

In addition, in LER 96–005, the licensee for the H.B. Robinson Steam Electric Plant, Unit 2, reported finding an item of debris larger than the ⅜-inch diameter of the holes in the containment spray nozzle in a pipe in the sump.

In LER 96–007, the licensee for Diablo Canyon Nuclear Power Plant, Unit 1, reported a radiograph inspection finding that openings in the Diablo Canyon plant's 3.81-cm (1½ in.) centrifugal charging pump runout protection manual throttle valves and safety injection (SI) to cold-leg 5.08-cm (2-in.) manual throttle valves were less than the 0.673-cm (0.265-inch) diagonal opening in the containment recirculation sump debris screen. Therefore, debris could potentially block charging or SI flow through these throttle valves during the recirculation phase of a LOCA. The licensee concluded that even with a postulated blockage of the throttle valves, the RHR system flow by itself would be sufficient to maintain adequate core cooling during recirculation following a postulated accident. As a corrective action, the Diablo Canyon licensee stated in LER 96–007 that the system would be modified to ensure that the throttle valve clearance is greater than the maximum sump screen opening.

After reviewing an Institute of Nuclear Power Operations (INPO) operational experience report on this event, the licensee for Millstone Nuclear Station, Unit 2, determined that eight throttle valves in the high-pressure safety-injection (HPSI) system injection lines were susceptible to the failure mechanism described in the Diablo Canyon Nuclear Power Plant LER 96–007. This situation is discussed in NRC IN 96–27, ''Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation,'' dated May 1, 1996. The Millstone Unit 2 licensee concluded that the type of debris that would pass through the screen openings would tend to be of low density and low structural strength and that material of this type would be reduced in size as it passed through the HPSI and containment spray pumps. In addition, the differential pressure across the HPSI system injection valves and containment spray nozzles would tend to force through the valves or nozzles any material that is ''marginally capable'' of obstructing flow. These conclusions may be plant specific and may not be applicable to other designs. The Millstone Unit 2 licensee committed to replace the sump screen with one that is consistent with the original design.

On May 6, 1996, the NRC issued Bulletin 96–03, ''Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,'' which requested actions by BWR addressees to resolve the issue of BWR strainer blockage because of excessive buildup of debris from insulation, corrosion products, and other particulates, such as paint chips and concrete dust. The bulletin proposed four options for dealing with this issue: (1) install large-capacity passive strainers, (2) install self-cleaning strainers, (3) install a safety-related backflush system that relies on operator action to remove debris from the surface of the strainer to keep it from clogging, or (4) propose another approach that offers an equivalent level of assurance that the ECCS will be able to perform its safety function following a LOCA. BWR addressees were requested to implement the requested actions of Bulletin 96–03 by the end of the first refueling outage beginning after January 1, 1997.

On October 30, 1996, the NRC issued IN 96–59, ''Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris,'' to alert addressees that the suppression pool and associated components of two BWRs, LaSalle County Station, Unit 2, and Nine Mile Point Nuclear Station, Unit 2, were found to contain foreign objects that could have impaired successful operation of emergency safety systems that used water from the suppression pool. In particular, debris was found in the downcomers (large-diameter pipes connecting the drywell to the suppression pool). Although the licensee for Nine Mile Point, Unit 2, had previously cleaned the suppression pool, the downcomers had not been inspected. In addition, the licensee found debris covers in place on seven of the eight downcomers located in the pedestal area directly under the reactor vessel. These debris covers had been in place since construction. LER 96–11–00 attributes this oversight to inadequate managerial methods and to environmental conditions since the ''accessibility of the pedestal area downcomers requires removal of grating in the undervessel area and climbing down to the dimly lit subpile floor. The plastic covers on the downcomers are not visible from the grating elevation because of the missile shield plates above the downcomer floor penetrations. Furthermore, since the first refueling outage, access to this area has been limited because of the high contamination levels and general ALARA [as low as reasonably achievable radiation dose] considerations.''

Although the NRC has not previously discussed the subject in a generic communication, licensee event reports have been submitted regarding the loss of control of containment sump access hatches, leaving them open during periods when ECCS sump integrity was required. For example, the licensee for Diablo Canyon Nuclear Power Plant, Unit 1, in LER 89–014–01, discussed the opening of the sump access hatch at various times at power ''without adequate consideration of ECCS operability.'' LER 96–006 (Watts Bar Nuclear Plant, Unit 1) reported that an operator observed a containment sump (trash screen) door open while ECCS operability was required.

