October 26, 2004

Rejane Spiegelberg Planer
Operational Safety Experience Specialist
IAEA IRS Coordinator
International Atomic Energy Agency
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5, P.O. Box 100
A-1400 Wien
AUTRICHE

Dear Ms. Spiegelberg Planer:

The following operating experience reports from United States reactors are enclosed for your consideration for including in the AIRS database:

NRC Information Notice 2004-15: Dual-Unit Scram at Peach Bottom Units 2 and 3

NRC Information Notice 2004-16: Tube Leakage Due to a Fabrication Flaw in a Replacement Steam Generator

NRC Information Notice 2004-17: Loose Part Detection and Computerized Eddy Current Data Analysis in Steam Generators

NRC Generic Letter 2004-01: Requirements for Steam Generator Tube Inspections

NRC Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors

Each report is being submitted in the following two media: (1) a hard copy of the input file for the AIRS database; and (2) a 3.5-inch HD diskette containing the input file for the AIRS database in WordPerfect format.

If you have any questions regarding these reports, please call Brett Rini of my staff. He can be reached at 301-415-3931.

Sincerely,

/RA/
Francis M. Costello, Acting Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enclosures:
Dr. Pekka T. Pyy
Administrator, Operating Experience & Human Factors
Nuclear Safety Division
Nuclear Energy Agency
OECD
Le Seine St. Germain, Batiment B
12, Boulevard des Iles
92130 - Issy-les-Moulineaux
FRANCE

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Francis M. Costello, Acting Chief
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DATE	10/15/2004	10/14/2004	10/26/2004	10/26/2004

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INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	09/15/2003	DATE RECEIVED

EVENT TITLE

NRC Information Notice 2004-15: Dual-Unit Scram at Peach Bottom, Units 2 and 3

COUNTRY
United States

PLANT AND UNIT
Peach Bottom Units 2 and 3

REACTOR TYPE
BWR

INITIAL STATUS
100%, 91%

RATED POWER (MWe NET)
1116

DESIGNER
General Electric

1st COMMERCIAL OPERATION
07/05/1974, 12/23/1974

ABSTRACT

The U.S. Nuclear Regulatory Commission is issuing this information notice to alert addressees to recent experience in which a dual unit facility lost offsite power, had a dual unit scram, and experienced other problems including the loss of a common emergency diesel generator.

NRC INFORMATION NOTICE 2004-15

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	Reporting Categories:	1.2.6	1.7	1.3.3
2.	Plant Status Prior to the Event:	2.1.1	2.1.2	
3.	Failed/Affected Systems:	3.EF	3.EA	3.AF
4.	Failed/Affected Components:	4.3.1	4.2.3	4.3.5
5.	Cause of the Event:	5.1.7.1	5.3.1	5.1.1.6
6.	Effects on Operation:	6.1.1	6.4	6.5.1
7.	Characteristics of the Incident:	7.9	7.5	7.11.3
8.	Nature of Failure or Error:	8.1	8.1	8.2
9.	Nature of Recovery Actions:	9.1.1	9.2	

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

July 22, 2004

NRC INFORMATION NOTICE 2004-15: DUAL-UNIT SCRAM AT PEACH BOTTOM UNITS 2 AND 3

Addressees:

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

Purpose:

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to recent experience in which a dual unit facility lost offsite power, had a dual unit scram, and experienced other problems including the loss of a common emergency diesel generator (EDG). It is expected that recipients will review this information for applicability to their facilities and consider actions, as appropriate. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On September 15, 2003, offsite power to the emergency buses at Peach Bottom Units 2 and 3 was lost for about 16 seconds when two of the three offsite power sources were briefly lost. All four EDGs automatically started and supplied power to the emergency buses. The third offsite power source remained available to two of the four plant non-emergency plant buses throughout the event.

The offsite power grid dispatcher notified the control room that the portion of the offsite power that was supplying the emergency buses was available half an hour after the event started. However, because the emergency buses were powered from the EDGs and plant transient response actions were the operational priority, operators did not transfer from the EDGs to offsite power for several hours until they were more certain of the reliability of the offsite power source. The licensee determined that the loss of offsite power was the result of a lightning strike approximately 35 miles northeast of the site.

Before the event, Unit 2 was operating at full power and Unit 3 was operating at 91 percent of full power. Both units automatically scrammed when power was lost to the reactor protection system motor generator sets. Containment isolation signals resulted in the closure of the main steam isolation valves and isolation of each reactor from its normal heat sink, the condenser.

ML041950006

All four EDGs automatically started and supplied power to the emergency buses; each EDG supplies power for two buses (one per unit). The licensee was able to safely bring both units into the cold shutdown condition. However, the shutdown of each unit was complicated both by equipment challenges and by procedural problems. The NRC organized an Augmented Inspection Team (AIT) because of the overall risk significance of the event and multiple failures in systems used to mitigate the event. The AIT mission was to determine the causes, conditions, and circumstances relevant to issues directly related to the event and to assess the safety significance of the event (NRC Augmented Inspection Team Report 05000277/2003013 and 05000278/2003013, ADAMS Accession No. ML033530016).

Discussion:

The most significant equipment problem during this event was the unexpected E2 EDG trip during the cooling of the Unit 2 torus while other EDGs were supplying power to the emergency buses. The E2 EDG shut down due to an engine protective trip initiated by low jacket water pressure. The AIT found that combustion gases entered the jacket water coolant system because of one or more leaking cylinder adapter gaskets, causing low jacket water pressure and automatic shutdown of the E2 EDG. The leakage was due to deficient installation procedures and stress relaxation of the cylinder adapter gaskets. These adapter gaskets, made of copper, provide a seal between high-pressure gases in each cylinder and the jacket water system. The licensee concluded that the root cause was inadequate initial pre-loading combined with the natural stress relaxation of the copper over time. The licensee has four Fairbanks Morse 12 cylinder, opposed piston diesel engines for both units.

The AIT found that the EDG cylinder liner replacement procedure did not incorporate adequate guidance to ensure proper sealing of the cylinder liner adapter gaskets. The gaskets relaxed over several years, allowing combustion gases to enter the jacket coolant system. Additionally, the licensee may have missed opportunities associated with jacket water anomalies. Degraded conditions, such as jacket water leaks and high vibration on the E2 EDG from 1996-2002, were tolerated and a condition adverse to quality following two instances of low jacket water pressure was not corrected.

The licensee performed a number of corrective actions to remedy the EDG gasket problem. The licensee replaced all adapter gaskets on the tripped EDG, inspected the cylinders during hydrostatic testing, temporarily installed a sight glass to ensure no combustion gas leakage, revised test and maintenance procedures, and sampled the expansion tank air space and jacket coolant heat exchanger for combustion gases. The final three actions were performed on all the EDGs.

The AIT found that the maintenance procedure for installing the cylinder liner adapter gaskets on the EDGs was deficient and that the licensee took inadequate corrective actions for the low jacket water pressure conditions observed on the E2 EDG in March and April 2003. Using the reactor safety Significance Determination Process (SDP), the AIT determined this incident to be a White finding for Unit 2 (i.e., a low-to-moderate safety-significant finding that may require additional NRC inspection) and a Green finding for Unit 3. The difference in risk significance between the units is due to differences in electrical bus loads.

The Unit 2 transient was complicated by the following factors:

1. Due to the momentary loss of offsite power, the controlling channel of the Unit 2 condenser hotwell level instrumentation failed low. This previously identified equipment deficiency resulted in the draining of the Unit 2 condensate storage tank to the Unit 2 condenser hotwell. The condensate storage tank is the preferred common suction of two Unit 2 mitigating systems, the high-pressure coolant injection system and the reactor core isolation cooling system. As a result of the decreasing condensate storage tank level, the suction of these mitigating systems automatically but unexpectedly changed from the Unit 2 condensate storage tank to the Unit 2 torus.

2. As a result of the transient, the Unit 2 torus water heated up, necessitating the use of residual heat removal (RHR) pumps to cool the Unit 2 torus. At the Peach Bottom site, there are 4 RHR pumps per unit and 4 EDGs common to both units. Therefore, 1 RHR pump from each unit is associated with 1 EDG. A minimum of 1 of the 4 RHR pumps per unit is required to satisfy the containment cooling design function. The licensee needed to use a minimum of 1 RHR pump on both units but was prohibited from simultaneously using pumps powered by the same electrical division, that is, off the same EDG. Thus the Unit 2 A RHR pump and the Unit 3 A RHR pump could not be used at the same time. This is due to electrical load restrictions on the EDG that supplies the same electrical division for both Units 2 and 3. This is a known design limitation of the Peach Bottom station involving the significant electrical load requirements for operating the RHR pump motors.

3. On isolation of the Unit 2 condenser hotwell due to closure of the main steam isolation valves, the associated E2 EDG unexpectedly tripped, stopping the Unit 2 torus cooling. The E2 EDG tripped on low jacket water coolant pressure, which stopped the inservice RHR pump and drained the B torus cooling loop, reducing the availability of torus cooling on Unit 2.

4. A number of other deficiencies complicated operator response and recovery actions.

The Unit 3 transient was complicated by several different factors:

1. The Unit 3 D safety relief valve opened as designed on high reactor pressure but failed to close at the appropriate decreasing reactor pressure setpoint. Over the next 15 minutes, reactor pressure decreased to 369 psig before the valve closed, which allowed injection by condensate pumps and an increase in reactor water level to the high-level setpoint before operators manually tripped these pumps. The valve closed with no operator action. The cause of the initial failure of the valve to close was determined to be pilot valve leakage.

2. The Unit 3 G safety relief valve initially opened automatically on high reactor pressure as designed and was subsequently remotely operated to control reactor pressure. However, on a reactor pressure control operation much later in the event, the valve failed to open on demand from the main control board control switch. The cause of the failure

of the valve to open was determined to be steam leaking through the valve packing into the air operator. The steam damaged the diaphragm of the air operator and prevented the valve from manually operating.

3. The Unit 3 D outboard main steam isolation valve failed to close upon receipt of the Group I isolation signal, remained open for 76 minutes, and then closed with no operator action. The redundant inboard main steam isolation valve appropriately closed as designed.

4. A number of other deficiencies complicated operator response and recovery actions.

This information notice requires no specific action or written response. If you have any questions about information in this notice, please contact one of the technical contacts listed below or the appropriate NRR project manager.

