UNITED STATES
NUCLEAR REGULATORY COMMISSION
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NRC GENERIC LETTER 98-04: 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 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 degrade the ECCS or safety-related CSS. No action or information is requested regarding these issues. The NRC has issued many previous generic communications on this subject, as discussed later in this generic letter, and assumes that addressees have had adequate prior notice to consider possible actions at their facilities to address these concerns.

The NRC expects addressees to ensure that the ECCS and the safety-related CSS remain capable of performing their intended safety functions. Due to the importance of these systems, the NRC may conduct inspections to ensure compliance.

The NRC is also issuing this generic letter to alert the addressees to the problems associated with the material condition of Service Level 1 (see definitions of Service Levels in Attachment 3) protective coatings inside the containment and to request information under 10 CFR 50.54(f) to evaluate the addressees' programs for ensuring that Service Level 1 protective coatings inside containment do not detach from their substrate during a DB LOCA and interfere with the

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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.

Background

Foreign Material Exclusion, Construction Deficiencies, and Design Deficiencies

In some recent events (discussed in Attachment 1 to this generic letter), 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 found several types of continuing problems.

(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 pressurized- water reactor (PWR) sumps, in boiling-water reactor (BWR) suppression pools and downcomers, and in safety-related pumps and piping.

(2) Deficiencies have been found in the construction of the ECCS sumps and 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 incorrectly sized screens.

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

(4) Design deficiencies have been found, including flow line valves 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 licensee event reports (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 describes the regulatory and safety bases for these concerns.

A more complete list of the above events is provided in Attachment 2. As discussed in Attachment 1, almost all of these events have been the subject of previous NRC generic communications and LERs. Apparently, past NRC generic communications have not been completely effective in focusing licensee attention to the potential areas of concern to control 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 existing plant licensing bases. The NRC intends to take enforcement action for discovered inadequacies consistent with NRC Enforcement Policy.

Protective Coatings

Protective coatings inside nuclear power plant containments serve three general purposes. Protective coatings are applied to carbon and low alloy steel and, less commonly, to aluminum and galvanized surfaces to control corrosion, control radioactive contamination levels, and to protect surfaces from wear. (Although aluminum and galvanized surfaces are not commonly coated, nothing in NRC requirements or industry standards prevent these surfaces from being coated.) A discussion on protective coatings inside the containment and the regulatory requirements and guidance for their use are discussed in Attachment 3.

It has been assumed that 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. The NRC is aware that not all coatings inside the containment are qualified and, therefore, the amount of unqualified coatings must be controlled since the unqualified coatings are assumed to detach from their substrates during a DB LOCA and may be transported to the emergency sump screens or suction strainers. Once in contact with sump screens or suction strainers, coating chips may adversely impact the net positive suction head (NPSH) available to the ECCS or CSS pumps.

Additionally, in some cases, coatings which were qualified failed during normal operation. Some of these events are discussed in Attachment 4.

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. Licensees are required to demonstrate this capability while assuming the most conservative single failure. Some addressees may credit CSSs in the licensing basis for radioactive-source-term and pressure reduction.

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, source-term and pressure reduction features of plant design.

Qualified coatings used inside containment should be capable of withstanding the environmental conditions of a postulated DB LOCA. Although small, localized areas of degraded coatings may not be indicative of widespread failure of the coatings, the condition of the coatings should be evaluated by suitable means. The Electric Power Research Institute (EPRI) has prepared a guidance document for containment coatings titled "Guidelines on the Elements of a Nuclear Safety-Related Coatings Program." Licensees may find EPRI

TR-109937, Final Report, dated April 1998 useful when evaluating coatings, although it has not been endorsed by the staff. The LERs and NRC inspection reports described in Attachment 4 to this generic letter provide evidence of weaknesses in addressee programs with regard to applications of protective coatings for Service Level 1. 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 a coating 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 problems with protective coatings, and 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 Attachment 5. They apply to both PWRs and BWRs. The events discussed in these generic communications, and 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 and the safety-related CSS. 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.