Appendix B—Operational Events Involving ECCS and Safety-Related Containment Spray Recirculation Flow Paths

Plant/report	Problems discussed
Haddam Neck NRC Inspection Report 50–213/96–08.	Six 55 drums of sludge with varying amounts of debris removed from ECCS sump (July 1975).
North Anna Units 1 and 2 LER 84–006–00	Galvanized ductwork painted with unqualified paint.
Millstone Unit 1 LER 88–004–00	Existing suction strainers smaller than allowed by criteria of RG 1.82 Rev.1. Strainers will be replaced with larger strainers if Integrated Safety Assessment Program criteria met.

Plant/report	Problems discussed
Surry Power Station Units 1 and 2 LER 88–017–01 IN 88–87 IN 89–77.	1. Foreign material from construction activities found in cone strainer of recirculation spray system. Material could have rendered system inoperable. 2. Gaps in sump screens since initial construction.
Trojan Nuclear Plant LER 89–016–01 IN 89–77	1. Wire mesh screen on top of sump trash rack not installed. 2. Screen damage. 3. Significant amount of debris discovered in the sump. Could have caused loss of a portion of ECCS.
Diablo Canyon Unit 1 LER 89–014–01 IN 89–77	1. Debris in sump. 2. As-built sump configuration not in accordance with design. 3. Safety function would not have been impaired.
TMI Unit 1. LER 90–002–00	Modification of sump access hatches left holes in top of sump screen cage. Potentially could damage pumps or clog spray nozzles.
McGuire Unit 1 LER 90–0112–00	Loose material discovered in upper containment prior to entry into Mode 4. Items found would not have made ECCS inoperable.
Calvert Cliffs Units 1 and 2 NRC Inspection Report March 5, 1991.	Unit 2 sump found to contain 25 lbs dirt, weld slag, pebbles, etc. Inspection of Unit 1 found less than 1 lb. debris. Possible minor damage to ECCS pumps.
Diablo Canyon Unit 2 LER 91–012–00	1. Numerous instances of material left unattended or abandoned in sump level of containment (tools, plastic tool bags, clothing, etc.). 2. Material would not have prevented ECCS recirculation function.
H.B. Robinson Unit 2 LER 92–013–00	''B'' safety injection pump reduced flow due to blockage in minimum flow recirculation check valve and flow orifice on July 8, 1992. ''A'' pump OK. Foreign material also found in refueling water storage tank (RWST).
H.B. Robinson Unit 2 LER 92–018–00	On August 24, 1992, following a reactor trip, ''A'' and ''B'' safety injection pumps inoperable due to reduced flow. Found during unscheduled surveillance to demonstrate safety injection (SI) operability.
Pt. Beach Unit 2 LER 92–003–01 IN 92–85	September 18, 1992: During technical specifications (inservice) testing of the ''A'' containment spray pump, the pump was declared inoperable. A foam rubber plug was blocking pump suction. Plug removed and pump tested satisfactorily. One train of Unit 2 residual heat removal, safety injection and containment spray systems inoperable for entire operating cycle. Plug was part of a cleanliness barrier.