/RA/
William D. Beckner, Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical contacts: Dr. C. Vernon Hodge, NRR
(301) 415-1861
Email: cvh@nrc.gov

Neil Perry, Region I
(610) 337-5225
Email: nsp@nrc.gov

Attachment: List of Recently Issued NRC Information Notice

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
2004-14	Use of less than Optimal Bounding Assumptions in Criticality Safety Analysis at Fuel Cycle Facilities	07/19/2004	All licensees authorized to possess a critical mass of special nuclear material.
2004-13	Registration, Use, and Quality Assurance Requirements for NRC-Certified Transportation Packages	06/30/2004	All materials and decommissioning reactor licensees.
2004-12	Spent Fuel Rod Accountability	06/25/2004	All holders of operating licenses for nuclear power reactors, research and test reactors, decommissioned sites storing spent fuel in a pool, and wet spent fuel storage sites.
2004-11	Cracking in Pressurizer Safety and Relief Nozzles and in Surge Line Nozzle	05/06/2004	All holders of operating licenses or construction permits for nuclear power reactors, except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.
2004-10	Loose Parts in Steam Generators	05/04/2004	All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

Note: NRC generic communications may be received in electronic format shortly after they are issued by subscribing to the NRC listserver as follows:

To subscribe send an e-mail to <listproc@nrc.gov >, no subject, and the following command in the message portion:

subscribe gc-nrr firstname lastname

OL = Operating License
CP = Construction Permit

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	02/19/2004	DATE RECEIVED

EVENT TITLE

NRC Information Notice 2004-16: Tube Leakage Due to a Fabrication Flaw in a Replacement Steam Generator

COUNTRY
US

PLANT AND UNIT
Palo Verde Unit 2

REACTOR TYPE
PWR

INITIAL STATUS
100%

RATED POWER (MWe NET)
1335

DESIGNER
Combustion CE80

1st COMMERCIAL OPERATION
09/19/1986

ABSTRACT

The U.S. Nuclear Regulatory Commission is issuing this information notice to inform addressees about recent operating experience with replacement steam generators. In particular, the potential for tubes to be damaged during fabrication and packaging.

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	Reporting Categories:	1.2.2	1.3.2	1.3.4
2.	Plant Status Prior to the Event:	2.1.1		
3.	Failed/Affected Systems:	3.AH		
4.	Failed/Affected Components:	4.2.6		
5.	Cause of the Event:	5.1.1.2	5.4.17	5.7.2
6.	Effects on Operation:	6.2		
7.	Characteristics of the Incident:	7.2		
8.	Nature of Failure or Error:	8.1		
9.	Nature of Recovery Actions:	9.0		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 3, 2004

NRC INFORMATION NOTICE 2004-16: TUBE LEAKAGE DUE TO A FABRICATION FLAW IN A REPLACEMENT STEAM GENERATOR

Addressees:

All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

Purpose:

The U.S. Nuclear Regulatory Commission is issuing this information notice to inform addressees about recent operating experience with replacement steam generators. In particular, the potential for tubes to be damaged during fabrication and packaging. The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking actions, as appropriate, to avoid similar issues. However, no specific action or written response is required.

Description of Circumstances:

Both steam generators at Palo Verde Nuclear Generating Station (PVNGS), Unit 2, were replaced in December 2003. These new steam generators incorporate many of the design enhancements present in other replacement steam generators (e.g., Alloy 690 tubes, stainless steel tube supports). The tube bundle in the new steam generators is consistent with the original Combustion Engineering design with a horizontal run between the hot and cold leg rather than the more typical U-shape.

When the plant was started up following the replacement of the steam generators in December 2003, the licensee observed a small primary-to-secondary leak measuring approximately 0.6 gallons per day (gpd) (2.3 liters per day (lpd)). Over the following 2 months, the leak rate varied between 0.4 and 0.7 gpd (1.5 and 2.6 lpd) until February 19, 2004, when the leak rate increased from approximately 0.7 to 11 gpd (2.6 to 41 lpd) in a 38-minute timeframe. Although the leak rate did not exceed the technical specification limit, the plant was shut down to identify the source of the leak.

While the plant was shut down, the secondary side of the steam generator was pressurized to 600 pounds per square inch (psi) (4137 kilopascal (kPa)) to assist in the identification of the leaking tube or tubes. During this pressure test, leakage was easily observed coming from a peripheral tube. This tube was subsequently inspected with both a bobbin and a rotating probe. These inspections did not reveal evidence of inservice degradation. These inspections did confirm the presence of a dent near a vertical support in the middle of the horizontal run of the tube that was detected in the preservice examination. This dent signal was considered anomalous because it differed from a typical dent signal in that it exhibited some flawlike characteristics (i.e., it had a vertical component). A comparison of the preservice bobbin and rotating probe inspection data to the data obtained during the outage revealed no significant differences in the dent signal. Although the dent signal was anomalous, there was no distinct indication of material volume loss.

Since the eddy current inspections of the affected tube did not provide conclusive evidence of a through wall flaw, additional testing was performed. This testing included primary and secondary side visual inspections and an in situ pressure test. The visual inspections confirmed the presence of a dent which did not appear to be due to the fabrication of the support structure since the dent was not located directly next to a support strap and did not appear to be the result of impact or leverage. During an in situ pressure test of the entire tube, 0.08 gpm (0.3 lpm) leakage was observed at the differential pressure associated with postulated accident conditions (e.g., a main steam line break), and the tube did not burst at three times the differential pressure associated with normal operating conditions. These tests confirmed the tube had adequate structural integrity. In addition, the leakage from this tube was well below the allowable leakage under postulated accident conditions. Following the in situ pressure test, the leaking tube was plugged and stabilized.

In response to the findings regarding the leaking tube, the rotating probe data for all dent signals obtained during the preservice examination were reviewed to ascertain whether similar anomalous dent signals existed. In addition, rotating probe inspections were performed at dents whose voltages exceeded a specific voltage (e.g., 2 to 5 volts) if these dents had not been inspected with a rotating probe during the preservice inspection. Based on these efforts, one additional tube was identified with an anomalous signal, but was not conclusively similar to the other indication with respect to the vertical presentation of the eddy current signal. This tube was plugged during the preservice examination because the dent obstructed the passage of the normal-sized bobbin probe and there was a concern regarding the future inspectability of this location.

Upon identification of the leaking tube, additional efforts were made to determine the root cause of the leak. These efforts included reviewing steam generator manufacturing records and developing mock-up specimens to simulate the anomalous eddy current signal in the leaking tube.

During the review of the manufacturing records of the steam generator, it was determined that one tube was scrapped during the fabrication of the replacement steam generators since it was damaged (or pierced) by a packing screw. Screws are used in the packing crate in which the tubes are shipped from the tubing manufacturer to the steam generator fabrication facility. The affected portion of this tube was sent back to the tubing manufacturer and corrective actions were taken; however, at the time of the discovery of this damaged tube, all of the tubes in one of the Unit 2 steam generators were installed and the other steam generator was in the process of being fabricated.

To simulate the anomalous dent signal in the leaking tube, a series of dents was fabricated in a mock-up facility. The simulation included impact dents from a nail, a screw, and a drill bit. A wood screw, similar to that used in the tube manufacturer’s crate, was driven through a piece of wood and into the sample tube. Eddy current testing was performed on these specimens and the damage caused by the wood screw yielded a similar anomalous signal to that found in the leaking tube at Palo Verde.

During its formal root cause evaluation, the licensee for PVNGS Unit 2 confirmed that the tube packing crate used wood spacers and cross brace materials that were assembled using common screws as the tubes were loaded into the crate. The design of this packing material placed the screws in close proximity to specific locations on some tubes, and the location, shape, and size of the deformation in the leaking tube are consistent with damage that would occur if a screw penetrated completely through the packing material and came in contact with the tube.

As a result of the findings, the licensee took many corrective actions, including performing inspection of selected tubes, plugging and stabilizing the leaking tube, adding additional quality control inspectors at the steam generator fabrication facility (since replacement steam generators for Unit 1 are being fabricated at this facility), modifying the receipt inspections performed (including procedural changes) on the tubes at the fabrication facility, evaluating/ modifying the packing procedure/design, identifying the tubes that were shipped in package locations where packing screw damage was possible, and initiating additional mock-up testing to improve the capability to identify and characterize volumetric flaws located within a dent (e.g., puncture-type defects).

After concluding that there was reasonable assurance of tube integrity, the licensee returned the plant to service. The primary-to-secondary leak rate following startup was near the detection threshold (i.e., less than 0.1 gpd (0.4 lpd)). In addition, following plant startup, six additional tubes were found at the fabrication facility during the unpacking of tubes for the Palo Verde Unit 1 replacement steam generators that had been pierced by a packing crate screw. These tubes were not installed in any of the steam generators being fabricated.

Discussion:

Steam generators have been replaced at many U.S. plants, and a number of other plants plan to replace in the next several years.

The finding of tubes damaged during the fabrication of the Palo Verde replacement steam generators illustrates the importance of monitoring the fabrication process. This includes the packing procedures for the tubes and the receipt inspections performed on these tubes once they arrive at the steam generator fabrication facility.

In addition, the findings at Palo Verde illustrate the importance of fully evaluating the implications of all abnormal conditions (i.e., conditions adverse to quality) identified during the fabrication process and communicating these results to all affected parties within an organization. In this instance, the personnel performing the preservice inspection at Palo Verde were not specifically notified of the identification of a tube that been damaged by a packing screw. As a result, they did not consider the potential for this type of flaw to exist in their review of the inspection data. By communicating non-conforming conditions observed during fabrication to the individuals responsible for the preservice examination, nondestructive examination techniques can be selected and the personnel trained to ensure potential defects are reliably detected and evaluated.

The findings at Palo Verde also illustrate the inspection challenges in finding flaws (such as from a screw) when they are located within a dent. These inspection challenges include determining the appropriate voltage threshold at which rotating probe examinations should be performed on a dent to detect flaws and determining the capability of the rotating probe to reliably identify flaws (e.g., holes) in a dent.

Lastly, the findings at Palo Verde indicate that the source of small amounts of primary-to-secondary leakage from volumetric defects can be determined through secondary side pressure tests.

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate project manager in the NRC's Office of Nuclear Reactor Regulation (NRR).