Applicable Regulatory Requirements

The requirements of 10 CFR 50.46(b)(5) and Appendix A to 10 CFR Part 50, Criterion 35, address long term cooling capability and emergency core cooling, respectively. The NRC staff considers that the requirements of 10 CFR Part 50, Appendix B, are germane to this issue for safety related containment coatings.

Section 50.65 of 10 CFR, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," (maintenance rule) 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.

Although this generic letter concerns coatings within the containment and requests information about coatings within containment, addressees should ensure that all coatings which meet the maintenance rule scoping criteria are included in the programs and procedures for implementing the maintenance rule.

The NRC has conducted numerous inspections in the areas addressed by this generic letter; for example, the NRC issued Temporary Instruction (TI) 2515/125, "Foreign Material Exclusion Controls," on August 25, 1994. Violations discovered during the TI 2515/125 inspections 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").

The NRC will continue to conduct inspections in these areas and will consider the long history of generic communications on the issues addressed by this generic letter as prior notice to licensees when assessing 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-specific FME program or preparing a response to the required information, an addressee determines that its facility is not in compliance with the Commission's requirements, the addressee is expected to take appropriate actions in accordance with both requirements stated in Appendix B to 10 CFR Part 50 and the plant technical specifications to restore the facility to compliance.

Licensees are encouraged to work closely with their owners groups and industry associations to coordinate the responses to this letter to improve the efficiency of the responses. The information submitted in response to this generic letter should be considered to be public information.

Required Information

As a result of NRC findings in these areas and due to the importance of ensuring system functionality, within 120 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 or programs implemented to ensure that Service Level 1 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. Maintenance activities involve reworking degraded coatings, removing degraded coatings to sound coatings, correctly preparing the surfaces, applying new coatings, and verifying the quality of the coatings.

(2) Information demonstrating compliance with item (i) or Item (ii):

(i) For plants with licensing-basis requirements for 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, the following information shall be provided to demonstrate compliance:

(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 conservatism 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 Service Level 1 applications inside the containment, discuss how the program adequately qualifies such a coating for Service Level 1 service. Identify which standards or other guidance are currently being used to dedicate containment coatings at your facility; or,

(ii) For plants without the above licensing-basis requirements, information shall be provided to demonstrate compliance with the requirements of 10 CFR 50.46b(5), "Long-term cooling" and the functional capability of the safety-related CSS as set forth in your licensing basis. If a licensee can demonstrate this compliance without quantifying the amount of unqualified coatings, this is acceptable. The following information shall be provided:

(a) If commercial-grade coatings are being used at your facility for Service Level 1 applications, and such coatings are not dedicated or controlled under your Appendix B Quality Assurance Program, 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 pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). This information will enable the Commission to determine whether a 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 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 or additions to systems, structures, and components or to the design or operation of an addressee's facility. It also does not it impose an interpretation of the

Commission's rules that is either new or different from a previous staff position. The staff, therefore, has not performed a backfit analysis.

Reasons for Information Requirement

This generic letter transmits a requirement to submit information 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. The required 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 facilities and whether the regulatory requirements pursuant to 10 CFR 50.46 are being 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 may make the safety-related CSS unable to satisfy the plant-specific licensing basis of controlling containment pressure and radioactivity following a LOCA.

Paperwork Reduction Act Statement

This generic letter mandates information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval number 3150-0011, which expires on September 30, 2000.

The public reporting burden for this collection of information is estimated to average 400 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. The NRC is seeking public comment on the potential impact of the collection of information requested in the generic letter and on the following issues:

(1) Is the proposed collection of information necessary for the proper performance of the functions of the NRC, and will the information have practical utility?

(2) Is the estimate of burden accurate?

(3) Is there a way to enhance the quality, utility, and clarity of the information to be collected?

(4) How can the burden of the collection of information be minimized? Can automated collection techniques be used?

Send comments on any aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, T-6F33, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail at BJS1@NRC.GOV: and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

If you have any questions about this matter, please contact one of the technical contacts or the lead project manager listed below, or the appropriate Office of Nuclear Reactor Regulation project manager.