Perry Nuclear Plant LER 93–011–00	May 1992: During refueling outage foreign objects discovered in the containment side of the suppression pool. Fouling of residual heat removal (RHR) strainers found. Strainers not cleaned. January 1993: RHR ''A'' and ''B'' strainers found deformed (collapsed inward in the direction of the fluid flow. Strainers replaced. March 1993: RHR ''A'' and ''B'' operated in suppression pool cooling mode. Pump suction pressure decreased. Could have compromised long-term RHR operation.
Susquehanna Units 1 and 2 LER 93–007–00 (Voluntary).	1. Assessing impact of debris and corrosion products adhering to fibrous materials that may be dislodged by a pipe break. 2. Developing procedures to backflush strainers.
Sequoyah Unit 2 LER 93–026–00	Design basis limit for unqualified coatings inside containment had been exceeded. Additional quantity of unqualified coatings on reactor coolant pump motor platform discovered. Path to ECCS sump. Screens will be installed before startup.
ANO Unit 2 LER 93–002–00 IN 89–77 Supplement 1.	Seven unscreened holes found in masonry grout below screen assembly of ECCS sump. Could potentially degrade both trains of the high pressure coolant injection system and containment spray. Had previously inspected sump because of IN 89–77. Did not discover problem. NRC estimate of incremental increase in core damage: 3×10^{-04}.
ANO Unit 1 LER 93–005–00 IN 89–77 Supplement 1.	1. 22 unscreened 6×3 pipe openings at base of sump curb. Occurred as a result of modification prior to initial operation. 2. Tears in screen. 3. Floor drains leading to sump not screened. 4. Licensee estimated increase in core damage frequency 5×10^{-05}.
San Onofre Units 1 and 2 LER 93–010–00 (Voluntary).	1. Irregular annular gap (approximately 6) surrounding 8 low temperature overpressure protection system discharge line penetrating horizontal steel cover plate. 2. Engineering analysis concluded both sump trains operable.
Vermont Yankee LER 93–015–00	1. Low pressure core spray suction strainers smaller than calculations assumed. Net positive suction head calculations performed in 1986 following change to NUKON™ insulation invalid. 2. Strainers replaced with larger strainers.
South Texas Unit 1/2 LER 94–001–00	1. Sump screen openings from initial construction discovered. Frame plate at floor warped, creating several openings approximately ⅝″. Additional ¼″ gaps discovered. Licensee concluded there was no safety significance to these deficiencies based on ECCS pump tests performed by the manufacturer.
Point Beach Unit 1 NRC Inspection Report May 6, 1994.	NRC inspector found grout deterioration under sump screens. Could result in flow bypass or particles of grout entering ECCS pumps.
LaSalle Unit 1 IN 94–57	April 26 and May 11, 1994: Divers inspecting suppression pool during outage found operational debris.
River Bend IN 94–57	June 13, 1994: Plant in refueling outage. Foreign material found in suppression pool. Plastic bag removed from ''B'' RHR pump suction strainer. Other objects: tools, grinding wheel, scaffolding knuckle, step off pad.