/RA/
Terrence Reis, Acting Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contacts: Charles Paulk, Region IV
817-860-8236
E-mail: cjp@nrc.gov

Kenneth Karwoski, NRR
301-415-2752
E-mail: kjk1@nrc.gov

Attachment: List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
2004-15	Dual-Unit Scram at Peach Bottom Units 2 and 3	07/22/2004	All holders of operating licenses for nuclear power reactors except those who have permanently ceased operation and have certified that fuel has been permanently removed from the reactor vessel.
2004-14	Use of less than Optimal Bounding Assumptions in Criticality Safety Analysis at Fuel Cycle Facilities	07/19/2004	All licensees authorized to possess a critical mass of special nuclear material.
2004-13	Registration, Use, and Quality Assurance Requirements for NRC-Certified Transportation Packages	06/30/2004	All materials and decommissioning reactor licensees.
2004-12	Spent Fuel Rod Accountability	06/25/2004	All holders of operating licenses for nuclear power reactors, research and test reactors, decommissioned sites storing spent fuel in a pool, and wet spent fuel storage sites.
2004-11	Cracking in Pressurizer Safety and Relief Nozzles and in Surge Line Nozzle	05/06/2004	All holders of operating licenses or construction permits for nuclear power reactors, except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

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OL = Operating License
CP = Construction Permit

INCIDENT REPORTING SYSTEM

IRS NO. **EVENT DATE** **2004** **DATE RECEIVED**

EVENT TITLE

NRC Information Notice 2004-17: Loose Part Detection and Computerized Eddy Current Data Analysis in Steam Generators

COUNTRY
US

PLANT AND UNIT
Shearon Harris

REACTOR TYPE
PWR

INITIAL STATUS
Shutdown

RATED POWER (MWe NET)
900

DESIGNER
Westinghouse 3 loop

1st COMMERCIAL OPERATION
05/02/1987

ABSTRACT

The U.S. Nuclear Regulatory Commission is issuing this information notice to inform addressees about recent operating experience with (1) challenges associated with detection of loose parts and related tube damage in steam generators and (2) computerized data screening algorithms used in the evaluation of steam generator tube eddy current data.

NRC INFORMATION NOTICE 2004-17

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	**1.2.2**	**1.4**	
2.	**Plant Status Prior to the Event:**	**2.0**		
3.	**Failed/Affected Systems:**	**3.AH**		
4.	**Failed/Affected Components:**	**4.2.6**		
5.	**Cause of the Event:**	**5.1.1.8**	**5.4.17**	
6.	**Effects on Operation:**	**6.0**		
7.	**Characteristics of the Incident:**	**7.2**		
8.	**Nature of Failure or Error:**	**8.2.2**		
9.	**Nature of Recovery Actions:**	**9.0**		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 25, 2004

NRC INFORMATION NOTICE 2004-17: LOOSE PART DETECTION AND COMPUTERIZED EDDY CURRENT DATA ANALYSIS IN STEAM GENERATORS

Addressees:

All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

Purpose:

The U.S. Nuclear Regulatory Commission is issuing this information notice to inform addressees about recent operating experience with (1) challenges associated with detection of loose parts and related tube damage in steam generators and (2) computerized data screening algorithms used in the evaluation of steam generator tube eddy current data. The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking actions, as appropriate, to avoid similar issues. However, no specific action or written response is required.

Description of Circumstances:

In 2001, at the Shearon Harris nuclear power plant, the licensee, Progress Energy, installed replacement recirculating steam generators containing thermally treated Alloy 690 tubing. The licensee conducted a pre-service eddy current inspection before the steam generators were placed in service and, in 2003, conducted the first inservice eddy current inspection of all active tubes. No tube degradation was detected. In late April 2004, however, the licensee detected minor primary-to-secondary leakage in the C steam generator. In May 2004, following a unit trip for an unrelated reason, the licensee investigated the source of the primary-to-secondary leak. In the days preceding the unit trip, leak rates varied between approximately 20 to 40 liters per day (5 to 10 gallons per day). A secondary side pressure test, an eddy current inspection and a visual foreign object search were performed to identify the source of the leakage. This investigation identified the leaking tube and damage in two adjacent tubes above the cold leg tubesheet. A foreign object search and retrieval (FOSAR) examination identified a loose part in the C steam generator at the damage location. This part was removed during the FOSAR examination. The metallic object was magnetic and approximately 57 mm (2.25 inches) long and had an irregular shape with some sharp edges.

ML042180094

Bobbin coil eddy current inspection performed during the May 2004 outage resulted in the identification of loose part wear in two of the three affected tubes, neither of which was the leaky tube. Even with the information from the secondary side pressure test, standard bobbin coil analysis techniques did not readily detect the damage in the tube with the primary-to-secondary leak. This indication, however, was apparent when a 3-frequency ("turbo") mix bobbin coil channel was used. Damage was also readily detected with the +Point™ coil. This damage was located about 4 mm (0.16 inch) above the tubesheet and was masked, when using standard bobbin coil analysis techniques, by signals from the nearby expansion transition located at the top of the cold leg tubesheet. The loose part wear that was detected in the other two tubes was located at about 12 mm (0.48 inch) and 20 mm (0.77 inch) above the cold leg tubesheet. A +Point™ coil measurement indicated the depth of loose part wear damage in the three tubes at 77 percent through-wall (the leaking tube), 80 percent through-wall, and 45 percent through-wall. This examination did not detect a loose part since the part had been removed prior to the +Point™ examination. Subsequent in situ pressure testing of the leaking tube and the tube with 80 percent through-wall wear (as measured by +Point™) demonstrated adequate structural integrity and indicated adequate leakage integrity.

After evaluating the 2004 inspection test results, the licensee reviewed the 2003 steam generator bobbin coil eddy current inspection results in the vicinity of the loose part location. Two of the three tubes with loose part wear, including the tube that leaked, had no detectable degradation and no signals indicative of a loose part in the 2003 data. However, a bobbin coil indication, estimated at 37 percent through-wall, was present in the third tube but was missed by both primary (manual) and secondary (computer) analysis. Since this tube was not inspected with a rotating probe in 2003, this depth estimate was obtained using bobbin coil phase angle analysis. This indication was estimated as 66 percent through-wall in 2004, using the same bobbin coil technique analysis.

Investigation into why the computerized data screening (CDS) algorithm used for secondary analysis did not detect this indication in 2003 revealed that some improper values were used in the CDS settings. Automated data analysis was facilitated by dividing the tube lengths into five regions. When the inspection parameters were entered into the CDS system for these regions, a 13-mm (0.5-inch) gap in the tube analysis (from 13 mm [0.5 inch] above the tubesheet to 25 mm [1 inch] above the tubesheet) was inadvertently created above the tubesheet. These input values caused the computerized tube analysis process to skip the portion of the tube containing the 37-percent through-wall bobbin indication in the 2003 analysis. The licensee properly adjusted the CDS settings and re-analyzed the portion skipped during the 2003 analysis. Other than the one wear indication discussed above, the licensee detected no other indications during the reanalysis of the 2003 data.

In addition to the eddy current data review from the 2003 outage, the licensee also reviewed secondary side steam generator FOSAR tapes from that outage. Though evident on the tape, the loose part was not identified in 2003. Comparison of the tapes from the 2003 and 2004 outages at the loose part location indicates that the loose part moved further into the tube bundle after the 2003 outage.

Discussion:

The staff recently issued Information Notice (IN) 2004-10, “Loose Parts In Steam Generators,” to discuss the potential for loose part degradation to affect steam generator tube integrity. IN 2004-10 discussed the importance of supplementing the steam generator tube eddy current examinations with secondary side visual inspections. A secondary side visual inspection was performed at Shearon Harris during the 2003 refueling but the presence of the loose part was not detected. If the part had been detected and removed, it would have prevented continued tube wear and the subsequent leak. In addition, the visual detection of the part would have given the licensee an opportunity to detect the associated tube degradation and possibly notice the inappropriate settings for the CDS software.

IN 2004-10 also discussed the possibility that tube damage from loose parts may not always be identified due to the presence of interfering signals. The recent operating experience at Shearon Harris confirms this statement. Interfering signals from the nearby tube expansion transition posed an inspection challenge for the bobbin coil (i.e., the interfering signals masked the flaw signal associated with the leaking tube). Although this particular indication was detected in hindsight analysis by the application of a 3-frequency bobbin coil mix, it is not known if this analysis technique would detect loose part damage in different circumstances. Inspection with the +Point™ coil clearly identified damage in all three affected tubes.

In addition to reinforcing some of the information communicated in IN 2004-10, the recent experience at Shearon Harris shows the importance of properly setting automated data screening parameters. Improper settings caused a small portion of tubing not to receive the secondary analysis of eddy current data. This analysis would have detected loose part wear in one tube during the previous outage inspection before the primary-to-secondary leak occurred. This shows the importance of verifying proper automated data screening parameters used for either primary or secondary analysis. Improper setting of these parameters may not be readily apparent during data analysis. In this particular instance, the eddy current computer screening system completed the site-specific performance demonstration, but the demonstration data did not contain indications in the 0.5-inch length of tube that was not analyzed by the computer. Also, proper setting of the computerized screening system parameters is necessary to ensure the analysis of eddy current data by two independent analysis teams.

Recent steam generator tube leaks at a number of plants with thermally treated Alloy 600 or Alloy 690 tubing (i.e., H.B. Robinson, Palo Verde, Shearon Harris) illustrate the need for thorough inspections and robust inservice inspection programs alert to the potential for tube degradation regardless of tube material, location, or steam generator history. Overall, experience with corrosion-related degradation mechanisms in these plants has been favorable. Nevertheless, this experience indicates that damage by loose parts or damage incurred during manufacture of steam generator tubes can result in primary-to-secondary system leakage. These experiences also show the importance of being alert to all potential tube degradation mechanisms and to aggressively interrogate eddy current inspection signals that may be associated with tube degradation.

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate project manager in the NRC's Office of Nuclear Reactor Regulation (NRR).

/RA/
Francis M. Costello, Acting Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contact: Paul Klein, NRR
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Attachment: List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
2004-16	Tube Leakage Due to a Fabrication Flaw in a Replacement Steam Generator	08/03/2004	All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.
2004-15	Dual-Unit Scram at Peach Bottom Units 2 and 3	07/22/2004	All holders of operating licenses for nuclear power reactors except those who have permanently ceased operation and have certified that fuel has been permanently removed from the reactor vessel.
2004-14	Use of less than Optimal Bounding Assumptions in Criticality Safety Analysis at Fuel Cycle Facilities	07/19/2004	All licensees authorized to possess a critical mass of special nuclear material.
2004-13	Registration, Use, and Quality Assurance Requirements for NRC-Certified Transportation Packages	06/30/2004	All materials and decommissioning reactor licensees.
2004-12	Spent Fuel Rod Accountability	06/25/2004	All holders of operating licenses for nuclear power reactors, research and test reactors, decommissioned sites storing spent fuel in a pool, and wet spent fuel storage sites.

Note: NRC generic communications may be received in electronic format shortly after they are issued by subscribing to the NRC listserver as follows:

To subscribe send an e-mail to <listproc@nrc.gov >, no subject, and the following command in the message portion:

subscribe gc-nrr firstname lastname

OL = Operating License
CP = Construction Permit

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	N/A	DATE RECEIVED

EVENT TITLE

NRC Generic Letter 2004-01: Requirements for Steam Generator Tube Inspections

COUNTRY	PLANT AND UNIT	REACTOR TYPE
US	All PWRs	PWR
INITIAL STATUS	**RATED POWER (MWe NET)**	
N/A	N/A	
DESIGNER	**1st COMMERCIAL OPERATION**	
N/A	N/A	

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to advise addressees that the NRC's interpretation of the technical specification requirements in conjunction with 10 CFR Part 50, Appendix B, raises questions as to whether certain licensee steam generator tube inspection practices ensure compliance with these requirements.