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Attachments:
1. ECCS sump and strainer events involving foreign material inside the containment and construction and design deficiencies
2. Operational events involving debris in ECCS recirculation flow paths
3. Background on regulatory basis for protective coatings
4. Chronology of incidents and activities related to protective coatings
5. Generic communications issued by the NRC on ECCS and safety-related CSS sump and strainer blockage
6. List of recently issued Generic Letters

ECCS SUMP AND STRAINER EVENTS INVOLVING 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 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 (RHR) 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. The material 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 pumps 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; 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 adversely affected 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 in a pipe in the sump an item of debris larger than the 0.95-cm (3/8-in.) diameter of the holes in the containment spray nozzle.

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 (1-1/2 in.) centrifugal-charging-pump runout-protection manual throttle valves and in the 5.08-cm (2 in.) safety-injection (SI) to cold-leg manual throttle valves were less than the 0.673-cm (0.265 in.) 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 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. However, in response to IN 96-27, 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, had been 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: the "accessibility of the pedestal area downcomers requires removal of grating in the under vessel 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, in LER 89-014-01, the licensee for Diablo Canyon Nuclear Power Plant, Unit 1, discussed the opening of the sump access hatch at various times at power "without adequate consideration of ECCS operability." In LER 96-006, the licensee for Watts Bar Nuclear Plant, Unit 1, reported that an operator observed a containment sump (trash screen) door open while ECCS operability was required.