Plant/report	Problems discussed
Quad Cities Unit 1 IN 94–57	July 14, 1994: Post-maintenance test of "A" loop RHR indicated a plugged torus cooling test return valve. Inspection discovered remains of shredded plastic bag in anti-cavitation trim installed during a recent outage. July 23, 1994: 4″ diameter wire brush and a piece of metal found wrapped around a vane of the "C" RHR pump.
Browns Ferry Units 1/2/3 May 20, 1994 Letter to NRC.	1. Unqualified coatings on T quenchers in suppression pool. 2. Continued operation acceptable. 3. Will remove coatings next refueling outage.
Palisades Plant LER 94–014–00	Signs, adhesive tape, and labels with potential to block the ECCS sump were found in containment. Containment spray and HPSI pumps declared inoperable. Engineering analysis concluded that the sump screen would not be significantly blocked.
Watts Bar Units 1 and 2 NRC Inspection Report 50–390 and 50–391/94–59 September 28, 1994.	Screens installed around reactor coolant pump motors to catch unqualified paint not adequately located to contain all unqualified coatings.
Indian Point Unit 2 LER 95–005–00	Licensee discovered portions of floor coating on containment Elevation 46 had lifted and cracked. In other locations, floor coating cracked when stepped on. Licensee concluded that sump function would not be compromised.
Susquehanna Units 1 and 2 LER 93–007–001 September 11, 1995.	Licensee took actions to address clogging ECCS suction strainers: removal of fibrous insulation from high energy line break areas, testing to characterize the debris threat to strainer blockage, quantification of corrosion products on structural steel in wetwell, establishment of a comprehensive analysis of containment debris effects. Coating and insulation procedures contain steps to reduce potential for strainer blockage.
Prairie Island Unit 2 NRC Inspection Report 50–282/05–009.	Broken labels for pipe hangers and labels affixed to wall with degrading adhesive discovered by NRC inspector after licensee closeout inspection. Licensee concluded that this would not affect operability of ECCS.
Palisades NRC Inspection Report 50–225/95–008.	Unsecured material stored on the landings of stairways. Broken glass and pieces of signboard and other "unauthorized" material found in area designated debris-free.
Limerick Unit 1 NRC Inspection Report 50–352/96–04.	Debris was allowed to collect in suppression pool so that "A" RHR pump was rendered inoperable when safety/relief valve lifted on September 11, 1995.
Duane Arnold NRC Inspection Report 50–331/95–003.	Foreign material exclusion controls inadequate in drywell. Hardhats and debris noted.
Foreign PWR NRC IN 96–10	1. Operator found debris in the sump. 2. Two of 4 ECCS lines taking suction from the sump were partially blocked by debris. Debris present since plant construction.
Millstone Unit 2 LER 96–008	Ten locations inconsistent with the specified screen opening size were identified. Placed plant outside original design basis. Sump screen replaced.
Watts Bar Unit 1 LER 96–006–00	Operator observed containment sump trash screen door was open when plant was in MODE 4 and ECCS required to be operable.
Calvert Cliffs Units 1 and 2 LER 96–003–00	Several holes identified in each units' containment sump screen larger than described in the Final Safety Analysis Report. Holes field-installed for transmitter tubing. Concluded not a threat to plant safety.
Diablo Canyon Unit 1 LER 96–007–00	Various debris that could pass through the containment sump screen could be larger than minimum clearances in the 1½″ centrifugal charging pump runout protection manual throttle valves and 2″ SI cold leg manual throttle valves.
Haddam Neck LER 96–014–00 NRC Inspection Report 50–213/96–08.	1. Discrepancies in sump screen mesh sizing, screen fitup, and method of attachment discovered. Sump screen replaced. Sump will be inspected after every refueling outage. Licensee reported that this condition could have prevented the fulfillment of a safety function. 2. Five 55-gallon drums of sludge removed from ECCS sump. Also, plastic, nuts and bolts, tie wraps, and pencil.
Big Rock Point NRC Inspection Report 50–155/96–004.	"Housekeeping in containment in the area under the emergency condenser and the reactor depressurization system isolation valves was poor."
Catawba Unit 1 NRC Inspection Report 50–413/96–11.	Six floor drains inside crane wall were not covered with screen that had a finer mesh than the sump screen. The holes were ¼″ rather than ⅛″ holes. Crane wall penetrations close to containment floor could allow the transport of debris to the sump screen. Penetrations sealed.
Millstone Unit 2 LER 50–336/96–08 NRC Inspection Report 50–336/96–08.	Containment sump screens had been incorrectly constructed so that larger debris than analyzed could pass through the ECCS.
Vogtle Unit 2 NRC Inspection Report 50–425/96–11 LER 96–007–00.	Containment integrity was established prior to startup. Upon subsequent containment entries personnel discovered various items of loose debris. Material removed while in MODE 4. Material would have resulted in inadequate NPSH for the "B" train of RHR and containment spray. NPSH for the "A" train of RHR and containment spray would have been adequate.
Nine Mile Point Unit 2 NRC Inspection Report 50–410/96–11 NRC Event Report 31172.	A significant amount of debris was found in the suppression pool and downcomers during refueling outage 5. The licensee's preliminary evaluation concluded that operability of ECCS could have been compromised.
LaSalle Unit 2 NRC Event Report 31159 LER 96–009–00.	Substantive foreign material recovered from suppression pool and downcomers which would challenge the operability of the ECCS. Items most likely from construction or early outages.
Millstone Unit 3 LER 96–039–00	1. Construction debris discovered in containment recirculation spray system (RSS) containment sump and in RSS suction lines. 2. Gaps discovered in RSS sump cover plates. 3. Later inspection found other sump enclosure gaps. 4. Bolts and clips missing from the vortex suppression grating 5. Debris found in all 4 RSS pump suction lines.