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	Reporting Categories:	1.2.2	1.4	
2.	Plant Status Prior to the Event:	2.0		
3.	Failed/Affected Systems:	3.AH		
4.	Failed/Affected Components:	4.2.6		
5.	Cause of the Event:	5.1.1.1	5.1.1.2	5.4.17
6.	Effects on Operation:	6.0		
7.	Characteristics of the Incident:	7.2		
8.	Nature of Failure or Error:	8.2		
9.	Nature of Recovery Actions:	9.0		

OMB Control No.: 3150-0011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

August 30, 2004

NRC GENERIC LETTER 2004-01: REQUIREMENTS FOR STEAM GENERATOR TUBE INSPECTIONS

Addressees

All holders of operating licenses for pressurized-water reactors (PWRs), 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 to

(3) advise addressees that the NRC's interpretation of the technical specification (TS) requirements in conjunction with 10 CFR Part 50, Appendix B, raises questions as to whether certain licensee steam generator (SG) tube inspection practices ensure compliance with these requirements,

(4) request that addressees submit a description of the tube inspections performed at their plants, including an assessment of whether these inspections ensure compliance with the TS requirements in conjunction with 10 CFR Part 50, Appendix B,

(5) request that addressees who conclude they are not in compliance with the SG tube inspection requirements contained in their TS in conjunction with 10 CFR Part 50, Appendix B, propose plans for coming into compliance with these requirements, and

(6) request addressees to submit a tube structural and leakage integrity safety assessment that addresses any differences between their practices and the NRC's position regarding the requirements of the TS in conjunction with 10 CFR Part 50, Appendix B. A safety assessment should be submitted for all areas of the tube required to be inspected by the TS where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used. This assessment should include an evaluation of whether the inspection practices rely on an acceptance standard different from the TS acceptance standards and whether the technical basis for these inspection practices constitutes a change to the "method of evaluation" (as defined in 10 CFR 50.59) for establishing the structural and leakage integrity of the tube-to-tubesheet joint.

Pursuant to 10 CFR 50.54(f), addressees are required to submit a written response to this generic letter.

ML042370766

Background

Steam generator tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this generic letter, tube integrity means that the tubes are capable of performing these functions in accordance with the plant design basis.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the SG tubing. Specifically, the general design criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB shall be "designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture" (GDC 14) and "designed, fabricated, erected, and tested to the highest quality standards practical" (GDC 30) and that RCPB components shall be "designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity" (GDC 32). For plants that were issued construction permits before the effective date of 10 CFR Part 50, Appendix A, the plant-specific principal design criteria (PDC) in the plant design basis established similar fundamental regulatory requirements pertaining to the integrity of the steam generator tubing.

Given the importance of SG tube integrity, all current PWR licensees have TS governing the surveillance of SG tubes. These TS typically do not prescribe nondestructive test methods for inspecting tubes or specify where a particular methodology should be used. For example, current TS may employ the following or similar general language:

> Tube inspection for tubes selected in accordance with Table [xxxx] means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg, excluding sleeved areas.

Although the TS do not prescribe the use of particular nondestructive test methods, the NRC position in Regulatory Guide 1.83 is that "...the equipment should be capable of locating and identifying defects due to stress corrosion cracking and due to tube wall thinning by mechanical damage, chemical wastage, or other causes." In addition, the TS surveillance requirements specify acceptance limits for SG tubes (often called plugging or repair limits) to be applied to the inspection results. The surveillance requirements seek to ensure that enough information is obtained about imperfections (e.g., flaws) in the tubes to determine if TS plugging limits are being met. Tube imperfections are defined in the TS and include circumferential and axial cracks.

SG tubes are also subject to the quality assurance requirements of 10 CFR Part 50, Appendix B. Specifically, SG tubes are safety-related components and, therefore, subject to the criteria of Appendix B. Notwithstanding that the TS do not specify nondestructive test methods or in what locations particular test methods must be employed, Criterion IX of 10 CFR Part 50, Appendix B, "Control of Special Processes," requires, in part, that nondestructive testing be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. In addition, Criterion XI, “Test Control,” requires, in part, that test procedures shall include provisions for assuring that all prerequisites for the given test have been met and that adequate test instrumentation is available and used. Moreover, Criterion XVI, “Corrective Action,” requires, in part, that “measures shall be established to assure that conditions adverse to quality ... are promptly identified and corrected.” This generic letter addresses the selection of appropriate inspection techniques for tube locations required to be inspected by plant TS, given the tubes selected for inspection based on plant TS sampling requirements, or for tube locations where licensees have reason to believe a condition adverse to quality may be present.

Licensees currently employ an eddy current test bobbin probe, at least, to inspect the entire length of tubing required by the TS. The bobbin probe is a high-speed probe which the industry has demonstrated is capable of reliably detecting volumetric flaws and axially oriented cracks in the absence of significant masking signals. Masking signals may be produced by tube geometry variations or irregularities along the tube axis (such as small-radius U-bends, dents and dings, and expansion transitions) or by tube surface irregularities. Masking signals can also be produced by deposits on the tube surface, adjacent support structures (such as the tubesheet), probe wobble, cold working, permeability variations, or electrical noise.

While the bobbin probe generally provides an effective means of SG tube inspection over much of the tube length, experience has shown that the bobbin probe may not be effective at locations where significant masking signals are present. In addition, the bobbin probe generally cannot detect circumferential cracks, which has been documented in previous NRC communications (e.g., Information Notices 90-49 and 94-88, and Generic Letter 95-03). Circumferential cracks can occur at locations of high axial stress (e.g., small-radius U-bends and the tubesheet expansion region).

Plant TS for virtually all PWRs require inspection of the entire length of the hot leg tube within the tubesheet. With some exceptions where specified by the plant TS, the acceptance limits (plugging limits) for these inspections apply to all imperfections along the full length of the tube in the tubesheet on the hot leg side, including axial and circumferential cracks. To the staff’s knowledge, however, the bobbin probe has not been demonstrated to be capable of reliably detecting axial or circumferential flaws in the expanded region of tubing inside the tubesheet. Specialized probes are available which have been demonstrated to be capable of detecting such flaws for this application.

Given the limitations of the bobbin probe, industry practice is to supplement the bobbin probe inspection with inspections by specialized probes, such as the rotating pancake coil or +Point™ probe. However, inspecting tubes with these specialized probes is slower than with the bobbin probe. Therefore, these slow-speed probes are typically not applied over the entire length of a tube that is subject to inspection, but only at tube locations where degradation which cannot be reliably detected with the bobbin probe (e.g., circumferential cracks, axial cracks in low-row U-bends and expansion transitions) is known to be present or considered to have a potential to occur. The practice of selecting the type of probe to be used at specific locations along the length of tube involves engineering analysis (termed “degradation assessment” in industry guidelines), which may include an element of judgment, to determine the potential for degradation to occur at various locations.

In 2002, the staff learned that several licensees were not fully implementing inspection methods capable of detecting circumferentially oriented cracks at all locations where the potential for such cracks exists and where, based on available evidence, there is reason to believe such cracks may be present. These licensees were performing full-length bobbin probe inspections of the tubes and were performing additional inspections using specialized probes to inspect for axial and circumferential cracks at certain locations, including the tube expansion transitions near the top of the tubesheet. The licensees conducted the specialized probe inspections at the tube expansion transitions in an area that extended from 2 inches above the top of the tubesheet to about 5 inches below the top of the tubesheet. At several facilities, circumferential cracks were identified at tube expansion transitions, as well as below the transitions near the bottom of the zone being inspected. These results indicate a potential for circumferential cracks to exist in the tubing below the zone inspected with the specialized probe. However, each licensee also performed an analysis indicating that circumferential cracks below the zone being inspected with the specialized probe would not be detrimental to tube structural and leakage integrity. These licensees concluded, therefore, that additional inspections for circumferential cracks with the specialized probe were unnecessary. These analyses had not been provided to the NRC staff.

The staff became aware of these activities during SG inspections conducted during refueling outages and asked these licensees to submit TS amendment requests or safety analyses to obtain NRC approval of their inspection approaches. The staff reviewed the resulting submittals on a one-cycle basis before the plants restarted. Subsequent to these plant-specific actions, the staff evaluated the appropriate method to interact with licensees on this issue. Given new inspection information indicating that circumferential cracks were occurring in tubes below the expansion transition region, and the potentially generic nature of the issue, the staff decided to communicate the issue to licensees through this generic letter.

Discussion

As part of the inspection process, licensees perform an engineering (degradation) assessment to determine the potential for degradation at specific locations of the tube. The staff recognizes that the potential for degradation may vary from plant to plant based on tube material, operating hours, and other plant-specific factors. However, once licensees have determined what degradation may be present at various locations along the length of the tube, it is the staff position that they should use probes capable of detecting these forms of degradation. Not to do so raises questions about whether the tube inspection practices ensure compliance with the TS in conjunction with 10 CFR Part 50, Appendix B. This staff position is consistent with the position expressed in Section 2.a of Regulatory Guide 1.83, Revision 1, issued in 1975.

In the aforementioned cases, tube inspections with a specialized probe near the top of the tubesheet clearly indicated the potential for circumferential cracks to occur deeper into the tubesheet, beyond the region inspected with the specialized probes. In each case the licensee was aware of the potential for such cracks to exist deeper into the tubesheet, but the licensee did not employ techniques capable of reliably detecting such cracks because the licensee’s analysis concluded that such cracks did not have safety implications.

In addition, the staff notes that not inspecting with techniques that are capable of detecting flaws of any type that may be present would allow any such flaws to remain in place. However, most plant TS state that only tubes with imperfections less than 40 percent of the nominal tube wall thickness are acceptable for continued service (there are exceptions specified in some plant TS). Therefore, if licensees do not use probes capable of detecting flaws that may potentially be present, licensees would be allowing flaws to remain inservice which may exceed the applicable TS acceptance criteria (i.e., tube repair or plugging limit). The staff notes that the acceptance or plugging limit for SG tube inspections is a specific TS limit that can only be changed through the license amendment process. Furthermore, even when a probe is capable of finding flaws potentially present, flaws may be inadvertently missed for a variety of reasons (e.g., the flaw size is below the threshold of detection). However, missing a flaw is different than using a probe which is not capable of detecting the forms of degradation that may be present. In other words, the objective of the inspection is to detect flaws of any type that may have the potential to be present along the length of the tube required to be inspected and that may meet or exceed the applicable tube repair criteria.

