DISTRIBUTION for NRR Director's Quarterly Status Report

Central File	RGEB R/F	FJMiraglia, EDO	SJCollins, NRR
RZimmerman, NRR	JJohnson, NRR	BWSheron, NRR	DBMatthews, NRR
CACarpenter, NRR	SWest, NRR	MMalloy, NRR	EMMcKenna, NRR
CPetrone, NRR	EYWang, NRR	BJSweeney, NRR	BABoger, NRR
JAZwolinski, NRR	GMHolahan, NRR	JRStrosnider, NRR	JESilber, NRR
RCEmrit, RES	Regional Administrators		

Mr. Ralph Beedle, Senior Vice President
& Chief Nuclear Officer
Nuclear Energy Institute
1776 I Street NW
Suite 400
Washington, D.C. 20006-3708

Nancy G. Chapman, SERCH Manager
Bechtel Power Corporation
9801 Washingtonian Blvd.
Gaithersburg, Maryland 20878-5356

Mr. R. P. LaRhette
Institute of Nuclear Power Operations
700 Galleria Parkway
Atlanta, Georgia 30339-5979

Mr. Lee Watkins
Assistant Manager For High Level Waste
U.S. DOE
P.O. Box A
Aiken, South Carolina 29892

Mr. S. Scott
Office of Nuclear Safety, DOE
Century 21 Building (E-H72)
19901 Germantown Road
Germantown, MD 20874-1290

Mr. Bob Borsum
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Ms. Norena G. Robinson, Licensing Technician
Nebraska Public Power District
Cooper Nuclear Station
P.O. Box 98
Brownsville, NE 68321

ADAMS ACCESSION NUMBER ML003766789

DISTRIBUTION for NRR Director's Quarterly Status Report

Central File	RGEB R/F	FJMiraglia, EDO	SJCollins, NRR
RZimmerman, NRR	JJohnson, NRR	BWSheron, NRR	DBMatthews, NRR
CACarpenter, NRR	SWest, NRR	MMalloy, NRR	EMMcKenna, NRR
CPetrone, NRR	EYWang, NRR	BJSweeney, NRR	BABoger, NRR
JAZwolinski, NRR	GMHolahan, NRR	JRStrosnider, NRR	JESilber, NRR
RCEmrit, RES	Regional Administrators		

Mr. Ralph Beedle, Senior Vice President
& Chief Nuclear Officer
Nuclear Energy Institute
1776 I Street NW
Suite 400
Washington, D.C. 20006-3708

Nancy G. Chapman, SERCH Manager
Bechtel Power Corporation
9801 Washingtonian Blvd.
Gaithersburg, Maryland 20878-5356

Mr. R. P. LaRhette
Institute of Nuclear Power Operations
700 Galleria Parkway
Atlanta, Georgia 30339-5979

Mr. Lee Watkins
Assistant Manager For High Level Waste
U.S. DOE
P.O. Box A
Aiken, South Carolina 29892

Mr. S. Scott
Office of Nuclear Safety, DOE
Century 21 Building (E-H72)
19901 Germantown Road
Germantown, MD 20874-1290

Mr. Bob Borsum
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Ms. Norena G. Robinson, Licensing Technician
Nebraska Public Power District
Cooper Nuclear Station
P.O. Box 98
Brownsville, NE 68321

ADAMS ACCESSION NUMBER ML003766789

DOCUMENT NAME: DIST.WPD
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" Copy with attachment/enclosure "N" = No copy

OFFICE	RGEB:DRIP		RGEB:DRIP		REXB:DRIP		RGEB:DRIP		RGEB:DRIP		RGEB:DRIP		D:DRIP	
NAME	BSweeney:bs		EMcKenna		CPetrone		MMalloy		SWest		CCarpenter		DMatthews	
DATE	11/01/00		11/08/00		11/06/00		11/06/00		11/07/00		11/08/00		11/16/00	

OFFICIAL RECORD COPY

INTRODUCTION

The purpose of this report is to provide information about generic activities, including generic communications, under the cognizance of the Office of Nuclear Reactor Regulation. This report, which focuses on compliance activities, complements NUREG-0933, "A Prioritization of Generic Safety Issues."

This report includes two attachments: 1) action plans and 2) generic communications under development and other generic compliance activities. Generic communications and compliance activities (GCCAs) are potential generic issues that are safety significant, require technical resolution, and possibly require generic communication or action.

Attachment 1, "NRR Action Plans," includes generic or potentially generic issues of sufficient complexity or scope that require substantial NRC staff resources. The issues covered by action plans include concerns identified through review of operating experience (e.g., Boiling Water Reactor Internals), and issues related to regulatory flexibility and improvements (e.g., Emergency Action Level Guidance Development). For each action plan, the report includes a description of the issue, key milestones, discussion of its regulatory significance, current status, and names of cognizant staff.

Attachment 2, "Open Generic Communications and Compliance Activities," consists of three status reports: 1) Open GCCAs, 2) GCCAs added since the previous report, and 3) GCCAs closed since the previous report. The generic communications listed in the attachment include bulletins, generic letters, regulatory issue summaries (which replace administrative letters), and information notices. Compliance activities listed in the attachment do not rise to the level of complexity that require an action plan, and a generic communication is not currently scheduled. For each GCCA, there is a short description of the issue, scheduled completion date, and name of cognizant staff.

DIRECTOR'S STATUS REPORT

on

GENERIC ACTIVITIES

Action Plans

Generic Communication and Compliance Activities

OCTOBER 2000

Office of Nuclear Reactor Regulation

ATTACHMENT 1

NRR ACTION PLANS

TABLE OF CONTENTS

BOILING WATER REACTOR INTERNALS

Open TAC Nos.: M96539, M98266, M99638, M99870, M99895, MA0782, MA0783, MA0784, MA0785, MA0786, MA0787, MA0788, MA0789, MA0790, MA0791, MA0792, MA0793, MA1926, MA1927, MA2326, MA2328, MA3673, MA4203, MA4464, MA4465, MA4467, MA4468, MA5012, MA5140, MA6015, MA7323, MA7356, MA9111

Last Update: 09/30/00
Lead NRR Division: DE
Supporting Division: DSSA
GSI: Not Available

MILESTONES	DATE (T/C)[1]
PART I: REVIEW OF GENERIC INSPECTION AND EVALUATION CRITERIA	
1. Issue summary NUREG-1544	03/96 (C)
○ Update NUREG-1544	3Q/01 (T)
2. Review BWRVIP Re-inspection and Evaluation Criteria	
○ Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03)	07/15/99 (CA)
○ BWRVIP-03, Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components	07/15/99 (CA)
○ BWR Vessel Shell Weld Inspection Recommendations (BWRVIP-05) . .	07/28/98 (CA)
○ BWR Axial Shell Weld Inspection Recommendations	03/07/00 (CA)
○ Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07)	04/27/98 (CA)
3. Review of generic repair technology, criteria, and guidance	TBD
4. Review generic mitigation guidelines and criteria	TBD
5. Review of generic NDE technologies developed for examinations of BWR internal components and attachments	TBD
6. Other Internals reviews (safety assessments, evaluations, mitigation measures, inspections, and repairs)	
○ Safety Assessment of BWR Reactor Internals (BWRVIP-06)	09/15/98 (CA)
○ Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08 & BWRVIP-46)	03/27/98 (CA)
○ Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14)	12/03/99 (CA)
○ Internal Core Spray Piping and Sparger Replacement Design Criteria (BWRVIP-16)	08/10/00 (CA)
○ Roll/Expansion of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)	03/13/98 (CD)
○ BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)	12/02/99 (CA)
○ BWRVIP-18, Appendix C, BWR Core Spray Internals Demonstration of Compliance With Technical Information Requirements of License Renewal Rule (10 CFR 54.21)	09/06/00 (CA)
○ Internal Core Spray Piping and Sparger Repair Design Criteria (BWRVIP-19)	08/10/00 (CA) 12/19/99 (CA)
○ Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25) . . .	