Plant/report	Problems discussed
H.B. Robinson Unit 2 LER 96–005–00	1. Openings found in sump screens that could allow debris above a certain size to enter the sump. Could have prevented the screens from performing their design function. 2. An item of debris in excess of ⅜″ diameter limit on containment spray nozzles found in 14'' sump drain pipe.
Zion Unit 1 LER 97–001–00	Two 1-inch holes were not in the sump cover as detailed on drawings. Holes allow air to escape as sump fills. Potential to hinder flow to RHR pump suction during a LOCA.
Zion Unit 2 NRC Inspection Report 50–295/96–20 50–304/96–20 March 24, 1997.	1. Miscellaneous debris located throughout containment. 2. Containment recirculation sump screen damage. 3. Peeling and flaking paint on containment surfaces.
Sequoyah Unit 1 10 CFR 50.72 Report 32139 April 11, 1997.	During shutdown on March 22, 1997, an oil cloth was introduced to containment which, if it had come free of its restraints, could have blocked one or both refueling drains so that water in upper containment may not have flowed freely to lower level of containment where sump is located.
Millstone Unit 1 10 CFR 50.72 Report 32161 April 16, 1997.	Most of the coating in the torus is unqualified, which could affect the operability of the low-pressure coolant injection and core spray systems.

Appendix C—Background On Regulatory Basis for Protective Coatings

This appendix discusses the regulatory basis for protective coatings inside the containment. Industry standards and regulatory guidance are included in this discussion. However, this discussion is only for information. Addressees should continue to comply with the plant licensing basis.

At nuclear power plants, coatings and paints serve to (1) protect ferritic steel, austenitic steel, galvanized (zinc-coated) steel, or aluminum surfaces against corrosive environments; (2) protect metallic, concrete, or masonry surfaces against erosion or wear during plant operation; and (3) allow for ease of decontamination of radioactive nuclides from the containment wall and floor surfaces. These coatings may come in inorganic forms, such as zinc-based paints, or organic forms, such as organic latex, polyurethane, or epoxy coatings.

There are two kinds of coatings applications at domestic nuclear power plants:

(1) Class I Service Applications, which are applications of coatings or paints to SSCs that are essential to prevent or mitigate the consequences of postulated accidents. Protective coatings applied to the interior wall and floor surfaces of the containment structure and to the exterior surfaces of most of the SSCs located inside the containment structure normally fall into this category.[1]

(2) Class II Service Applications, which are applications of coatings or paints to SSCs that are essential to the achievement of normal operating performance.

Protective coatings applied to the interior surfaces of the containment structure and to SSCs inside the containment are considered qualified coatings if they have been subjected to physical property (adhesion) tests under conditions that simulate the projected environmental conditions of a postulated design basis (DB) LOCA and have demonstrated the capability of maintaining their adhesive properties under these simulated conditions. These tests are typically conducted in accordance with the guidelines, practices, test methods, and acceptance criteria specified in applicable industry standard procedures (such as those issued by the American National Standards Institute, Inc. [ANSI], or the American Society for Testing and Materials [ASTM]) for coatings applications. However, the licensing basis for Class I coating applications may contain exceptions to or provide alternative means of meeting the intent of the test methods in these standards, provided an adequate safety basis was given to and accepted by the NRC staff as to why accepting the exceptions or alternatives could not have the potential to affect the performance of the ECCS and safety-related CSS during a postulated DB LOCA. In regard to protective coatings used for Class I service applications inside the containment, the staff normally concludes that a coating system is acceptable for service if it has been demonstrated that the coating system is qualified to maintain its integrity during a postulated DB LOCA and if the programs for controlling applications of coating systems for Class I service applications are implemented in accordance with a quality assurance (QA) program that meets the requirements of Appendix B to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR).

Protective coatings that have not been successfully tested in accordance with the provisions in the applicable ANSI or ASTM standards or have not met the acceptance criteria of the standards are considered to be ''unqualified''; that is, they are assumed to be incapable of maintaining their adhesive properties during a postulated DB LOCA. The staff normally assumes that ''unqualified'' coatings applied to the interior surfaces of the containment structure and to SSCs inside the containment structure will form solid debris products under DB LOCA conditions. These debris products should, therefore, be evaluated for their potential to clog ECCS sump screens or strainers and their effect on the operability of safety-related pumps taking suction from ECCS sumps and suppression pools during a postulated DB LOCA.

The NRC has issued Regulatory Guide (RG) 1.54–1973, ''Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants,'' to give the industry an acceptable method for complying with the QA requirements of 10 CFR Part 50, Appendix B, as they relate to protective coating systems applied to ferritic steel, aluminum, stainless steel, zinc-coated (galvanized) steel, or masonry surfaces of water-cooled nuclear power reactors. In RG 1.54–1973, the NRC stated that the guidelines for coating applications in ANSI Standard N101.4–1972, ''Quality Assurance for Protective Coatings Applied to Nuclear Facilities,'' as subject to the additional regulatory positions in RG 1.54–1973, delineate acceptable QA criteria for providing confidence that ''shop or field coating work [will] perform satisfactorily in service.'' The quality assurance provisions stated in ANSI Standard N101.4–1972, as endorsed by the staff in RG 1.54–1973, are considered by the staff to provide an adequate basis for complying with the pertinent QA requirements of 10 CFR Part 50, Appendix B. These standards delineate the type of tests to be performed to qualify a given coating for nuclear applications. However, how a licensee implements its program for controlling activities related to protective coating applications at a particular nuclear plant depends on the plant's licensing basis. Although neither RG 1.54–1973 nor the applicable ANSI standards are NRC requirements, they do delineate acceptable programs and practices for controlling coatings application activities at nuclear power plants.