The staff acknowledges that there may be circumstances in which certain flaws at certain locations may not impair tube integrity even if the TS plugging limit is exceeded. In such circumstances, the staff has reviewed and approved TS amendment requests for alternative tube repair criteria (ARCs) applicable to specified flaw types and/or locations. Some of these ARCs have included special inspection requirements defining the method of inspection to be used when implementing the ARCs. It is the staff's position that if there are locations where certain flaw types can be allowed to exceed existing TS plugging limits, the TS need to be amended to allow the practice. In general, the amendment could include provisions for an ARC and sometimes accompanying special inspection requirements, consistent with past licensing practice. Alternatively, in the case of the aforementioned tubesheet inspection issue, such an amendment could simply clarify the extent of the tube to be inspected within the thickness of the tubesheet, if there is a supporting technical basis that flaws at locations not to be inspected will not impair tube integrity irrespective of the size of the flaws. Pending the submission of such amendment requests, it is the staff's position that licensees are required under existing requirements (TS in conjunction with 10 CFR Part 50, Appendix B) to employ inspection techniques capable of detecting all flaw types which may be present at locations which are required to be inspected pursuant to the TS.

Although this specific example involves inspections in the tubesheet region at plants where cracking had the potential to occur, similar situations could exist at other tube locations for certain degradation mechanisms. As a result, the staff's position applies to all tube locations. In addition, it applies to all PWRs since tube degradation can occur in any steam generator and similar situations could exist at any plant.

Also, for the instances cited above, the safety basis developed by the licensees for not expanding the scope of the specialized probe inspection beyond a specific distance (x inches) into the tubesheet was that any cracks below that distance were not detrimental to tube integrity. This was based on analyses indicating that tubes only needed a minimum embedment of x inches into the tubesheet to exhibit acceptable structural and accident leakage integrity. The staff notes that this is a different acceptance standard than the TS acceptance standards (i.e., plugging limits or tube repair criteria) that have been reviewed and approved by the NRC staff. If the licensee is utilizing a less restrictive acceptance standard than to the standards in the technical specifications, a license amendment will be needed in order to implement such a standard.

Furthermore, these analyses have been performed to demonstrate that cracks below this embedment distance do not impair SG tube integrity even if these cracks cause complete severance of the tube. According to many plant final safety analysis reports (FSARs), the SGs were designed in accordance with Section III of the American Society of Mechanical Engineers (ASME) Code. In accordance with Section III of the Code, the original design basis pressure boundary for the tube-to-tubesheet joint included the tube and tubesheet extending down to and including the tube-to-tubesheet weld. The criteria of Section III of the ASME Code constitute the "method of evaluation" for the design basis. These criteria provide a sufficient basis for evaluating the structural and leakage integrity of the original design basis joint. However, the criteria of Section III do not provide a sufficient basis by themselves for evaluating the structural and leakage integrity of a mechanical expansion joint consisting of a tube expanded against the tubesheet over some minimum embedment distance. If a licensee is redefining the design basis pressure boundary and is using a different method of evaluation to demonstrate the structural and leakage integrity of the revised pressure boundary, an analysis under 10 CFR 50.59 would determine whether a license amendment is required.

In summary, for the cases discussed above, the TS required a tube inspection for the full length of the tube within the tubesheet (scope), and the findings from this inspection were required to be evaluated against a repair (plugging) criterion. Neither the scope nor the repair criteria in the TS contained provisions for limiting the inspections through a licensee-approved process.

For the cases cited above, the NRC cannot conclude that the licensees are in compliance with their TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B, with regard to the inspections they are performing. This concern stems, in part, from experience. Some licensees have relied on licensee-controlled analyses to justify not inspecting for degradation in areas where it had the potential to exist. By not inspecting such areas, the licensees have allowed flaws that may have been detected and that may exceed the repair or plugging limit to remain in service. These inspection practices are contrary to the requirements in the TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B, which require the identification of conditions adverse to quality by using qualified techniques and adequate test instrumentation and do not provide for limiting SG tube inspections in the manner described above. In addition, this practice appears contrary to the consistent past practice of amending the TS in cases where existing TS plugging limits are determined to be overly conservative for certain flaw types at certain locations. It is the staff's position that pending a license amendment clarifying the inspection approach to be followed, licensees are required to employ inspection methods capable of detecting all flaw types that may be present at locations that are required to be inspected by the TS and where flaws at those locations may exceed the applicable TS tube repair criteria.

Based on these staff concerns, the NRC is issuing this generic letter, consistent with the requirements in 10 CFR 50.54(f), to obtain information necessary for the staff to determine if addressees are in compliance with the TS in conjunction with 10 CFR Part 50, Appendix B. In addition, licensees who have not been implementing inspections consistent with the staff's position should submit a safety assessment that demonstrates their ability to ensure continued safe operation and addresses any differences between their practices and those called for by the staff's position. Safety assessments should be submitted to the NRC for all areas of the tube required to be inspected by the TS where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used.

Requested Information

Within 60 days of the date of this generic letter, addressees are requested to provide the following information to the NRC:

(1) Addressees should provide a description of the SG tube inspections performed at their plant during the last inspection. In addition, if they are not using SG tube inspection methods whose capabilities are consistent with the NRC's position, addressees should provide an assessment of how the tube inspections performed at their plant meet the inspection requirements of the TS in conjunction with Criteria IX and XI of 10 CFR Part 50, Appendix B, and corrective action taken in accordance with Appendix B, Criterion XVI. This assessment should also address whether the tube inspection practices are capable of detecting flaws of any type that may potentially be present along the length of the tube required to be inspected and that may exceed the applicable tube repair criteria.

(2) If addressees conclude that full compliance with the TS in conjunction with Criteria IX, XI and XVI of 10 CFR Part 50, Appendix B, requires corrective actions, they should discuss their proposed corrective actions (e.g., changing inspection practices consistent with the NRC's position or submitting a TS amendment request with the associated safety basis for limiting the inspections) to achieve full compliance. If addressees choose to change their TS, the staff has included in the attachment suggested changes to the TS definitions for a tube inspection and for plugging limits to show what may be acceptable to the staff in cases where the tubes are expanded for the full depth of the tubesheet and where the extent of the inspection in the tubesheet region is limited.

(3) For plants where SG tube inspections have not been or are not being performed consistent with the NRC's position on the requirements in the TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B, the licensee should submit a safety assessment (i.e., a justification for continued operation based on maintaining tube structural and leakage integrity) that addresses any differences between the licensee's inspection practices and those called for by the NRC's position. Safety assessments should be submitted for all areas of the tube required to be inspected by the TS where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used, and should include the basis for not employing such inspection techniques. The assessment should include an evaluation of (1) whether the inspection practices rely on an acceptance standard (e.g., cracks located at least a minimum distance of x below the top of the tube sheet, even if these cracks cause complete severance of the tube) which is different from the TS acceptance standards (i.e., the tube plugging limits or repair criteria), and (2) whether the safety

assessment constitutes a change to the "method of evaluation" (as defined in 10 CFR 50.59) for establishing the structural and leakage integrity of the joint. If the safety assessment constitutes a change to the method of evaluation under 10 CFR 50.59, the licensee should determine whether a license amendment is necessary pursuant to that regulation.

Required Response

In accordance with 10 CFR 50.54(f), addressees are required to submit written responses to this generic letter. There are two options:

(a) Addressees may choose to submit written responses providing the information requested above within the requested time period. (Addressees who are implementing SG tube inspections in accordance with the staff position set forth in this GL need only describe the last inspections of their SG tubes to allow the staff to verify compliance).

(b) Addressees who cannot meet the requested completion date or who choose an alternate course of action are required to notify the NRC of these circumstances in writing as soon as possible but no later than 30 days from the date of this generic letter. The response must address any alternative course of action proposed, including the basis for the acceptability of the proposed alternative course of action and the basis for finding that the SGs remain operable. If the information requested in the previous section of this GL will be subsequently provided, the response must set forth the schedule for submitting the information.

The required written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, Maryland 20852, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, a copy of the response should be sent to the appropriate regional administrator.

Reasons for Requested Information

This generic letter requests addressees to submit information. The requested information will enable the NRC staff to determine whether licensees are implementing SG tube inspections in accordance with applicable requirements. In cases where licensees are not implementing inspections in such a manner, the requested information will allow the staff to determine whether the licensee's program complies with existing requirements (the plant TS in conjunction with 10 CFR Part 50, Appendix B, and the GDC or the plant-specific design basis, as appropriate).

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this generic letter transmits an information request for the purpose of verifying compliance with applicable existing requirements. Specifically, the requested information will enable the NRC staff to determine whether applicable requirements (plant TS in conjunction with 10 CFR Part 50, Appendix B) are being met. No backfit is either intended or approved in the context of issuance of this generic letter. Therefore, the staff has not performed a backfit analysis.

Federal Register Notification

A notice of opportunity for public comment on this generic letter was published in the *Federal Register* on May 14, 2003 (68 FR 25909). A total of 15 comments were received, 13 from the nuclear industry and 2 from the public. The staff considered all comments that were received. The staff's evaluation of the comments is publicly available through the NRC's Agencywide Documents Access and Management System (ADAMS) under Accession No. ML041690373.

Paperwork Reduction Act Statement

This generic letter contains information collection requirements 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 February 28, 2007.

The burden to the public for these mandatory information collections is estimated to average 60 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or any other aspect of these information collections, including suggestions for reducing the burden, to the Records and FOIA/Privacy Services Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@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.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

If you have any questions about this matter, please contact one of the persons listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
Bruce A. Boger, Director
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Attachment: As stated

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Sample Changes to the TS for Plants Limiting Inspections in the Tubesheet Region

Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. All tubes with degradation in the portion of the tube from x-inches below the bottom of the expansion transition (or the top of the tubesheet, whichever is lower) to the bottom of the expansion transition (or the top of the tubesheet, whichever is lower), shall be removed from service.

Tube Inspection means an inspection of the steam generator tube from x-inches below the hot-leg expansion transition or the top of tubesheet, whichever is lower, completely around the U-bend to the top support of the cold leg.

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	N/A	DATE RECEIVED

EVENT TITLE

NRC Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors

COUNTRY
US

PLANT AND UNIT
ALL PWRs

REACTOR TYPE
PWR

INITIAL STATUS
N/A

RATED POWER (MWe NET)
N/A

DESIGNER
N/A

1st COMMERCIAL OPERATION
N/A

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to request that addressees perform an evaluation of the emergency core cooling system and containment spray system recirculation functions in light of the information provided in this letter and, if appropriate, take additional actions to ensure system function.