MILESTONES	DATE (T/C)[1]
○ Top Guide Inspection and Flaw Evaluation Guideline (BWRVIP-26)	09/29/99 (CA)
○ Standby Liquid Control System / Core Plate ΔP Inspection and Flaw Evaluation Guidelines (BWRVIP-27)	04/27/99 (CA)
○ Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)	04/10/00 (CA)
○ Technical Basis for Part Circumferential Weld Overlay Repair of Vessel Internal Core Spray Piping (BWRVIP-34)	TBD
○ Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)	07/24/00 (CA)
○ BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)	06/20/00 (CI) 05/26/00 (CA)
○ BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42)	03/27/98 (CA)
○ Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)	10/13/99 (CA) 09/29/99 (CA)
○ BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47)	08/04/98 (CA)
○ Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (BWRVIP-48)	11/01/00 (T) 11/01/00 (T)
○ Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49)	11/01/00 (T) 11/01/00 (T)
○ Top Guide / Core Plate Repair Design Criteria (BWRVIP-50)	12/31/00 (T)
○ Jet Pump Repair Design Criteria (BWRVIP-51)	12/31/00 (T)
○ Shroud Support and Vessel Repair Design Criteria (BWRVIP-52)	12/31/00 (T)
○ Standby Liquid Control Line Repair Design Criteria (BWRVIP-53)	12/31/00 (T)
○ Lower Plenum Repair Design Criteria (BWRVIP-55)	
○ LPCI Coupling Repair Design Criteria (BWRVIP-56)	09/19/00 (CI)
○ Instrument Penetrations Repair Design Criteria (BWRVIP-57)	07/08/99 (CA)
○ CRD Internal Access Weld Repair (BWRVIP-58)	
○ Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals (BWRVIP-59)	02/01/01 (T)
○ BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants (BWRVIP-60)	04/18/00 (CI)
○ Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)	02/28/01 (T)
○ Shroud Vertical Weld Inspection and Evaluation Guidelines (BWRVIP-63)	09/15/00 (CI)
○ BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)	02/28/01 (T)
○ Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)	02/28/01 (T)
○ BWR Core Shroud Inspection & Flaw Evaluation Guidelines (BWRVIP-76)	
○ BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material (BWRVIP-78)	

[1] CA = Complete, Acceptable (i.e., final SER); CI= Complete, Interim (i.e., draft SER); CD = Complete, Denied

Description: Many components inside boiling water reactor (BWR) vessels (i.e., internals) are made of materials such as stainless steel and various alloys that are susceptible to corrosion and cracking. This degradation can be accelerated by stresses from temperature and pressure changes, chemical interactions, irradiation, and other corrosive environments. This action plan is intended to encompass the evaluation and resolution of issues associated with intergranular stress corrosion cracking (IGSCC) in BWR internals. This includes plant specific reviews and the assessment of the generic criteria that have been proposed by the BWR Owners Group and the BWRVIP technical subcommittees to address IGSCC in core shrouds and other BWR internals.

Historical Background: Significant cracking of the core shroud was first observed at Brunswick, Unit 1 nuclear power plant in September 1993. The NRC notified licensees of Brunswick's discovery of significant circumferential cracking of the core shroud welds. In 1994, core shroud cracking continued to be the most significant of reported internals cracking. In July 1994, the NRC issued Generic Letter (GL) 94-03 which requires licensees to inspect their shrouds and provide an analysis justifying continued operation until inspections can be completed.

A special industry review group (Boiling Water Reactor Vessels and Internals Project - BWRVIP) was formed to focus on resolution of reactor vessel and internals degradation. This group was instrumental in facilitating licensee responses to NRC's GL 94-03. The NRC evaluated the review group's reports, submitted in 1994 and early 1995, and all plant specific responses.

All of the plants evaluated were able to demonstrate continued safe operation until inspection or repair on the basis of: 1) no 360° through-wall cracking observed to date, 2) low frequency of pipe breaks, and 3) short period of operation (2-6 months) before all of the highly susceptible plants complete repairs of or inspections to their core shrouds.

In late 1994, extensive cracking was discovered in the top guide and core plate rings of a foreign reactor. The design is similar to General Electric (GE) reactors in the U.S., however, there have been no observations of such cracking in U.S. plants. GE concluded that it was reasonable to expect that the ring cracking could occur in GE BWRs with operating time greater than 13 years. In the special industry review group's report, that was issued in January 1995, ring cracking was evaluated. The NRC concluded that the BWRVIP's assessment was acceptable and that top guide ring and core plate ring cracking is not a short term safety issue.

Proposed Actions: The staff has been interacting with the BWRVIP and individual licensees. In an effort to lower the number of industry and staff resources that will be needed in the future, it is important for the staff to continue interacting with the industry on a generic basis in order to encourage them to continue their proactive efforts to resolve IGSCC of BWR internals as a voluntary industry initiative. The BWRVIP has submitted 54 generic documents, supporting plant-specific submittals, for staff review. The staff is ensuring that the generic reviews are incorporating recent operating experience on all BWR internals.

Originating Document: Generic Letter 94-03, issued July 25, 1994, which requested BWR licensees to inspect their core shrouds by the next outage and to justify continued safe operation until inspections can be completed.

Regulatory Assessment: In July 1994, the NRC issued Generic Letter 94-03 which required licensees to inspect their shrouds and provide an analysis justifying continued operation until inspections could be performed. The staff has concluded in all cases that licensees have provided sufficient evidence to support continued operation of their BWR units to the refueling outages in which shroud inspections or repairs have been scheduled. In addition, in October 1995, industry's special review group submitted a safety assessment of postulated cracking in all BWR reactor internals and attachments to assure continuing safe operation.

Current Status: Almost all BWRs completed inspections or repairs of core shrouds during refueling outages in the fall of 1995. Various repair methods have been used to provide alternate load carrying capability, including preemptive repairs, installation of a series of clamps and use of a series of tie-rod assemblies. The NRC has reviewed and approved all shroud modification proposals that have been submitted by BWR licensees. Review by NRC continues on individual plant reinspection results and plant-specific assessments.