ANSI Standard N101.4–1972 provides recommended guidelines for implementing QA programs regarding coating applications at domestic nuclear power plants. ANSI Standard N101.4–1972, as endorsed in RG 1.54–1973, delineates recommended guidelines and criteria for establishing QA and quality control programs for coating activities, including activities for controlling work conditions, for controlling the ambient environmental conditions for coating applications, for controlling selection and procurement activities for coatings, for controlling preparation of substrates, for establishing QA procedures for coating applications, for qualifying personnel involved in coating preparation, application, and inspection activities, and for establishing coating inspection guidelines and acceptance criteria. The scope of ANSI Standard N101.4–1972, as endorsed by RG 1.54–1973, also includes recommended QA records on coatings activities.

[1] Coatings applied to non-safety-related small-scale components inside the containment structure, such as small lighting fixtures or small non-safety-related power buses, are an exception to this statement.

ANSI Standard N101.4–1972 states that ANSI Standard N5.9, ''Protective Coatings (Paints) for the Nuclear Industry'' (later reissued as ANSI Standard N512) and ANSI Standard N101.2, ''Protective Coatings (Paints) for Light-Water Nuclear Reactor Containment Facilities,'' are additional acceptable standards for governing activities related to the selection and evaluation of protective coatings applied both in the shop (i.e., at vendor or manufacturer facilities) or in the field.

RG 1.54 is currently undergoing a major revision (it was last revised in 1973). Many of the documents referenced in RG 1.54 are outdated and have been replaced by newer ASTM or ANSI standards. ASTM Committee D–33, ''Coatings for Power Generation Facilities,'' has developed the standards that replace many of the standards referenced in RG 1.54–1973. At the request of the NRC staff, this committee is currently developing a maintenance standard for qualified coatings. This standard will cover inspection of existing coatings, application of new coatings over the original substrate (steel, concrete, galvanized steel, aluminum), new coatings over a substrate-old coating interface, and new coatings over old, qualified coatings. When this standard is approved, RG 1.54–1973 will be revised to reflect current standards. Utilizing more modern industry standards for protective coatings may require a change to the existing licensing basis. Use of these standards must conform with existing NRC requirements, including 10 CFR 50, Appendix B.

Appendix D—Chronology of Incidents and Activities Related to Protective Coatings

In January 1997, Commonwealth Edison Company (ComEd), the licensee for the Zion Nuclear Plant, Unit 2, discovered flaking and unqualified paint applied to the containment surfaces (IN 97–13, ''Deficient Conditions Associated With Protective Coatings At Nuclear Power Plants''). The peeling of the protective coatings was determined to occur at the horizontal junction lines located between the concrete shells that were used in construction of the Zion Unit 2 containment structure. ComEd estimated that the total weight of degraded coatings (peeling paint) was approximately 445 N (100 lb). ComEd also initially estimated that an additional 557–650 m^2 (6000–7000 ft^2) of coatings on surfaces inside containment were not qualified to withstand the environmental conditions of a postulated DB LOCA, in accordance with the testing criteria of ANSI Standard N512–1974. ComEd determined that the peeling of the qualified coatings on the containment surfaces was due to improper surface preparation, resulting in inadequate adhesion of the coating following application.

ComEd corrected the condition of the paint by removing all of the degraded ''qualified'' paint inside the Zion Unit 2 containment and by removing all of the additional ''unqualified'' paints that were determined to be located within the analytically determined zone of influence.[2] ComEd also performed 33 random adhesion or ''pull'' tests on the remaining, intact, ''qualified'' paint inside the containment structure. All of these tests were performed in accordance with the applicable testing requirements specified in ANSI Standard N512–1974. All of the tests exhibited ''pulls'' in excess of the 890 N (200 lb) required by the standard, thus demonstrating that the remaining qualified coatings were acceptable for service during the next operating cycle.