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	1.2.5	1.4	1.3.1
2.	**Plant Status Prior to the Event:**	2.0		
3.	**Failed/Affected Systems:**	3.BG		
4.	**Failed/Affected Components:**	4.2.8	4.2.1	
5.	**Cause of the Event:**	5.1.1.8	5.7.1	
6.	**Effects on Operation:**	6.0		
7.	**Characteristics of the Incident:**	7.5		
8.	**Nature of Failure or Error:**	8.3		
9.	**Nature of Recovery Actions:**	9.0		

OMB Control No.: 3150-0011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555

September 13, 2004

NRC GENERIC LETTER 2004-02: POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS

Addressees

All holders of operating licenses for pressurized-water nuclear power reactors, except those who have 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 to:

(2) Request that addressees perform an evaluation of the emergency core cooling system (ECCS) and containment spray system (CSS) recirculation functions in light of the information provided in this letter and, if appropriate, take additional actions to ensure system function. Additionally, addressees are requested to submit the information specified in this letter to the NRC. This request is based on the identified potential susceptibility of pressurized-water reactor (PWR) recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of ECCS or CSS and on the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.

(3) Require addressees to provide the NRC a written response in accordance with 10 CFR 50.54(f).

Background

In 1979, as a result of evolving staff concerns related to the adequacy of PWR recirculation sump designs, the NRC opened Unresolved Safety Issue (USI) A-43, “Containment Emergency Sump Performance.” To support the resolution of USI A-43, the NRC undertook an extensive research program, the technical findings of which are summarized in NUREG-0897, “Containment Emergency Sump Performance,” dated October 1985. The resolution of USI A-43 was subsequently documented in Generic Letter (GL) 85-22, “Potential for Loss of

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Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985. Although the staff's regulatory analysis concerning USI A-43 did not support imposing new sump performance requirements upon licensees of operating PWRs or boiling-water reactors (BWRs), the staff found in GL 85-22 that the 50-percent blockage assumption (under which most nuclear power plants had been licensed) identified in Regulatory Guide (RG) 1.82, Sumps for Emergency Core Cooling and Containment Spray Systems, Revision 0 should be replaced with a more comprehensive requirement to assess debris effects on a plant-specific basis. The 50-percent screen blockage assumption does not require a plant-specific evaluation of the debris-blockage potential and may result in a nonconservative analysis for screen blockage effects. The staff also updated the NRC's regulatory guidance, including Section 6.2.2 of the Standard Review Plan (NUREG-0800) and RG 1.82 to reflect the USI A-43 technical findings documented in NUREG-0897.

Following the resolution of USI A-43 in 1985, several events challenged the conclusion that no new requirements were necessary to prevent the clogging of ECCS strainers at operating BWRs:

- On July 28, 1992, at Barsebäck Unit 2, a Swedish BWR, the spurious opening of a pilot-operated relief valve led to the plugging of two containment vessel spray system suction strainers with mineral wool and required operators to shut down the spray pumps and backflush the strainers.

- In 1993, at Perry Unit 1, two events occurred during which ECCS strainers became plugged with debris. On January 16, ECCS strainers were plugged with suppression pool particulate matter, and on April 14, an ECCS strainer was plugged with glass fiber from ventilation filters that had fallen into the suppression pool. On both occasions, the affected ECCS strainers were deformed by excessive differential pressure created by the debris plugging.

- On September 11, 1995, at Limerick Unit 1, following a manual scram due to a stuck-open safety/relief valve, operators observed fluctuating flow and pump motor current on the A loop of suppression pool cooling. The licensee later attributed these indications to a thin mat of fiber and sludge which had accumulated on the suction strainer.

In response to these ECCS suction strainer plugging events, the NRC issued several generic communications, including Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994; Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995; and, Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. These bulletins requested that BWR licensees implement appropriate procedural measures, maintenance practices, and plant modifications to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a loss-of-coolant accident (LOCA). The NRC staff has concluded that all BWR licensees have sufficiently addressed these bulletins.

However, findings from research to resolve the BWR strainer clogging issue raised questions concerning the adequacy of PWR sump designs. In comparison to the technical findings of the earlier USI A-43 research program on PWRs, the BWR research findings demonstrated that the amount of debris generated by a high-energy line break (HELB) could be greater, that the debris could be finer (and thus more easily transportable), and that certain combinations of debris (e.g., fibrous material plus particulate material) could result in a substantially greater head loss than an equivalent amount of either type of debris alone. These research findings prompted the NRC to open Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." The objective of GSI-191 is to ensure that post-accident debris blockage will not impede or prevent the operation of the ECCS and CSS in recirculation mode at PWRs during LOCAs or other HELB accidents for which sump recirculation is required.

On June 9, 2003, having completed its technical assessment of GSI-191 (summarized below in the Discussion section of this generic letter), the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors." As a result of the emergent issues discussed therein, the bulletin requested an expedited response from PWR licensees on the status of their compliance on a mechanistic basis with regulatory requirements concerning the ECCS and CSS recirculation functions. Addressees who chose not to confirm regulatory compliance were asked to describe any interim compensatory measures that have been implemented or will be implemented to reduce risk until the analysis could be completed. All licensees have since responded to Bulletin 2003-01. In developing Bulletin 2003-01, the NRC staff recognized that it may be necessary for addressees to undertake complex evaluations to determine whether regulatory compliance exists in light of the concerns identified in the bulletin and that the methodology needed to perform these evaluations was not currently available. As a result, that information was not requested in the bulletin, but addressees were informed that the staff was preparing a generic letter that would request this information. This generic letter is the follow-on to the bulletin.

In response to Bulletin 2003-01, PWR licensees that chose not to confirm regulatory compliance implemented or planned to implement compensatory measures to reduce risk or otherwise enhance the capability of the ECCS and CSS recirculation functions. Addressees' understanding of their facilities' ECCS and CSS recirculation capabilities may change when they resolve the potential concerns identified in this generic letter, and revise their analyses of sump performance. Therefore, addressees may find it necessary to reevaluate the adequacy of their compensatory measures in light of the new information and take further action as appropriate and necessary. Upon resolution of the potential concerns identified in this generic letter and the completion of any corrective actions resulting from that resolution, addressees may consider continuing, revising, or retiring their compensatory measures as appropriate.

The NRC has developed a Web page to keep the public informed of generic activities on PWR sump performance at (http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance.html). This page provides links to information on PWR sump performance issues, along with documentation of NRC interactions with industry (industry submittals, meeting notices, presentation materials, and meeting summaries). The NRC will continue to update this Web page as new information becomes available.

Discussion

In the event of an HELB inside the containment of a PWR, energetic pressure waves and fluid jets would impinge upon materials in the vicinity of the break, such as thermal insulation, coatings, and concrete, damaging and dislodging them. Debris could also be generated through secondary mechanisms, such as severe post-accident temperature and humidity conditions, flooding of the lower containment, and the impact of containment spray droplets. In addition to debris generated by jet forces from the pipe rupture, debris could be created by the chemical reaction between the materials in containment and the chemically reactive spray solutions used following a LOCA. These reactions might generate additional debris such as disbonded coatings and chemical precipitants. Through transport methods such as entrainment in the steam/water flows issuing from the break and containment spray washdown, a fraction of the generated debris and foreign material in the containment would be transported to the pool of water formed on the containment floor. Subsequently, if the ECCS or CSS pumps took suction from the recirculation sump, the debris suspended in the containment pool would begin to accumulate on the sump screen or be transported through the associated system. The accumulation of this suspended debris on the sump screen could create a roughly uniform covering on the screen, referred to as a debris bed, which would tend to increase the head loss across the screen through a filtering action. If a sufficient amount of debris accumulated, the debris bed would reach a critical thickness at which the head loss across the debris bed would exceed the net positive suction head (NPSH) margin required to ensure the successful operation of the ECCS and CSS pumps in recirculation mode. A loss of NPSH margin for the ECCS or CSS pumps as a result of the accumulation of debris on the recirculation sump screen, referred to as sump clogging, could result in degraded pump performance and eventual pump failure. Debris could also plug or wear close-tolerance components within the ECCS or CSS systems. This plugging or wear might cause a component to degrade to the point where it could not perform its designated function (i.e., pump fluid, maintain system pressure, or pass and control system flow.)

The primary object of the NRC's technical assessment of GSI-191 was to assess the likelihood that the ECCS and CSS pumps at domestic PWRs would experience a debris-induced loss of NPSH margin during sump recirculation. The NRC's technical assessment culminated in a parametric study documented in Volume 1 of NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2002. This study was a mechanistic treatment of phenomena associated with debris blockage using analytical models of domestic PWRs generated with a combination of generic and plant-specific data. The GSI-191 parametric study concluded that recirculation sump clogging was a credible concern for domestic PWRs. As a result of the limitations of plant-specific data and other modeling uncertainties, however, the parametric study did not definitively show whether particular PWR plants were vulnerable to sump clogging when phenomena associated with debris blockage were modeled mechanistically.

The methodology employed by the GSI-191 parametric study is based upon the substantial body of test data and analyses that are documented in technical reports generated during the NRC's GSI-191 research program and earlier technical reports by the NRC and the industry during the resolution of the BWR strainer clogging issue and USI A-43. Four of these NRC technical reports on debris generation, transport, accumulation, and head loss, are incorporated by reference into the GSI-191 parametric study:

- NUREG/CR-6770, "GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences," August 2002

- NUREG/CR-6762, Vol. 3, "GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation," August 2002

- NUREG/CR-6762, Vol. 4, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," August 2002

- NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA-Generated Debris," October 1995

In addition to demonstrating the potential for debris to clog containment recirculation sumps, operational experience and the NRC's technical assessment of GSI-191 have also identified three integrally related modes by which post-accident debris blockage could adversely affect the sump screen's design function of intercepting debris that could impede or prevent the operation of the ECCS and CSS in recirculation mode.

First, as a result of the 50-percent blockage assumption, most PWR sump screens were designed assuming that relatively small structural loadings would result from the differential pressure associated with debris blockage. Consequently, PWR sump screens may not be capable of accommodating the increased structural loadings that would occur due to mechanistically determined debris beds that cover essentially the entire screen surface. Inadequate structural reinforcement of a sump screen may result in its deformation, damage, or failure, which could allow large quantities of debris to be ingested into the ECCS and CSS piping, pumps, and other components, potentially leading to their clogging or failure. The credibility of this concern for screens and strainers that have not been designed with adequate reinforcement was shown by the ECCS strainer plugging and deformation events that occurred at Perry Unit 1 (further described in Information Notice (IN) 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993, and License Event Report (LER) 50-440/93-011, "Excessive Strainer Differential Pressure Across the RHR Suction Strainer Could Have Compromised Long-Term Cooling During Post-LOCA Operation," submitted May 19, 1993).