The BWRVIP has submitted Appendices to the Inspection and Flaw Evaluation Guidelines. These appendices address the use of BWRVIP generic inspection guidelines for compliance with requirements of the license renewal rule (10 CFR Part 54). The staff is reviewing these appendices in conjunction with its review of the BWRVIP guidelines, and has issued the first several of thirteen license renewal SEs on BWR internals, with the remaining expected to be completed by February 2001. This schedule change is primarily attributed to BWRVIP-38, -41, -74, and -76, since the staff is waiting for the BWRVIP to supplement its original submittal in accordance with the open items in the staff's initials SE's on these reports.

The BWRVIP submitted BWRVIP-28 to address the safety implications of recent cracking found in BWR jet pump riser elbows. The staff issued NRC Information Report IN 97-02, "Cracks Found in Jet Pump Riser Assembly Elbows at Boiling Water Reactors," on February 6, 1997.

Information Notice 97-17, "Cracking of Vertical Welds in the Core Shroud and Degraded Repair," was issued April 4, 1997, to inform the industry of vertical weld cracks and a degraded core shroud repairs found at Nine Mile Point, Unit 1.

By letters dated April 25 and May 30, 1997, the BWRVIP provided a reaffirmation of the BWR member licensees to the BWRVIP, and committed, on behalf of their member licensees, to several actions, including implementing the BWRVIP topical reports at each BWR as appropriate considering individual plant schedules, configurations and needs, and providing timely notification to the NRC staff if a plant does not implement the applicable BWRVIP products.

NRR Technical Contacts: Robert Hermann, EMCB, 415-2768
Amy Cubbage, SRXB, 415-2875
Jai Rajan, EMEB, 415-2788

NRR Lead PM: C. E. Carpenter, EMCB, 415-2169

References:

Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," July 25, 1994.

Action Plan dated April 1995.

STEAM GENERATORS

TAC Nos. M88885, M99432, and MA4265 Last Update: 09/30/00
Lead Division: DE (#394)

MILESTONE		DATE (T/C)
1.	SECY Paper delaying issuance of GL while staff works on industry initiative (NEI 97-06)	10/98 (C)
2.	NEI submits NEI 97-06	02/04/00 (C)
3.	Commission Paper outlining staff review and implementation process for NEI 97-06	03/30/00 (C)
4.	Staff completes review and draft safety evaluation of NEI 97-06	01/01 (T)
5.	Staff briefs CRGR	03/01 (T)
6.	Publish draft SE in FR for public comment	03/01 (T)
7.	ACRS review	0401 (T)
8.	Staff endorses NEI 97-06	06/01 (T)

Brief Description: The NRC originally planned to develop a rule pertaining to steam generator tube integrity. The proposed rule was to implement a more flexible regulatory framework for steam generator surveillance and maintenance activities that allows a degradation specific management approach. The results of the regulatory analysis suggested that the more optimal regulatory approach was to utilize a generic letter. The NRC staff suggested, and the Commission subsequently approved, a revision to the regulatory approach to utilize a generic letter. In SECY-98-248, the staff recommended to the Commission that the proposed GL be put on hold for 3 months while the staff works with NEI on their NEI 97-06 initiative. In the staff requirements memorandum dated December 21, 1998, the Commission did not object to the staff's recommendation. In late 1998 and 1999 the NRC and industry addressed NRC technical and regulatory concerns with the NEI 97-06 initiative, and on February 4, 2000, NEI submitted the generic licensing change package for NRC review. The generic licensing change package included NEI 97-06, Revision 1, proposed generic technical specifications, and a model technical requirements manual section.

Regulatory Assessment: The current regulatory framework provides reasonable assurance that operating PWRs are safe. The current regulatory framework that implements governing requirements through the plant technical specifications can be improved.

Current Status:
- Briefed ACRS on ANPRM -- August 1994
- SG rule ANPRM -- September 1994
- SECY-95-131 -- May 1995 -- justifies continuation of rulemaking
- Briefed Commission on SG rule -- June 1995
- Briefed Commission on SG rule status -- February 1996
- Memo to Commission re. revised schedule -- May 1996
- Briefed Chairman on status -- July 1996
- Information Brief for CRGR -- October 1996
- ACRS Brief on SG rule -- November 5, 6, 1996
- Briefed Chairman on SG rule status -- December 1996
- Briefed ACRS re. risk-informed approach for SG rule -- January 1997

- Briefed ACRS re. risk assessment and regulatory analysis results -- March 4, 5, and April 3, 1997
- COMSECY-97-013 suggests revising approach to a GL -- May 1997
- Briefed Commissioner Assistants re. revised approach -- June 5, 1997
- SRM of June 30, 1997, agrees with revised regulatory approach
- Briefed ACRS re. revised approach -- June 12, 1997
- Met with NEI/industry senior mgmt re. GL status -- July 22, 1997
- Briefed ACRS re. GL/DG-1074/DPO issues -- August 26, 27, September 3, 1997
- Information Brief for CRGR re. GL and backfit -- September 9, 1997
- Met w/NEI re. GL/DG-1074/TSs -- September 11, 1997
- ACRS endorsement to issue GL and DG-1074 for public comment -- September 15, 1997
- Briefed ACRS re. DPO issues -- October 2, 1997
- ACRS endorsement to issues DPO document for public comment -- October 10, 1997
- GL package into concurrence -- October 21, 1997
- NEI submits NEI 97-06 "Steam Generator Program Guidelines" -- December 16, 1997
- CRGR package concurred on by NRR and sent to CRGR April 14, 1998
- Met with CRGR on June 12, 1998, for information briefing on package
- Met with CRGR on July 21, 1998, for detailed review of proposed GL package
- Memo from Collins to Callan dated September 11, 1998, suggests putting proposed GL on hold for 3 months to work with NEI on NEI 97-06
- Staff issued Commission paper SECY-98-248 (October 28, 1998) recommending a 3 month hold on issuance of proposed GL. SECY-98-248 also recommended issuance of (1) DG-1074, (2) the DPO consideration document, and (3) the September 1998 Hopenfeld memorandum to the Commission, for public comment
- The Commission, in SRM dated December 21, 1998, agreed to above recommendations
- Held technical and management meetings with industry on 10/7/98, 10/28-29/98, 11/12/98, 11/18/98, 2/10/99, and 2/24/99 to resolve technical and regulatory implementation issues regarding NEI 97-06.
- Draft regulatory guide DG-1074 was issued for public comment (appeared in federal register on 1/20/99) with the DPO consideration document, and the Hopenfeld memorandum to the Commission
- Internal guidance to SG inspectors was issued on 1/25/99 indicating that DG-1074 should not be used for inspection guidance as directed by the Commission's SRM of 12/21/98.
- Briefed ACRS Materials S/C on 3/24/99 regarding the status of current regulatory approach.
- May 6, 1999 technical meeting with NEI/Industry: Discussed industry proposed TS & Technical Requirements Manual (TRM).
- May 18, 1999 senior management meeting with NEI/Industry: Discussed status
- June 24, 1999 technical meeting with NEI/Industry: Discussed open technical issues; NEI presented a revised TS & TRM.
- June 30, 1999, comment period closes for draft regulatory guide (DG-1074) and DPO consideration document.
- August 27, 1999, senior management meeting with NEI/industry to discuss remaining open issues.
- October 5, 1999, reached tentative agreement with NEI/industry regarding open issues.
- February 4, 2000, NEI submitted the steam generator generic licensing change package.
- March 17, 2000, senior management meeting with NEI/industry to discuss the package and its implementation.
- July 26, 2000, meeting with NEI/industry to discuss changes to the package based on additional industry comments.