On March 10, 1995, Consolidated Edison Company (ConEd), the licensee for Indian Point Station, Unit 2, reported in LER 95–005–00 that paint was peeling off the floor at the 14-meter (46-ft) elevation of the Indian Point Unit 2 containment structure. The paint was applied to the 14-meter (46-foot) floor elevation during the 1993 refueling outage as an interim measure for reducing personnel radiation exposures until a more permanent floor resurfacing could be accomplished. ConEd determined that the following factors contributed to the cracking and delamination of the paint: (1) in some areas, the paint had been applied in excess of the dry film thickness recommended by the manufacturer of the paint; (2) during preparation of the paint, too much paint thinner was added to the paint, which led to an excessive amount of coating shrinkage when the paint dried; (3) no scarification of the floor surface was performed before application of the paint to remove old coatings, greases, or silicone or wax buildups from the floor surface; and (4) the painters had not been trained to apply the particular brand of paint. ConEd determined the root cause of the coatings event to be the painters' failure to follow controlled procedures for applying the particular brand of paint. To address the nonconforming condition of the paint, ConEd removed all of the old paint from the 14-m (46-foot) floor elevation and repainted the floor elevation with a qualified coating in accordance with the station's procedural requirements and the manufacturer's recommendations for the paint. ConEd also retrained the paint specialists to reindoctrinate them regarding the importance of complying with the station's procedures and standards for coating applications.

On October 18, 1993, the Tennessee Valley Authority (TVA) reported in LER 93–026 the use of unidentified coatings on the surfaces of the No. 4 reactor coolant pump (RCP) motor housings at the Sequoyah Nuclear Plant, Units 1 and 2. These coatings were not accounted for in the licensee's QA Uncontrolled Coatings Log. TVA determined that the No. 4 RCP motor housings are completely within the zones of influence of the containment sumps at both Sequoyah units. The unqualified coating on each No. 4 RCP motor housing amounted to an additional 13.3 m^2 (143 ft^2); this amount was not accounted for by TVA in its 1986 assessment of unqualified coatings on the RCP motor housings. The omission is significant because the maximum amount of uncontrolled coatings allowed by the Uncontrolled Coatings Logs for the Sequoyah units is 5.3 m^2 (56.5 ft^2); this is the maximum amount of uncontrolled coatings that can be in the zone of influence of the containment sump without having the potential to affect the operability of the ECCS and safety-related CSS.

The NRC summarized its review of the safety significance of the amount of unqualified paint on the No. 4 RCP motor housings in Inspection Reports (IR) Nos. 50–327/93–42 and 50–328/93–42 and in IR Nos. 50–327/94–25 and 50–328/94–25, dated November 9, 1993, and September 12, 1994, respectively. In IR Nos. 50–327/94–25 and 50–328/94–25, the NRC concluded that if the unqualified coatings on or within the RCP motor housings failed, they could potentially migrate to the containment sump during a postulated DB LOCA and impair the performance of the containment ECCS and the containment spray system during the event. TVA addressed this issue by modifying the RCP motor housings to include ''catch'' screens designed to prevent coating material on the motor housings from reaching the strainers in the containment sumps.

On July 2, 1993, and September 11, 1995, the Pennsylvania Power and Light Company (PP&L) issued LERs 93–007–00 and 93–007–01, respectively, to summarize its reassessment of ECCS performance at Susquehanna Steam Electric Station, Units 1 and 2, during a postulated DB LOCA. In its initial analysis of ECCS performance during a postulated DB LOCA, PP&L determined that sources of fibrous insulating materials would not have the potential to impair the operability of the ECCS at Susquehanna Units 1 and 2. However, PP&L's initial analysis did not account for ''unqualified'' coatings as potential sources of debris.

In LER 93–007–00, PP&L discussed the effect of debris on the performance of the ECCS during a postulated DB LOCA. In the LER, PP&L stated that its increased awareness of the quantity of unqualified coatings and corrosion products (''other material'') inside the containment was a key factor in deciding to reassess the sources of debris inside the Susquehanna Units 1 and 2 containments during a postulated DB LOCA. PP&L considered fibrous insulation material, unqualified coatings, and corrosion products as the sources of debris. PP&L's evaluation of the debris during the postulated event contained the following uncertainties: (1) uncertainty in qualifying the sources of debris within the containment, (2) uncertainty in determining the amount of debris that could be dislodged during a postulated DB LOCA, and (3) uncertainty in establishing exactly how the debris would be transported from its source to the ECCS strainers during the postulated event. Because of these uncertainties, PP&L stated in the licensee event report that if unqualified coatings and corrosion products were included among the materials that could become sources of debris, some potential existed for complete blockage of the suppression pool strainers during the event.

PP&L addressed this issue, in part, by requiring that DB LOCA qualification testing be performed on all inorganic zinc paints inside the Susquehanna containments. PP&L

[2] All of the unqualified paint within the containment sump's zone of influence was removed, with the exception of approximately 112 ft^2 of unqualified paint applied to small components, such as lighting fixtures or name tags.

also implemented improved administrative housekeeping and inventory controls and issued an administrative coating specification that restricted any coatings applied inside the containment structures to qualified coatings.