OPERATIONAL EVENTS INVOLVING DEBRIS IN ECCS RECIRCULATION FLOW PATHS

PLANT/REPORT	PROBLEMS DISCUSSED
Haddam Neck NRC Inspection Report 50-213/96-08	In July 1975, six 55-gallon drums of sludge with varying amounts of debris removed from ECCS sump.
North Anna Units 1/2 LER 84-006-00	Galvanized ductwork painted with unqualified paint.
Millstone Unit 1 LER 88-004-00	Existing suction strainers too small when criteria of RG 1.82 Rev. 1 applied. Strainers will be replaced with larger strainers.
Surry Power Station Units 1/2 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 part or all 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	1. Modification of sump access hatches left holes in top of sump screen cage. 2. Could damage pumps or clog spray nozzles.
McGuire Unit 1 LER 90-0112-00	Loose material discovered in upper containment before entry into Mode 4. Items found would not have made ECCS inoperable.
Calvert Cliffs Units 1/2 NRC Inspection Report	Unit 2 sump found to contain 11.3 kg (25 lb) 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 flow reduced due to blockage in minimum flow recirculation check valve and flow orifice on July 8, 1992. A pump OK. Foreign material also found in the refueling water storage tank.
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 SI operability.
Pt. Beach Unit 2 LER 92-003-01 IN 92-85	September 18, 1992: During Technical Specification inservice inspection 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 a refueling outage, foreign objects discovered in the containment side of the suppression pool. Fouling of RHR strainers found. Strainers not cleaned. January, 1993: RHR A/B strainers found deformed (collapsed inward in the direction of the fluid flow). Strainers replaced. March, 1993: RHR A/B operated in suppression pool cooling mode. Pump suction pressure decreased. Could have compromised long-term RHR operation.
Susquehanna Units 1/2 LER 93-007-00 (voluntary)	1. Assessing impact of debris and corrosion products adhering to fibrous materials that may be dislodged by 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 HPSI 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 15.2 cm x 7.6 cm (6"X3") pipe openings at base of sump curb, the result of a modification before initial operation. 2. Tears in screen. 3. Floor drains leading to sump not screened. 4. Licensee estimated increase in core damage frequency $5X10^{-05}$.
San Onofre Units 1 and 2 LER 93-010-00 (voluntary)	1. Irregular annular gap (approximately 15.2 cm [6"]) surrounding 20.3 cm (8") low temperature overpressure discharge line penetrating horizontal steel cover plate. 2. Engineering analysis concluded both sump trains operable.
Vermont Yankee LER 93-015-00	1. LPCS suction strainers smaller than calculations assumed. NPSH calculations performed in 1986 following change to NUKON™ insulation invalid. 2. Strainers replaced with larger strainers.
South Texas Units 1/2 LER 94-001-00	Sump screen openings from initial construction discovered. Frame plate at floor warped, creating several openings approximately 1.6 cm (5/8"). Additional 0.6 cm (1/4") gaps discovered. Based on ECCS pump tests performed by the manufacturer, the licensee concluded the deficiencies had no safety significance.
Point Beach Unit 1 NRC Inspection Report 50-266/94-06	NRC inspector found grout deterioration under sump screens. Could result in flow bypass, or particles of grout could enter 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, stepoff pad.
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 1/2 NRC Inspection Report 50-390 and 50-391/94-59	Screens installed around RCP 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 the floor on Elevation 46 in containment had lifted and cracked. In other locations, floor coating cracked when stepped on. Licensee concluded that sump function would not be compromised.
Susquehanna Units 1/2 LER 93-007-001 (Voluntary)	Licensee took actions to address concern of clogging ECCS suction strainers. Among these actions: removal of fibrous insulation from HELB areas, testing to determine whether the debris could block the strainer, quantification of corrosion products on structural steel in wetwell, 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 potential debris 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 rendering the A RHR pump inoperable when safety/relief valve lifted on September 11, 1995.
Duane Arnold NRC Inspection Report 50-331/95-003	FME 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	10 locations with screens whose mesh size was inconsistent 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 in MODE 4 and ECCS required to be operable.
Calvert Cliffs Units 1/2 LER 96-003-00	Several holes identified in each unit's containment sump screen larger than described in the FSAR. 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 openings in the 3.8 cm (1-1/2") centrifugal-charging-pump runout-protection manual throttle valves and 5.1 cm (2") SI-to-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 this as a condition which alone could have prevented the fulfillment of a safety function. 2. Five 208 L (55-gallon) drums of sludge removed from ECCS sump. Also, plastic sheeting, nuts, and bolts, tie wraps, and pencils.
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	6 floor drains inside crane wall were not covered with screen that had a finer mesh than the sump screen. 0.6 cm (1/4") holes rather than 0.3 cm (1/8") holes. Crane wall penetrations close to containment floor could allow the transport of debris to the sump screen. Penetrations sealed.
Millstone Unit 2 LER 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	Loose debris in "readily accessible areas" identified by NRC inspectors inside containment. Had the potential to block emergency sump screens during accident conditions. Licensee's evaluation concluded that debris did not represent "substantial challenge" to ECCS. 0.6 m^2 (6 ft^2) of debris estimated. Additional items identified by licensee and NRC inspector during startup while in Mode 3. Further evaluation by licensee concluded that RHR pump would not have had adequate NPSH because of debris.
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. Licensee's preliminary evaluation concluded that operability of ECCS could have been compromised.
LaSalle Unit 2 NRC Event Report 31159 LER 96-009-00	Foreign material recovered from suppression pool and downcomers. This material would challenge the operability of the ECCS. Approximately 0.7 m^2 (7 ft^2) per strainer removed from suppression pool. Material most likely from construction or early outages. Special multiple ECCS pump runs performed with satisfactory results. No apparent transport of the foreign material discovered during this outage.
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.
H.B. Robinson Unit 2 LER 96-005-00	1. Openings found in sump screens. They could have allowed debris above a certain size to enter the sump or prevented the screens from performing their design function. 2. An item of debris in excess of the 1 cm (3/8") diameter containment spray nozzles was found in 36 cm (14") sump drain pipe.
Zion Unit 1 LER 97-001-00	Two 2.5 cm (1") holes detailed on drawings were not in the sump cover. 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	1. Miscellaneous debris found throughout containment. 2. Containment recirculation sump screen damage. 3. Peeling and flaking paint on containment surfaces.