<u>NRR Technical Contact</u>: Ted Sullivan, EMCB, 415-2796

<u>RES Contact</u>: N/A

OKONITE CABLE LOCA TEST FAILURES

TAC Nos. MA8193, MA9199, MA9200, & MA9201

Last Update: 09/30/00
Lead Division: DE

MILESTONES	DATE (T/C)
1. Meet with Okonite to discuss LOCA test #5 cable failure results	2/08/00 (C)
2. Meet with nuclear industry to discuss LOCA test #5 cable failure results	2/16/00 (C)
3. Issue letter to Okonite with BNL test report	5/17/00 (C)
4. Issue letter to NEI with BNL test report	5/18/00 (C)
5. Meet with NEI and Okonite to discuss impact on operating reactors and responses being considered by NRC and industry	6/22/00 (C)
6. Based on the 10/12 meeting with industry and Okonite to discuss the results of the NEI survey, staff will determine if any of the following regulatory actions are warranted:	
a. If a small number of plants are affected, they will be addressed individually.	TBD
b. If industry sufficiently addresses the issues and several plants are affected, the staff will publish a Regulatory Issue Summary in accordance with SECY-99-143.	TBD
c. If the industry initiative is inadequate, the staff will issue a generic letter to licensees to obtain information on affected safety-related equipment and plants.	TBD

Description: This plan is intended to guide staff efforts to address the issues raised by the Office of Nuclear Regulatory Research (RES) in a memorandum dated May 2, 2000, concerning the results of Loss of-Coolant-Accident (LOCA) testing of bonded-jacket Okonite single-conductor instrumentation and control low-voltage cables conducted in November 1999, by Brookhaven National Laboratories (BNL) at Wyle Laboratories for RES as part of Generic safety Issue 168, “Environmental Qualification of Electrical Equipment.”

Background: In related past research, Sandia National Laboratories, under contract to the NRC, performed tests on the same Okonite cable, along with several other cables. The results of this testing are described in NUREG/CR-5772, “Aging, Condition Monitoring, and Loss-of-Coolant Accident (LOCA)

Tests of Class 1E Electrical Cables, “Volumes 1, 2, and 3. In that program, one of the cable types that failed during the accident tests was the Okonite/Okalon single-conductor cable. A similar failure mechanism was found, namely splitting and opening of the jacket. On the basis of these findings, the NRC issued Information Notice 92-81, “Potential Deficiency of Electrical Cables With Bonded Hypalon Jackets,” to alert licensees to a potential deficiency in the environmental qualification of electrical cables with bonded jackets. RES was doing additional testing on this and other cable types as part of GSI-168.

Proposed Actions: The action plan is divided into three parallel efforts. Once we get feedback from Okonite and the industry we will determine if any regulatory action is warranted. There are three potential courses of action we may pursue once we have responses from the vendor and the industry:

(1) If only a small number of safety-related equipment items are affected, or only a small number of plants are affected, the staff may address these cases individually.

(2) If the industry initiative sufficiently addresses the issue and several plants are affected, the staff will publish a Regulatory Issue Summary to document the resolution of the issue in accordance with SECY-99-143, “Revisions to Generic Communication Program.”

(3) If the industry initiative is inadequate, the staff may issue a generic letter to nuclear power plant licensees to obtain information on the affected safety-related equipment and plants.

Originating Document: Memorandum from Brian Sheron to Samuel Collins dated May 9, 2000, informing Mr. Collins of the action plan to address the LOCA test failures of Okonite single-conductor bonded jacket cables based on the May 2, 2000, memorandum from Ashok Thadani to Samuel Collins.

Regulatory Assessment: The NRR staff is continuing to work with the vendor, industry, and RES to determine if any regulatory action is warranted. Based on industry statements in previous meetings related to the application and limited use of the subject cable, the staff believes that continued operation of nuclear power plants is warranted while it evaluates the potential deficiency of these cables.

The Code of Federal Regulations (10 CFR 50.49) requires that each item of electric equipment important to safety is qualified for its application, and meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

The staff believes that there is sufficient new information and concerns relative to the operability of Okonite single-conductor bonded jacket cable under design basis conditions to warrant the actions outlined in the action plan dated May 9, 2000.

Current Status: The staff conducted meetings with representatives from Okonite and industry on February 8, and 16, 2000, respectively. By letters dated May 17 and 18, 2000, the staff requested Okonite to evaluate the BNL test report to determine if the test failures represent a deviation or a failure to comply with 10 CFR 21 and, NEI to schedule a meeting to discuss possible options for addressing the issue. At the June 22, 2000, meeting, NEI committed to conduct a survey of all nuclear power plants. The results of the NEI survey will be presented to the staff in a meeting on October 12, 2000.

NRR Technical Contacts: P. Shemanski, DE/EEIB, 415-1377
D. Skeen, DRIP/REXB, 415-1174

RES Technical Contact: S. Aggarwal, DET/MEB, 415-6005

<u>References</u>:

1. Memorandum from Jack Strosnider to Brian Sheron, January 21, 2000.
2. Memorandum from Ashok Thadani to Samuel Collins, May 2, 2000.
3. Memorandum from Brian Sheron to Samuel Collins, May 9, 2000.
4. Letter from Samuel Collins to Okonite, May 17, 2000.
5. Letter from Samuel Collins to NEI, May 18, 2000.
6. Letter Report from BNL on LOCA Test #5, March 26, 2000.
7. Minutes of NRC Meeting on February 8, 2000, with Okonite.
8. Minutes of NRC Public Meeting on February 16, 2000.
9. Minutes of NRC Public Meeting on June 22, 2000.

EMERGENCY ACTION LEVEL GUIDANCE DEVELOPMENT

TAC No.: MA3695	Revision to NESP-007	Last Update: 9/30/00
M98020	Shutdown EAL Guidance	Lead NRR Division: DIPM

EAL GUIDANCE FOR COLD SHUTDOWN, REFUELING AND LONG TERM FUEL STORAGE ("SHUTDOWN EAL GUIDANCE" NEI-99-01)

	MILESTONES	DATE (T/C)
1.	Meet with NEI to resolve staff concerns on NEI's guidance (proposed in NEI-97-03) for EALs applicable in the shutdown mode of operation	1/28/99 (C)
2.	NEI to provide new shutdown EAL guidance (NEI-99-01) for NRC review	4/07/99 (C)
3.	NRC provides comments to NEI on NEI-99-01	5/11/99 (C)
4.	Meet with NEI to discuss comments	5/13/99 (C)
5.	Comments resolved and final draft of NEI-99-01 submitted for endorsement	7/99 (C)
6.	Draft guide developed endorsing NEI-99-01 developed in form of a draft guide for CRGR/ACRS review.	3/6/00 (C)
7.	Determination made on whether to issue a Generic Letter on plant-specific implementation of shutdown EALs - no GL to be issued	8/30/00 (C)
8.	CRGR/ACRS meeting on generic letter - canceled	8/30/00 (C)
9.	Draft Guide issued for public comment	03/22/00 (C)
10.	Public comments addressed (NEI-99-01 revised as needed)	7/14/00 (C)
11.	CRGR/ACRS meeting on final guide NEI 99-01	10/30/00 (T)
12.	Regulatory Guide issued	12/30/00 (T)

Description: This action plan is intended to guide staff efforts to review (and endorse, if appropriate) a revision to industry-developed emergency action level (EAL) guidance. The current industry-developed EAL guidance is contained in NUMARC/NESP-007, Revision 2. The industry is revising this guidance to clarify it based upon lessons-learned from implementation of the existing guidance for EALs and to incorporate new guidance for EALs applicable to (1) the shutdown and refueling modes of reactor operation, (2) permanently defueled plants, and (3) for long-term fuel storage at operating reactor sites.