Appendix E—Generic Communications Issued by the NRC on the Subject of ECCS and Safety-Related CSS Sump and Strainer Blockage

Generic Letter 85–22,''Potential for Loss of Post LOCA Recirculation Capability Due to Insulation Debris Blockage,'' December 3, 1985.

IN 88–28, ''Potential for Loss of Post LOCA Recirculation Capability Due to Insulation Debris Blockage,'' May 19, 1988.

IN 89–77, ''Debris in Containment Emergency Sumps and Incorrect Screen Configurations,'' November 21, 1989.

IN 92–71, ''Partial Blockage of Suppression Pool Strainers at a Foreign BWR,'' September 30, 1992.

IN 92–85, ''Potential Failures of Emergency Core Cooling Systems by Foreign Material Blockage,'' December 23, 1992.

IN 93–34, ''Potential for Loss of Emergency Core Cooling Function Due to a Combination of Operational and Post LOCA Debris in Containment,'' April 26, 1993.

IN 93–34, Supplement 1, ''Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post LOCA Debris in Containment,'' May 6, 1993.

Bulletin 93–02, ''Debris Plugging of Emergency Core Cooling Suction Strainers,'' May 11, 1993.

NRC Bulletin 93–02, Supplement 1, ''Debris Plugging of Emergency Core Cooling Suction Strainers,'' February 18, 1994.

IN 94–57, ''Debris in Containment and the Residual Heat Removal System,'' August 12, 1994.

IN 95–06, ''Potential Blockage of Safety Related Strainers by Material Brought Inside Containment,'' January 25, 1995.

IN 95–47, ''Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage,'' October 4, 1995.

Bulletin 95–02, ''Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in the Suppression Pool Cooling Mode,'' October 17, 1995.

IN 95–47 Revision 1: ''Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage,'' November 30, 1995.

IN 96–10, ''Potential Blockage by Debris of Safety System Piping Which is Not Used During Normal Operation or Tested During Surveillances,'' February 13, 1996.

Bulletin 96–03, ''Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors,'' May 6, 1996.

IN 96–27, ''Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation,'' May 1, 1996.

IN 96–55, ''Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps Under Design Basis Accident Conditions,'' October 22, 1996.

IN 96–59, ''Potential Degradation of Post LOCA Recirculation Capability as a Result of Debris,'' October 30, 1996

IN 97–13, ''Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants'', March 24, 1997.

Appendix F—Enforcement Actions Taken by the NRC Dealing With Construction and Protective Coatings Deficiencies and Foreign Material Exclusion

Plant	Date of inspection	Severity level/civil penalty	Description
Surry Unit 1	7/30/88	3 $50,000	Debris in containment sump.
Trojan	8/8/89	2 $280,000	Inoperable recirculation sump.
Diablo Canyon	12/8/89	3 $50,000	1. Gaps in sump screens 2. Opening sump access hatches when sump operability is required 3. Debris in sump.
Perry	6/23/93	3 $200,000	Clogged RHR strainers.
Arkansas Nuclear One Unit 1	10/25/93	3 $0	Degradation of containment sump screens.
Browns Ferry Unit 2	5/17/94	4 $0	Unqualified protective coatings applied to safety/relief valve discharge quenchers.
Point Beach Unit 2	10/12/92	3 $75,000	Foreign material in containment spray.
Sequoyah Units 1 and 2	9/3/94	4 $0	Unqualified coatings on RCP motor stand.
Nine Mile Point Unit 2	April 10, 1997 * .	3 ** $200,000	Debris in suppression pool and downcomers.

* Date enforcement action issued.
** Combined with other enforcement actions.

Dated at Rockville, Maryland, this 8th day of May, 1997.

For the Nuclear Regulatory Commission.

Marylee M. Slosson,

Acting Director, Division of Reactor Program Management, Office of Nuclear Reactor Regulation.

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NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of May 12, 19, 26, and June 2, 1997.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of May 12

Monday, May 12

1:30 p.m.
Meeting with Foreign Dignitaries (Closed—Ex.1)

3:00 p.m.
Meeting with Boiling Water Reactor Vessel and Internals Project (BWRVIP) and NRC Staff (Public Meeting)

Tuesday, May 13

2:00 p.m.