Second, in some PWR containments, the flowpaths by which containment spray or break flows return to the recirculation sump may include chokepoints, where the flowpath becomes so constricted that it could become blocked with debris following an HELB. Examples of potential chokepoints are pool drains, cavities, isolated containment compartments, and constricted drainage paths between physically separated containment elevations. Debris blockage at certain chokepoints could hold up substantial amounts of water required for adequate recirculation or cause the water to be diverted into containment volumes that do not drain to the recirculation sump. The holdup or diversion of water assumed to be available to support sump recirculation could result in an available NPSH for ECCS and CSS pumps that is lower than the analyzed value, thereby reducing assurance that recirculation would successfully function. A reduced available NPSH directly concerns sump screen design because the NPSH margin of the ECCS and CSS pumps must be conservatively calculated to determine correctly the required surface area of passive sump screens when mechanistically determined debris loadings are considered. Although the parametric study (NUREG/CR-6762, Vol. 1) did not analyze in detail the potential for the holdup or diversion of recirculation sump inventory, the NRC's GSI-191 research identified this phenomenon as an important and potentially credible concern. A number of LERs associated with this concern further confirm its credibility and potential significance:

- LER 50-369/90-012, "Loose Material Was Located in Upper Containment During Unit Operation Because of an Inappropriate Action," McGuire Unit 1, August 30, 1990

- LER 50-266/97-006, "Potential Refueling Cavity Drain Failure Could Affect Accident Mitigation," Point Beach Unit 1, February 19, 1997

- LER 50-455/97-001, "Unit 2 Containment Drain System Clogged Due to Debris," Byron Unit 2, April 17, 1997

- LER 50-269/97-010, "Inadequate Analysis of ECCS Sump Inventory Due to Inadequate Design Analysis," Oconee Unit 1, January 8, 1998

- LER 50-315/98-017, "Debris Recovered from Ice Condenser Represents Unanalyzed Condition," D.C. Cook Unit 1, July 1, 1998

Third, debris blockage at flow restrictions within the ECCS recirculation flowpath downstream of the sump screen is a potential concern for PWRs. Debris that is capable of passing through the recirculation sump screen may have the potential to become lodged at a downstream flow restriction, such as a high-pressure safety injection (HPSI) throttle valve or fuel assembly inlet debris screen. Debris blockage at such flow restrictions in the ECCS flowpath could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, debris blockage at flow restrictions in the CSS flowpath, such as a containment spray nozzle, could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Debris may also accumulate in close-tolerance subcomponents of pumps and valves. The effect may be either to plug the subcomponent, thereby rendering the component unable to perform its function, or to wear critical close-tolerance subcomponents to the point at which component or system operation is degraded and unable to fully perform its function. Considering the recirculation sump screen's design function of intercepting potentially harmful debris, it is essential that the screen openings be adequately sized and that the sump screen's current configuration be free of gaps or breaches which could compromise the ECCS and CSS recirculation functions. It is also essential that system components be designed and evaluated to be able to operate as necessary with debris laden fluid post-LOCA.

Section 50.46(c)(2) of Title 10 of the Code of Federal Regulations (10 CFR 50.46(c)(2)) defines an evaluation model as the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. An evaluation model includes one or more computer programs and all other information necessary for applying the calculational framework to a specific LOCA (the mathematical models used, the assumptions included in the programs, the procedure for treating the program input and output information, the parts of the analysis not included in the computer programs, values of parameters, and all other information necessary to specify the calculational procedure). Although not traditionally considered as a component of the 10 CFR 50.46 ECCS evaluation model, the calculation of sump performance is necessary to determine if the sump and the ECCS are predicted to provide enough flow to ensure long-term cooling.

Based on the new information identified during the efforts to resolve GSI-191, the staff has determined that the previous guidance used to develop current licensing basis analyses does not adequately and completely model sump screen debris blockage and related effects. As a result, due to the deficiencies in the previous guidance, an analytical error could be introduced which results in ECCS and CSS performance that does not conform with the existing applicable regulatory requirements outlined in this generic letter. Therefore, the staff is revising the guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS. In light of this revised staff guidance, it is appropriate to request that addressees perform new, more realistic analyses and submit information to confirm the functionality of the ECCS and CSS during design basis accidents requiring recirculation operations.

To assist in determining, on a plant-specific basis, the impact on sump screen performance and other related effects of extended post-accident operation with debris-laden fluids, addressees may use the guidance in Regulatory Guide (RG) 1.82, Revision 3, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” dated November 2003. Revision 3 enhanced the debris blockage evaluation guidance for PWRs provided in Revision 1 of the regulatory guide to better model sump screen debris blockage and related effects. Revision 1 replaced the 50-percent blockage assumption in Revision 0 with a comprehensive, mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. This was in response to the findings of USI A-43. The staff issued Revision 2 of the RG after evaluating blockage events such as the Barsebäck Unit 2 event mentioned above but for BWRs only. The NRC staff determined after the issuance of Revision 2 that research for PWRs indicated that the guidance in that revision was not comprehensive enough to ensure adequate evaluation of a PWR plant’s susceptibility to the detrimental effects of debris accumulation on debris interceptors (e.g., trash racks and sump screens). This led to the issuance of Revision 3 to address the PWRs. In addition, the NRC staff is reviewing generic industry guidance and will issue a safety evaluation endorsing acceptable portions or all of the generic industry guidance. Once approved, this guidance may also be used to assist in determining the status of regulatory compliance. For areas not addressed in the industry guidance, the NRC will provide guidance in the safety evaluation. Individual addressees may also develop an alternative to the approaches mentioned in this paragraph for responding to this generic letter; however, additional staff review may be required to assess the adequacy of such approaches.

The timeframes for addressee responses in this generic letter were selected to (1) allow adequate time for addressees to perform an analysis, (2) allow addressees to properly design and install any identified modifications, (3) allow addressees adequate time to obtain NRC approval, as necessary, for any licensing basis changes, (4) allow addressees adequate time to obtain NRC approval, as necessary, for any exemption requests, and (5) allow for the closure of the generic issue in accordance with the published schedule. These timeframes are appropriate since all addressees have responded to Bulletin 2003-01 and will, if necessary, implement compensatory measures until the issues identified in this generic letter are resolved.

The staff has assessed whether existing PWRs should continue operation while responding to this generic letter in light of the GSI-191 resolution schedule, proposed through December 31, 2007, and determined that continued operation is justified. The staff released a justification for continued operation in the “Summary of July 26-27, 2001, Meeting with Nuclear Energy Institute and Industry on ECCS Strainer Blockage in PWRs,” dated August 14, 2001. As discussed in this justification, continued plant operation is still justified for several reasons. First, the probability of the most severe initiating event (i.e., large and intermediate break LOCAs) is extremely low. More probable (although still low probability) small LOCAs would require less ECCS flow, take more time to use up the water inventory in the refueling water storage tank (RWST), and in some cases may not even require the use of recirculation from the ECCS sump because the flow through the break would be small enough that the operator will have sufficient time to initiate RHR operation and depressurize the reactor coolant system to terminate the loss of reactor coolant system inventory for higher elevation breaks. In addition, there are PWR design features which would tend to prevent blockage of the ECCS sumps during a LOCA. These features would tend to be effective for insulation and coating debris. For instance, the containments in PWRs tend to be very compartmentalized making the transport of debris to the sump screens more difficult. In addition, PWRs typically do not need to switch over to recirculation from the sump during a LOCA until greater than 20-30 minutes after the large break LOCA initiation and the elapsed time for all LOCAs will allow time for some of the debris to settle in other places within the containment. Coating debris, which is a major contributor to the latent debris in containment, would have a significant amount of time to settle. In addition, all PWRs have received approval by the staff for leak-before-break (LBB) credit on their largest RCS primary coolant piping. While LBB is not acceptable for demonstrating compliance with 10 CFR 50.46, it does demonstrate that LBB-qualified piping is sufficiently tough that it will most likely leak (even under safe shutdown earthquake conditions) rather than rupture. This would allow operators adequate opportunity to shut the plant down safely. Additionally, the staff notes that there are sources of margin in PWR designs which are not always credited in the licensing basis for each plant. For instance, NPSH analyses for most PWRs do not credit containment overpressure (which may be present during a LOCA). Any containment pressure greater than assumed in the NPSH analysis provides additional margin for ECCS operability during an accident. Another source of margin is that it has been shown that low pressure ECCS pumps would be able to continue operating in many cases for some time under cavitation conditions. Some licensees have vendor data demonstrating this. Such design margins such as these examples may prevent complete loss of ECCS recirculation flow or increase the time available for operator action (e.g., refilling the RWST) prior to loss of flow. Moreover, in response to Bulletin 2003-01, addressees have implemented or will implement interim compensatory measures to reduce the risk.

The staff has also determined that addressees are not required to be in compliance with the newly issued analysis using a NRC-approved methodology, until after all plant modifications (if required) are completed in accordance with the resolution schedule (i.e. December 31, 2007), which is located below in paragraph 2(b) of the Requested Information section, and addressees have changed their licensing basis, as appropriate. However, if a non-compliance with the existing licensing design basis that affects the operability of an ECCS or CSS design feature is identified while taking actions in response to the generic letter, addressees should comply with established regulatory requirements.

<u>Applicable Regulatory Requirements</u>

NRC regulations in Title 10, of the Code of Federal Regulations Section 50.46,10 CFR 50.46, require that the ECCS have the capability to provide long-term cooling of the reactor core following a LOCA. That is, the ECCS must be able to remove decay heat, so that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core.

Similarly, for PWRs licensed to the General Design Criteria (GDCs) in Appendix A to 10 CFR Part 50, GDC 38 provides requirements for containment heat removal systems, and GDC 41 provides requirements for containment atmosphere cleanup. Many PWR licensees credit a CSS, at least in part, with performing the safety functions to satisfy these requirements, and PWRs that are not licensed to the GDCs may similarly credit a CSS to satisfy licensing basis requirements. In addition, PWR licensees may credit a CSS with reducing the accident source term to meet the limits of 10 CFR Part 100 or 10 CFR 50.67. GDC 35 is listed in 10 CFR 50.46(d) and specifies additional ECCS requirements. PWRs that are not licensed to the GDCs typically have similar requirements in their licensing basis.

Applicable Regulatory Guidance[1]

Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," November 2003.

Requested Action

All addressees are requested to take the following actions:

Using an NRC-approved methodology, perform a mechanistic evaluation of the potential for the adverse effects of post-accident debris blockage and operation with debris-laden fluids to impede or prevent the recirculation functions of the ECCS and CSS following all postulated accidents for which the recirculation of these systems is required. Individual addressees may also use alternative methodologies to those already approved by the NRC; however, additional staff review may be required to assess the adequacy of such approaches.

Implement any plant modifications that the above evaluation identifies as being necessary to ensure system functionality.