Historical Background: 10 CFR 50.47(b)(4) and Appendix E to 10 CFR Part 50 require licensees to develop EALs for activating emergency response actions. NUREG-0654/FEMA-REP-1, issued in 1980, provides example initiating conditions for development of EALs [1].

The NRC's evaluation of the 1990 Vogtle Loss Vital AC Power event identified two areas where NRC's EAL guidance and licensee's EAL schemes were deficient: (1) loss of power EALs were ambiguous and (2) EAL guidance for classifying events that could occur in the shutdown mode of plant operations was not available [2]. The NRC's evaluation of shutdown and low power operation in NUREG-1449 also identified a need for guidance for EALs applicable in the shutdown mode of operation [3].

In 1992, the industry issued EAL guidance in NUMARC/NESP-007, Revision 2 [4]. This guidance is more detailed than the guidance provided in NUREG-0654 (e.g., it includes example EALs and bases for the EALs in addition to example initiating conditions) and is based upon 10 years of industry experience in developing EAL schemes. In 1993, the NRC endorsed the industry guidance as an acceptable alternative to the NUREG-0654 guidance in Regulatory Guide 1.101, Revision 3 [5]. The industry guidance addressed the concerns regarding ambiguities in the loss of power EALs and, to a limited degree, addressed concerns with EAL guidance for events initiated in the shutdown mode of operation. However, it was recognized that further guidance for EALs applicable in the shutdown mode was needed.

In September 1997, the Nuclear Energy Institute (NEI) submitted a proposed revision to NUMARC/NESP-007 (issued as NEI 97-03) [6]. This revision provided additional guidance for EALs applicable in the shutdown and refueling modes of plant operation and incorporated a number of improvements and clarifications to the existing EAL guidance in NUMARC/NESP-007. The need for these changes was identified during the development and review of site-specific EAL schemes based on the NUMARC/NESP-007 guidance.

Proposed Actions: Endorse industry-developed EAL guidance in revisions to Regulatory Guide 1.101. Determine whether development of a Generic Letter which requests licensees to incorporate EAL guidance for classifying events initiated in the shutdown and refueling modes of plant operation is warranted. Issue generic letter if it is determined to be warranted.

Originating Documents: Vogtle IIT EDO Staff Action Item 4a [7]
NUREG-1449

Regulatory Assessment: EALs are used to classify events in order to initiate emergency response efforts. Multiple indicators are used in EAL schemes to determine the significance of events. Licensees' current EAL schemes include EALs that can be used to classify events initiated in the shutdown and refueling modes of operation (e.g., radiation monitor-based EALs and judgement EALs). However, guidance is needed to improve licensees' capability (with regard to timeliness and accuracy) for assessing and classifying the significance of events that occur in the shutdown mode of plant operation.

Current Status:
NRC decided not to proceed with endorsing NEI 97-03 because the guidance in NEI 97-03 is subsumed (with minor updates) in NEI 99-01 and NEI 99-01 has proceeded to the point where it should be endorsed by the NRC in the near term (the industry had requested NRC endorsement of NEI 97-03 in order to provide updated EAL guidance as early as possible).

Draft Guide 1075, has been developed to propose guidance on methods acceptable to the NRC staff for complying with the NRC's regulations for emergency response plans and preparedness at nuclear power plants. Guide was issued for public comment on March 22, 2000, in Federal Register, Vol. 65, No. 56 pages 15397 and 15398. Comments were received by May 22, 2000. A public meeting was held on July 14, 2000, to resolve public comments. Comments were resolved and a final Guide has been developed.

The final guide, NEI 99-01, was considered during the 475th meeting of the ACRS and, in a memo to William D. Travers from John T. Larkins dated September 7, 2000, ACRS informed the staff that it had no objection to publication of NEI 99-01.

The staff contacted CRGR regarding issuing NEI 99-01 and in an email dated September 18, 2000, to Joseph Birmingham, the CRGR stated that it did not require a formal briefing on NEI 99-01. The staff is preparing a package for CRGR internal review only.

References:
1. NUREG-0654/FEMA-REP-1, “Criteria for the Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, November 1980.
2. NUREG-1410, “Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990,” June 1990.
3. NUREG-1449, “Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States,” September 1993.
4. NUMARC/NESP-007, Revision 2, “Methodology for Development of Emergency Action Levels,” January 1992.
5. Regulatory Guide 1.101, Rev. 3, “Emergency Planning and Preparedness for Nuclear Power Reactors,” August 1992.
6. Letter from A. Nelson to J. Roe, September 16, 1997.
7. Memorandum from J. Taylor to T. Murley, June 21, 1990.
8. Letter from B. Zalcman to A. Nelson, March 13, 1998.
9. Memorandum from S. Magruder to T. Essig, June 26, 1998.
10. Letter from C. Miller to A. Nelson, August 3, 1998.
11. Letter from A. Nelson to C. Miller, August 13, 1998.
12. Letter from A. Nelson to T. Essig, January 11, 1999.
13. Letter from T. Essig to A. Nelson, May 11, 1999.
14. Memorandum from J. Larkins to W. Travers, June 3, 1999.
15. Memorandum from J. Larkins to W. Travers, September 10, 1999.
16. Letter from J. Birmingham to A. Nelson, August 8, 2000.
17. Memorandum from J. Larkins to W. Travers, September 7, 2000.
18. Email from M. Federline to J. Birmingham, September 18, 2000.