Sequoyah Unit 1 NRC Event Report 32139	During shutdown on March 22, 1997, an oil cloth was introduced into containment. If it had come free, it could have blocked one or both refueling drains so that water in upper containment might not have flowed freely to lower level of containment, where sump is located.
Millstone Unit 1 NRC Event Report 32161	Most of the coating in the torus is unqualified, which could affect the operability of the low-pressure coolant injection and core spray systems.
Clinton NRC Event Report 32633	Significant degradation in protective coatings in the containment wetwell. Some degradation in the drywell. Licensee concluded that the amount of degraded coatings from the containment and the drywell could have exceeded the ECCS strainer loading under accident conditions.
St. Lucie Unit 2 LER 50-389/97-002	Containment sump screens with gaps in screen enclosure contrary to design.
DC Cook Units 1/2 NRC Event Report 32875	A 1 cm (0.25") particulate retention requirement for the containment recirculation sump was not properly established following sump modifications. Inadvertent pathways with openings greater than 1 cm (0.25") were found, including 3 cm (0.75") vents in roof of sump (see following item). Licensee concluded that the ECCS was outside its design basis.
Turkey Point Units 3/4 NRC Event Report 32910	Gaps greater than 1 cm (0.25") found in screens for Unit 3 and 4 sumps.
D.C. Cook Units1/2 NRC Event Report 32948	Enough fibrous material was found in both Unit 1 and Unit 2 containments to potentially cause excessive blockage of the containment recirculation sump screen during the recirculation phase of a LOCA. Both units were already shut down for other reasons. The material was removed from both units before startup.

BACKGROUND ON REGULATORY BASIS FOR PROTECTIVE COATINGS

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

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

ANSI Standards N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," and ANSI N101.4, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," classify coatings as Service Level 1, Service Level 2, or Service Level 3.

Service Level 1 coatings are used in areas where coating failure could adversely affect the operation of post-accident fluid systems and, thereby, impair safe shutdown. With few exceptions, Service Level 1 applies to coatings inside primary containment.

Service Level 2 coatings are used in areas where coating failure could impair, but not prevent, normal operating performance. The function of Service Level 2 coatings is to provide corrosion protection and improve the ability to decontaminate those areas outside primary containment subject to radiation exposure and radionuclide contamination.

A Service Level 3 coating is used on any exposed surface area located outside containment whose failure could adversely affect normal plant operation or orderly and safe plant shutdown.

This generic letter concerns the possible detrimental effects of failed coatings on a plant's ability to recirculate coolant following a LOCA. Therefore, this generic letter is concerned with Service Level 1 coatings.

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 been demonstrated to maintain 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 standards for coatings applications (such as those issued by the American National Standards Institute, Inc. [ANSI], or the American Society for Testing and Materials [ASTM]). However, the licensing basis for Service Level I coating applications may contain exceptions to, or provide alternative means of meeting the intent of, the test methods in these standards. This requires that an adequate safety basis is given to and accepted by the NRC staff as to why accepting the exceptions or alternatives would not affect the performance of the

ECCS and safety-related CSS during a postulated DB LOCA. In regard to protective coatings used for Service Level 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 Service Level I service applications are implemented in accordance with a quality assurance (QA) program that meets the requirements of Appendix B to 10 CFR Part 50.

According to Regulatory Guide (RG) 1.54, 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 and strainers and to affect the operability of safety-related pumps taking suction from ECCS sumps and suppression pools during a postulated DB LOCA.

The NRC issued 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 carbon and low alloy steel, austenitic stainless steel, aluminum, galvanized steel, and 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 supplemented 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. Neither RG 1.54-1973 nor the applicable ANSI standards are NRC requirements: they merely delineate acceptable programs and practices for controlling coating 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. Such programs should control work conditions, the ambient environmental conditions for coating applications, selection and procurement activities for coatings, and preparation of substrate surfaces; establish QA procedures for coating applications; qualify personnel involved in coating preparation, application, and inspection activities; and establish coating inspection guidelines and acceptance criteria. ANSI Standard N101.4-1972, as endorsed by RG 1.54-1973, also recommends keeping certain QA records on coatings activities.

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 governing activities related to the selection and evaluation of protective coatings applied either 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 have replaced 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. Using more up-to-date industry standards for protective coatings may require changing a plant's licensing basis. Use of these standards must conform with existing NRC requirements, including 10 CFR Part 50, Appendix B.