Requested Information

All addressees are requested to provide the following information:

(a) Within 90 days of the date of the safety evaluation report providing the guidance for performing the requested evaluation, addressees are requested to provide information regarding their planned actions and schedule to complete the requested evaluation. The information should include the following:

    (a) A description of the methodology that is used or will be used to analyze the susceptibility of the ECCS and CSS recirculation functions for your reactor to the adverse effects identified in this generic letter of post-accident debris blockage and operation with debris-laden fluids identified in this generic letter. Provide the completion date of the analysis that will be performed.

    (b) A statement of whether you plan to perform a containment walkdown surveillance in support of the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of debris blockage identified in this generic letter. Provide justification if no containment walkdown surveillance will be performed. If a containment walkdown surveillance will be performed, state the planned methodology to be used and the planned completion date.

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1 The NRC staff is currently reviewing evaluation guidance developed by the industry. The NRC staff intends to document its review in a safety evaluation which licensees can reference as regulatory guidance.

2. Addressees are requested to provide the following information no later than September 1, 2005:

   (a) Confirmation that the ECCS and CSS recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This submittal should address the configuration of the plant that will exist once all modifications required for regulatory compliance have been made and this licensing basis has been updated to reflect the results of the analysis described above.

   (b) A general description of and implementation schedule for all corrective actions, including any plant modifications, that you identified while responding to this generic letter. Efforts to implement the identified actions should be initiated no later than the first refueling outage starting after April 1, 2006. All actions should be completed by December 31, 2007. Provide justification for not implementing the identified actions during the first refueling outage starting after April 1, 2006. If all corrective actions will not be completed by December 31, 2007, describe how the regulatory requirements discussed in the Applicable Regulatory Requirements section will be met until the corrective actions are completed.

   (c) A description of the methodology that was used to perform the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. The submittal may reference a guidance document (e.g., Regulatory Guide 1.82, Rev. 3, industry guidance) or other methodology previously submitted to the NRC. (The submittal may also reference the response to Item 1 of the Requested Information described above. The documents to be submitted or referenced should include the results of any supporting containment walkdown surveillance performed to identify potential debris sources and other pertinent containment characteristics.)

   (d) The submittal should include, at a minimum, the following information:

      (i) The minimum available NPSH margin for the ECCS and CSS pumps with an unblocked sump screen.

      (ii) The submerged area of the sump screen at this time and the percent of submergence of the sump screen (i.e., partial or full) at the time of the switchover to sump recirculation.

      (iii) The maximum head loss postulated from debris accumulation on the submerged sump screen, and a description of the primary constituents of the debris bed that result in this head loss. In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) and CSS washdown should be considered in the analyses. Examples of this type of debris are disbonded coatings in the form of chips and particulates and chemical precipitants caused by chemical reactions in the pool.

(iv) The basis for concluding that the water inventory required to ensure adequate ECCS or CSS recirculation would not be held up or diverted by debris blockage at choke-points in containment recirculation sump return flowpaths.

(v) The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flowpaths downstream of the sump screen, (e.g., a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles). The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

(vi) Verification that close-tolerance subcomponents in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive wear due to extended post-accident operation with debris-laden fluids.

(vii) Verification that the strength of the trash racks is adequate to protect the debris screens from missiles and other large debris. The submittal should also provide verification that the trash racks and sump screens are capable of withstanding the loads imposed by expanding jets, missiles, the accumulation of debris, and pressure differentials caused by post-LOCA blockage under predicted flow conditions.

(viii) If an active approach (e.g., backflushing, powered screens) is selected in lieu of or in addition to a passive approach to mitigate the effects of the debris blockage, describe the approach and associated analyses.

(e) A general description of and planned schedule for any changes to the plant licensing bases resulting from any analysis or plant modifications made to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. Any licensing actions or exemption requests needed to support changes to the plant licensing basis should be included.

(f) A description of the existing or planned programmatic controls that will ensure that potential sources of debris introduced into containment (e.g., insulations, signs, coatings, and foreign materials) will be assessed for potential adverse effects on the ECCS and CSS recirculation functions. Addressees may reference their responses to GL 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," to the extent that their responses address these specific foreign material control issues.

Required Response

In accordance with 10 CFR 50.54(f), the PWR addressees are required to submit written responses to this generic letter. This information is sought to verify licensees' compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter once their licensing basis has been updated to reflect the results of the mechanistic analysis requested in this generic letter. This request is based on the identified potential susceptibility of PWR recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of ECCS and CSS and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage. The addressees have two options:

(a) Addressees may choose to submit written responses providing the information requested above within the requested time period.

(b) Addressees who choose not to provide information requested or cannot meet the requested completion dates are required to submit written responses within 30 days of the date of this generic letter. The responses must address any alternative course of action proposed, including the basis for the acceptability of the proposed alternative course of action.

The required written responses should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, Maryland 20852, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, a copy of a response should be submitted to the appropriate regional administrator.

The NRC staff will review the responses to this generic letter and will notify affected addressees if concerns are identified regarding compliance with NRC regulations. The staff may also conduct inspections to determine addressees' effectiveness in addressing the generic letter.

Reasons for Information Request

As discussed above, research and analysis suggest that (1) the potential for the failure of the ECCS and CSS recirculation functions as a result of debris blockage is not adequately addressed in most PWR licensees' current safety analyses, and (2) the ECCS and CSS recirculation functions at a significant number of operating PWRs could potentially become degraded as a result of the effects of debris blockage or extended operation with debris-laden fluids as identified in this generic letter. An ECCS that is incapable of providing long-term reactor core cooling through recirculation operation would be in violation of 10 CFR 50.46. A CSS that is incapable of functioning in recirculation mode may not comply with GDCs 38 and 41 or other plant-specific licensing requirements or safety analyses. Bulletin 2003-01 requested information to verify addressees' compliance with NRC regulations and to ensure that any interim risks associated with post-accident debris blockage are minimized while evaluations to determine compliance proceed. This generic letter is the follow-on generic communication to Bulletin 2003-01 and requests information on the results of the evaluations referenced in the bulletin. This information is sought to verify licensees' compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter once their licensing basis has been updated to reflect the results of the mechanistic analysis requested in this generic letter. This request is based on the identified potential susceptibility of

PWR recirculation sump screens to debris blockage during design basis accidents requiring recirculation operation of the ECCS and CSS and the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.

The NRC staff will also use the requested information to (1) determine whether a sample auditing approach is acceptable for verifying that addressees have resolved the concerns identified in this generic letter, (2) assist in determining which addressees will be subject to the proposed sample audits, (3) provide confidence that any nonaudited addressees have addressed the concerns identified in this generic letter, and (4) assess the need for and guide the development of any additional regulatory actions that may be necessary to address the adequacy of the ECCS and CSS recirculation functions.

Related Generic Communications

- Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," June 9, 2003.

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

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

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

- Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," February 18, 1994.
- 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," July 14, 1998.

- Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," October 7, 1997.

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

- Generic Letter 91-18, Rev. 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," October 8, 1997.

- Information Notice 97-13, "Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants," March 24, 1997.

- Information Notice 96-59, "Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris," October 30, 1996.

- Information Notice 96-55, “Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps Under Design Basis Accident Conditions,” October 22, 1996.

- Information Notice 96-27, “Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation,” May 1, 1996.

- Information Notice 96-10, “Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillances,” February 13, 1996.

- Information Notice 95-47, “Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage,” October 4, 1995.

- Information Notice 95-47, Revision 1, “Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage,” November 30, 1995.

- Information Notice 95-06, “Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment,” January 25, 1995.

- Information Notice 94-57, “Debris in Containment and the Residual Heat Removal System,” August 12, 1994.

- Information Notice 93-34, “Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment,” April 26, 1993.

- Information Notice 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.
- Information Notice 92-85, “Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage,” December 23, 1992.

- Information Notice 92-71, “Partial Plugging of Suppression Pool Strainers at a Foreign BWR,” September 30, 1992.

- Information Notice 89-79, “Degraded Coatings and Corrosion of Steel Containment Vessels,” December 1, 1989.

- Information Notice 89-79, Supplement 1, “Degraded Coatings and Corrosion of Steel Containment Vessels,” June 29, 1990.

- Information Notice 89-77, “Debris in Containment Emergency Sumps and Incorrect Screen Configurations,” November 21, 1989.

- Information Notice 88-28, “Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage,” May 19, 1988.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, 10 CFR 50.109(a)(4)(i) and 10 CFR 50.54(f), this generic letter requests that addressees evaluate their facilities to confirm compliance with the existing applicable regulatory requirements as outlined in this generic letter. This generic letter also transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements. The staff has determined that, in light of the information identified during the efforts to resolve GSI-191, the previous guidance used to develop most addressees' current licensing basis analyses does not adequately and completely model sump screen debris blockage and related effects. Due to the deficiencies in the previous guidance, a potential analytical error could have been introduced which results in ECCS and CSS performance that does not conform with existing applicable regulatory requirements. In response, the staff revised its guidance for determining the susceptibility of PWR recirculation sump screens to the adverse effects of debris blockage during design basis accidents requiring recirculation operation of the ECCS or CSS to ensure compliance with existing applicable regulatory requirements. Thus, the information requested by this generic letter is considered a compliance exception to the rule in accordance with 10 CFR 50.109(a)(4)(i).

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this generic letter is subject to the Small Business Regulatory Enforcement Fairness Act of 1996. Office of Management and Budget (OMB) has declared the letter not to be a major rule. Notification of the letter has been sent to Congress.

Federal Register Notification

A notice of opportunity for public comment on this generic letter was published in the Federal Register (69 FR16980) on March 31, 2004. Comments were received from ten industry groups, one non-profit organization, one private citizen, and the State of New Jersey. The staff considered all comments that were received. The staff's evaluation of the comments is publicly available through the NRC's Agencywide Documents Access and Management System (ADAMS) under Accession No. ML042260161.

Paperwork Reduction Act Statement

This generic letter contains 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 (OMB) under approval number 3150-0011, which expires on February 28, 2007.

The burden to the public for these mandatory information collections is estimated to average 7000 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the necessary data, and completing and reviewing the information collections. The staff received two public comments on the estimated burden to the public. In both comments, the burden was estimated to be between 5,000 and 10,000 hours. The staff solicited input from three addressees to better estimate the burden to the public. Based on the public comments and the solicited input, the staff estimates the burden as shown above. Send comments regarding this burden estimate or any other aspect of these information collections, including suggestions for reducing the burden, to the Records and

FOIA/Privacy Services Branch (T-5F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@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.

Public Protection Notification

The NRC may neither conduct nor sponsor, and an individual is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

If you have any questions about this matter, please contact the technical contacts or lead project manager listed below.

/RA/
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