NRR Technical Contacts: P. Milligan, DIPM, 415-2223
L. Lois, DSSA, 415-3233

Lead Project Manager: J. Birmingham, DRIP, 415-2829

ECCS SUCTION BLOCKAGE

TAC Nos. MA6454, MA0704, MA2452, MA4014, MA6204, and MA0698

Last Update: 10/1/2000
Lead NRR Division: DSSA
Supporting Divisions: DE, DIPM, and DET (RES)
GSI: 191

MILESTONES	DATE (T/C)
PART I: **BWR ECCS SUCTION STRAINER CLOGGING ISSUE**	
1. NRCB 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors"	
○ Complete review of licensee responses	3/01 (T)
○ Complete audits of 4-6 plants	8/00 (C)
○ Complete hydrodynamic load review	3/01 (T)
○ Evaluate impact of coatings research on BWR resolution	1/01 (T)
PART II: **NPSH EVALUATIONS**	
1. GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps"	
○ Complete review of licensee responses	3/00 (C)
○ Complete revision of RG 1.1/RG 1.82	12/01 (T)
PART III: **CONTAINMENT COATINGS**	
1. 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"	
○ Complete review of licensee responses	12/99 (C)
○ Complete revision of RG 1.54	7/00 (C)
○ Publish summary of GL responses	12/00 (T)
2. NRC-sponsored research program on the potential for coatings to fail during an accident	
○ Phase I analytical evaluation/coating degradation model	12/98 (C)
○ Phase II test program to validate model and test key parameters	12/00 (T)
○ Evaluate need for regulatory action based on research results	1/01 (T)
PART IV: **GSI 191, "ASSESSMENT OF DEBRIS ACCUMULATION ON PRESSURIZED WATER REACTOR SUMP PERFORMANCE"**	
1. NRC-sponsored research program on the potential for loss of ECCS NPSH during a LOCA due to clogging by debris	
○ Preliminary (qualitative) risk assessment (NRR)	3/99 (C)
○ Complete collection of plant data to support research program	6/99 (C)
○ Integrate industry activities into this Action Plan	4/00 (C)
○ Complete research program on PWR sump blockage (including final risk assessment)	12/01 (T)
○ Evaluate need for regulatory action based on research program results (NRR)	3/02 (T)

Description: This action plan has been prepared to comprehensively address the adequacy of ECCS suction design, and to ensure adequate ECCS pump net positive suction head (NPSH) during a loss-of-coolant accident (LOCA). Specifically, the concern is whether debris could clog ECCS suction strainers or sump screens during an accident and prevent the ECCS from performing its safety function. The plan will be risk informed. For pressurized-water reactors (PWRs), a detailed risk assessment will be conducted when sufficient information is gathered to perform an assessment of the potential for clogging the ECCS sump screens. A preliminary risk assessment has been performed by the staff and the results are discussed below under the Regulatory Assessment. For boiling-water reactors (BWRs), a risk assessment was performed as part of the development of NRC Bulletin (NRCB) 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors" dated May 6, 1996. This risk assessment formed part of the basis for issuing NRCB 96-03.

This plan has four parts. First, for boiling-water reactors (BWRs), this issue has been addressed by licensee responses to NRCB 96-03. The staff is currently confirming the adequacy of the licensee solutions implemented in response to the bulletin; therefore, the staff's confirmatory effort is included in this action plan for completeness. Second, the adequacy of licensee (both PWR and BWR) net positive suction head (NPSH) calculations was evaluated through NRR review of licensee responses to Generic Letter (GL) 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," dated October 7, 1997. The third part of the plan consists of two efforts by the staff. The first effort assessed the adequacy of the implementation and maintenance of current licensee coating programs through NRR review of licensee 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," dated July 14, 1998. The second effort is a research program to assess the potential for coatings to become debris, including the timing of any failures that might occur, and the cause and the characteristics of the debris. These two efforts combined will provide NRR the necessary technical bases on which to assess the potential threat to the ECCS by coating debris and the adequacy of current coating licensing bases (both PWR and BWR). The results of these two programs will also feed into the fourth part of the action plan: an evaluation of the potential for clogging of PWR ECCS recirculation sumps during a LOCA. As with the coating research discussed above, this part of the plan is being conducted by the Office of Nuclear Regulatory Research (RES). RES is evaluating the potential for PWR sumps to become clogged during an accident based on new information learned during the development of NRCB 96-03 for the BWRs.

Historical Background: During licensing of most domestic power plants, consideration of the potential for loss of adequate NPSH due to blockage of the ECCS suction by debris generated during a LOCA was inadequately addressed by both the NRC and licensees. The staff first addressed ECCS clogging issues in detail during its review of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." The NRC staff's concerns related to the potential loss of post-LOCA recirculation capability due to insulation debris were discussed in Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," dated December 3, 1985. This generic letter documented the NRC's resolution of USI A-43. The staff concluded at that time that no new requirements would be imposed on licensees; however, the staff did recommend that Regulatory Guide 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," be used as guidance for the conduct of 10 CFR 50.59 reviews dealing with change out and/or modification of thermal insulation installed on primary coolant system piping and components. NUREG-0897, Revision 1, "Containment Emergency Sump Performance" (October 1985), contained technical findings related to USI A-43, and was the principal reference for developing the revised regulatory guide.

Since the resolution of USI A-43, new information has arisen which challenged the adequacy of the NRC's conclusion that no new requirements were needed to prevent clogging of ECCS strainers in boiling-water reactors (BWRs). On July 28, 1992, an event occurred at Barsebäck Unit 2, a Swedish boiling-water reactor (BWR), which involved the plugging of two containment vessel spray system (CVSS) suction strainers. The strainers were plugged by mineral wool insulation that had been dislodged by steam from a pilot-operated relief valve that spuriously opened while the reactor was at 435 psig. Two of the three strainers on the suction side of the CVSS pumps that were in service became partially plugged with mineral wool. Following an indication of high differential pressure across both suction strainers 70 minutes into the event, the operators shut down the CVSS pumps and backflushed the strainers. The Barsebäck event demonstrated that the potential exists for a pipe break to generate insulation debris and transport a sufficient amount of the debris to the suppression pool to clog the ECCS strainers.

Similarly, on January 16 and April 14, 1993, two events involving the clogging of ECCS strainers occurred at the Perry Nuclear Power Plant, a domestic BWR. In the first Perry event, the suction strainers for the residual heat removal (RHR) pumps became clogged by debris in the suppression pool. The second Perry event involved the deposition of filter fibers on these strainers. The debris consisted of glass fibers from temporary drywell cooling unit filters that had been inadvertently dropped into the suppression pool, and corrosion products that had been filtered from the pool by the glass fibers which accumulated on the surfaces of the strainers. The Perry events demonstrated the deleterious effects on strainer pressure drop caused by the filtering of suppression pool particulates (corrosion products or “sludge") by fibrous materials adhering to the ECCS strainer surfaces. This sludge is typically present in varying quantities in domestic BWRs, since it is generated during normal operation. The amount of sludge present in the pool depends on the frequency of pool cleaning/desludging conducted by the licensee. The effect of particulate filtering on head loss had been previously unrecognized and therefore its effect on PWRs had not been previously considered.