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 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 all of the additional "unqualified" paints that were determined to be located within the analytically determined zone of influence[1]. 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 pull tests exhibited values 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-m (46-ft) elevation of the Indian Point Unit 2 containment structure. The paint was applied to the 14-m (46-ft) 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-ft) floor elevation and

[1] All of the unqualified paint within the containment sump's zone of influence was removed, with the exception of approximately 1 cm^2 (11 ft^2) of unqualified paint applied to small components, such as lighting fixtures or name tags.

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 indoctrinate 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 were 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, to summarize its reassessment of ECCS performance at Susquehanna Steam Electric Station, Units 1 and 2, respectively, 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 could not 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 Unit 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 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.

On April 16, 1997, the licensee for Millstone Nuclear Power Station, Unit 1, a BWR-3 with a Mark I containment, reported to the NRC that a significant amount of coating work inside the Millstone Unit 1 torus (suppression pool) was unqualified. Millstone Unit 1 LER 97-026 stated that a number of different coating materials had been used inside the torus, but the locations and extent of various coating systems were unclear.

On July 15, 1997, the licensee for Clinton Station, a BWR-6 with a Mark III containment, reported to the NRC that a significant quantity of degraded protective coatings was removed from the primary containment and the drywell. The licensee stated that due to the indeterminate condition of these degraded coatings, reasonable assurance could not be given that they would not disbond from their substrates enough to clog the ECCS suction strainers during accident conditions.

GENERIC COMMUNICATIONS ISSUED BY THE NRC ON ECCS AND SAFETY-RELATED CSS SUMP AND STRAINER BLOCKAGE

Generic Letter 85-22, "Potential for Loss of Post Loss of Coolant Accident Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985.

IN 88-28, "Potential for Loss of Post Loss of Coolant Accident 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 Loss of Coolant Accident 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 Loss of Coolant Accident 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 Loss of Coolant Accident 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.

LIST OF RECENTLY ISSUED GENERIC LETTERS

GENERIC LETTER	SUBJECT	DATE OF ISSUANCE	ISSUED TO
98-03	NMSS Licensees' and Certificate Holders' Year 2000 Readiness Programs	06/22/98	All licensees or certificate holders for uranium hexafluoride production plants, uranium enrichment plants, and uranium fuel fabrication plants, except those that have permanently ceased operations
98-02	Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition	05/28/98	All holders of OLS for PWRs, except those who have permanently ceased operations, and have certified that fuel has been permanently removed from the reactor vessel.
98-01	Year 2000 Readiness of of Computer Systems at Nuclear Power Plants	05/12/98	All holders of OLS for nuclear power plants, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel
97-06	Degradation of Steam Generator Internals	12/30/97	All holders of OLS for pressurized-water reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel

OP = Operating License
CP = Construction Permit
NPR = Nuclear Power Reactors

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

If you have any questions about this matter, please contact one of the technical contacts or the lead project manager listed below, or the appropriate Office of Nuclear Reactor Regulation project manager.

Original signed by

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Attachments:
1. ECCS sump and strainer events involving foreign material inside the containment and construction and design deficiencies
2. Operational events involving debris in ECCS recirculation flow paths
3. Background on regulatory basis for protective coatings
4. Chronology of incidents and activities related to protective coatings
5. Generic communications issued by the NRC on ECCS and safety-related CSS sump and strainer blockage
6. List of recently issued Generic Letters

Print out filed in Jacket

*See previous concurrence
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OFFICE	DE:NRR*		OGC*		BC:RECB		(A)D:DRPM	
NAME	JDavis		MRafky		JStolz		JRoe	
DATE	2/12/98		3/30/98		7/14/98		7/14/98	

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The Office of Enforcement reviewed this generic letter and has no objection to it. The Office of the General Counsel reviewed this generic letter and has no legal objection to it.

The staff intends to issue this generic letter 5 working days after the date of this information paper.

L. Joseph Callan
Executive Director
for Operations

Attachments: Proposed NRC Generic Letter: "Potential for Degradation of the Emergency Core Cooling and Containment Spray System Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"

Paul Keene	3/14/98
Tech Editor	Date

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