On September 11, 1995, Limerick Unit 1 was being operated at 100-percent power when control room personnel observed alarms and other indications that one safety relief valve (SRV) was open. Attempts by the reactor operators to close the valve were unsuccessful, and a manual reactor scram was initiated. Prior to the opening of the SRV, the licensee had been running the "A" loop of suppression pool cooling to remove heat being released into the pool by leaking SRVs. Shortly after the manual scram, and with the SRV still open, the "B" loop of suppression pool cooling was started. The reactor operators continued their attempts to close the SRV and reduce the cooldown rate of the reactor vessel. Approximately 30 minutes later, operators observed fluctuating motor current and flow on the "A" loop of suppression pool cooling. Cavitation was believed to be the cause, and the loop was secured. After it was checked, the "A" pump was successfully restarted and no further problems were observed. After the cooldown following the blowdown event, the licensee sent a diver into the Unit 1 suppression pool to inspect the condition of the strainers and the general cleanliness of the pool. The diver found that both suction strainers in the "A" loop of suppression pool cooling were almost entirely covered with a thin "mat" of material, consisting mostly of fibers and sludge. The "B" loop suction strainers had a similar covering, but less of it. Analysis showed that the sludge primarily consisted of iron oxides and the fibers were polymeric in nature. The source of the fibers was not positively identified, but the licensee determined that the fibers did not originate within the suppression pool, and contained no trace of either fiberglass or asbestos. This event at Limerick demonstrated the importance of foreign material exclusion (FME) practices to ensure adequate suppression pool and containment cleanliness. In addition, it re-emphasized that materials other than fibrous insulation could clog strainers.

NRCB 96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,” was issued on May 6, 1996, requesting BWR licensees to implement appropriate procedural measures and plant modifications to minimize the potential for clogging of ECCS suction strainers by debris generated during a LOCA. Regulatory Guide 1.82, Revision 2, (RG 1.82), “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” was issued in May 1996 to provide non-prescriptive guidance on performing plant-specific analyses to evaluate the ability of the ECCS to provide long-term cooling consistent with the requirements of 10 CFR 50.46. On November 20, 1996, the Boiling Water Reactor Owners Group (BWROG) submitted NEDO-32686, "Utility Resolution Guidance for ECCS Suction Strainer Blockage" (also known as the URG) to the staff for review. The purpose of the URG is to give BWR licensees detailed guidance for complying with the requested actions of NRCB 96-03. The staff approved the URG in a safety evaluation report (SER) dated August 20, 1998. In response to NRCB 96-03, all affected BWR licensees have installed new large-capacity passive strainers. As noted above, the staff is presently confirming the adequacy of licensee resolutions implemented in response to NRCB 96-03.

RES has begun an evaluation of the potential for PWRs to lose NPSH due to clogging of ECCS sump screens by debris during an accident because of new information learned during the development of NRCB 96-03. As noted above, the effect of filtering of particulates on head loss across the sump screen had previously been unrecognized. In addition, it was also learned that more debris could be generated than was previously assumed, and that the debris would be significantly smaller than was previously expected. With more and finer debris, the potential for clogging of the ECCS sump screen becomes greater leading to the need for the staff to evaluate the potential for clogging of PWR sumps. RES's evaluation will include a risk assessment.

Recent events at a number of plants have raised concerns regarding potential for coatings to form debris during an accident which could clog an ECCS suction. Several cases have occurred where qualified coatings have delaminated during normal operating conditions. Typically, the root cause has been attributed to inadequate surface preparation. This led the staff to raise questions regarding the adequacy of licensee coating programs. The staff issued GL 98-04 to obtain necessary information from licensees to evaluate how they implement and maintain their coating programs. In addition, Regulatory Guide (RG) 1.54 has been revised with the objective to update guidance for the selection, qualification, application, and maintenance of protective coatings in nuclear power plants to be consistent with currently employed ASTM Standards. The endorsement of industry consensus standards is responsive to OMB Circular A-119 and the NRC's Strategic Plan. RES has also begun a research program aimed at providing sufficient technical information regarding the failure of coatings to allow the staff to evaluate the potential for clogging of ECCS suctions by coating debris (or for coatings to contribute to ECCS suction clogging). The program will evaluate the failure modes of coatings, the likely causes, the characteristics (e.g., size, shape) of the debris, and the timing of when coatings would likely fail during an accident. This information will be used to evaluate the ability of the coating debris to transport to the ECCS suction screens or strainers during an accident and the ultimate effect on head loss. The conclusions from the coatings portion of this action plan will be utilized in both RES's assessment of PWR sump clogging and in the staff's confirmatory evaluation of BWR solutions to the strainer clogging issue.

<u>Proposed Actions</u>: This action plan is divided into four parallel efforts. The first effort is for the staff to complete its review of the resolution of NRCB 96-03. Most licensees installed their new strainers under 10 CFR 50.59, concluding that installing the new strainer modification did not constitute an unreviewed safety question. Since the staff did not receive detailed responses from these licensees describing their resolutions, the staff audited 4 plants to determine if any significant issues exist. No significant safety issues were identified. Upon completion of the staff's review of hydrodynamic loads for the General Electric and Mark III strainer designs, the issue will be closed based on the audit findings and the findings of the staff's review of coatings related issues (discussed below).

The second effort was the staff's review of GL 97-04 responses. This review ensured that there are acceptable methods utilized throughout the industry for evaluating NPSH margin. This is important to the ECCS clogging issue because the calculation of adequate NPSH is the ultimate success criteria for determining ability of the ECCS to provide the required flow needed to meet the criteria of 10 CFR 50.46. This review is now complete. A summary of the review results is provided in a memorandum from K. Kavanagh to G. Holahan, "Report on Results of Staff Review of NRC Generic Letter 97-04, 'Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,'" dated June 26, 2000.

The third effort involves the evaluation of coatings as a potential debris source. Concerns raised in this area are due to recent events where qualified coatings have failed during normal operation at a number of sites. The failure of qualified coatings during normal operation has led to two specific staff concerns. The first concern is whether the qualification of coatings is adequate to ensure that coatings do not pose a potential threat to the ECCS. Accordingly, the staff has begun a research effort led by RES to evaluate the potential for coatings to become debris during an accident and consequently, become a threat to the ECCS performing its safety function. The second concern relates to the adequacy of licensee programs to apply and maintain coatings consistent with their licensing bases. This concern will be addressed by NRR staff through review of license responses to GL 98-04. The staff has completed its review of licensee responses to GL 98-04 to determine if licensee coating programs (application and maintenance of protective coatings in containment) are adequate to meet their current licensing bases. A summary report of the staff's findings is being written. This issue is applicable to BWRs and PWRs.

The fourth effort involves an evaluation of PWR sumps based on new information learned during the development of the staff's resolution for NRCB 96-03. RES has begun a program to evaluate PWR sump designs and their susceptibility to blockage by debris. This evaluation will include a detailed risk assessment. Risk insights will be used to support any conclusions drawn relative to the need for licensees to address the potential for ECCS suction clogging.

Support for the research program was needed from the industry to provide RES with the necessary plant data so that RES can bound the problem to be evaluated. The Nuclear Energy Institute (NEI) conducted a survey of PWR licensees and has provided the information needed by RES. The staff will also coordinate its work with industry to eliminate duplication of effort and to ensure effective utilization of resources.

<u>Originating Document</u>: Not Applicable.

<u>Regulatory Assessment</u>: Title 10, Section 50.46 of the *Code of Federal Regulations* (10 CFR 50.46) requires that licensees design their ECCS systems to meet five criteria, one of which is to provide the capability for long-term cooling. Following a successful system initiation, the ECCS shall be able to provide cooling for a sufficient duration that the core temperature is maintained at an acceptably low value. In addition, the ECCS shall be able to continue decay heat removal for the extended period of time required by the long-lived radioactivity remaining in the core. The ECCS is designed to meet this criterion, assuming the worst single failure.

However, for BWRs, experience gained from operating events and detailed analyses (including a detailed risk assessment) demonstrated that excessive buildup of debris from thermal insulation, corrosion products, and other particulates on ECCS pump strainers could occur during a LOCA. This created the potential for a common-cause failure of the ECCS, which could prevent the ECCS from providing long-term cooling following a LOCA. This led to the issuance of NRCB 96-03, and the subsequent installation of new larger strainers by BWR licensees.

The staff believes that there is sufficient new information and concerns raised relative to the potential for debris clogging in PWRs that part of this action plan has been prepared to address PWR sump blockage concerns. However, it is not clear whether a significant threat to PWR ECCS operation exists. The staff believes that continued operation of PWRs is justified because of 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 difficult. In addition, PWRs typically do not need to switchover to recirculation from the sump during a LOCA until 20-30 minutes after the accident initiation allowing time for much of the debris to settle in other places within the containment. Coating debris, in particular, would have plenty of time to settle. Clearly, the results of the staff's research program are needed before a final conclusion regarding the potential to clog the ECCS sump can be reached. In addition to these design considerations, the staff considers continued operation of PWRs to be justified because the probability of the initiating event (i.e., large break LOCA) is extremely low. More probable (although still low probability) LOCAs (small, intermediate) will 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 safely shut the plant down. 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 of sufficient toughness that it will most likely leak (even under safe shutdown earthquake conditions) rather than rupture. This, in turn, would allow operators adequate opportunity to shut the plant down safely (although debris generation and transport for an LBB size through-wall flow will still be investigated). Additionally, the staff notes that there are sources of margin in PWR designs which may not be credited in the licensing basis for each plant. For instance, NPSH analyses for most PWRs do not credit containment overpressure (which would likely 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 example of margin would be that it has been shown, in many cases, that ECCS pumps would be able to continue operating for some period of time under cavitation conditions. Some licensees have vendor data demonstrating this. 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.

GL 97-04 is a review of NPSH calculations. No specific generic concerns were identified in the review of licensee responses.

The Probabilistic Safety Assessment Branch of NRR recently completed a preliminary assessment of the risk associated with the potential clogging of the ECCS sump in PWRs during a LOCA. In a memo from Richard J. Barrett to John N. Hannon dated March 26, 1999, it was concluded that "(d)ue to the unavailability of probabilistic models for debris-induced loss of ECCS NPSH and the plant-specific nature of the sump screen clogging issue, the scope of this risk assessment was limited to assessing the frequency of accident sequences requiring ECCS recirculation to prevent core damage for an average PWR plant. Because the probability and timing of sump screen clogging depends on LOCA size and location, among other parameters, an effort was made to present the results, for each LOCA category, separately.

The following major conclusions were reached by performing this preliminary risk assessment.

1. Results presented in this analysis strongly justify research to re-evaluate the potential for clogging of PWR sump screens by taking into account new information, thus enabling more realistic evaluation and management of associated risks.

2. Continued operation of PWRs is justified because, based on available current information, there is no evidence that the risk associated with the sump clogging issue is high enough to compromise public health and safety."

These conclusions clearly support this action plan as outlined herein.

<u>Current Status</u>: The review of NRCB 96-03 responses is nearly complete. Some plants have not been closed because they are awaiting the results of the staff's hydrodynamic load review for Mark III and GE strainer designs. The staff has completed four audits and written the audit reports for Grand Gulf, Dresden, Duane Arnold. No safety issues have been identified in the audits. GE has provided a second supplement to its strainer topical report, but did not fully address the staff's hydrodynamic load concerns. One open issue remains relating to determining the hydrodynamic mass of the strainers. The staff sent

GE a letter requesting that they perform additional testing to resolve the issue, and GE declined to perform additional tests. The staff is utilizing contract resources to obtain the services of a hydrodynamic load expert to assist in resolving the issue with GE.

NRR review of GL 97-04 responses is complete.

The review of Generic Letter (GL) 98-04 responses is complete pending final closeout by the Lead Project Manager. No significant issues were identified in the review. In addition, RES is continuing its coating research program and is on schedule to complete work by the end of December 2000. Available evidence from limited industry tests of the transport of coating debris indicates that coating debris may not transport very well under conditions approximating those of containment sump flow. In fact, very small amounts of debris actually reached the screens in these tests. This consideration, in addition to the low probability of the initiating event and the difficulty of transporting the debris to the sump given the circuitous geometry of a containment flow path, leads to the conclusion that the current target date for completing the Phase 2 coatings program is acceptable.

RES's PWR sump study is ongoing. No problems have been identified. At the current time, the industry has chosen monitor the NRC's activities in this area rather than conduct any testing or research of their own. The staff will continue to hold regular public meetings with the three PWR owners groups and NEI to keep them informed on the progress of the GSI-191 research program.

NRR Lead PMs:	Donna Skay, PD III-2, 415-1322 (NRCB 96-03, GL 97-04 & PWR Sumps) Bob Pulsifer, PD III-2, 415-3016 (Containment Coatings, GL 98-04)
NRR Lead Technical Reviewer:	Rob Elliott, SPLB, 415-1397
NRR Technical Contacts:	Jim Davis, EMCB, 415-2713 Rich Lobel, SPLB, 415-2865 Kerri Kavanagh, SRXB, 415-3743 Larry Campbell, HQMB, 415-2976 Nicholas Saltos, SPSB, 415-1072
RES Technical Contacts:	Michael Marshall, ERAB, 415-5895 Al Serkiz, ERAB, 415-6563

References:

Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps" (Safety Guide 1), dated November 1970.

Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants" (Draft DG-1076, Proposed Revision 1, published March 1999), dated June 1973.

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

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

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

Generated Debris" dated October 1995.

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

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

Regulatory Guide 1.82, Revision 2, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated May 1996.

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

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," dated July 14, 1998.

Memorandum from Richard J. Barrett to John N. Hannon, "Preliminary Risk Assessment of PWR Sump Screen Blockage Issue," dated March 26, 1999.

Memorandum from K. Kavanagh to G. Holahan, "Report on Results of Staff Review of NRC Generic Letter 97-04, 'Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,'" dated June 26, 2000.

ATTACHMENT 2

GENERIC COMMUNICATION AND COMPLIANCE ACTIVITIES

OCTOBER 2000 DIRECTOR's QUARTERLY STATUS REPORT

Open Generic Communications and Compliance Activities

The attached GCCA portion of the Director's Quarterly Status Report is based on data current as of

October 11, 2000

DIRECTOR's QUARTERLY STATUS REPORT
October 2000

Open Generic Communication and Compliance Activities
Sorted by Lead Technical Division and Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description

Division of Engineering

Electrical & Instrumentation & Controls Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MA7871	IN	DLSkeen	09/30/00		IN: Potential Aging Degradation of Medium Voltage Power Cables	To inform licensees of 4kv power cables that may be performing premature aging degradation.
MA8193	RI	DLSkeen	11/11/11	10/30/2000 T	RIS: LOCA Test Failures of Single Conductor Control Cable	RIS will document industry initiative if issue is found to be generic - decision by June 2000.

EEIB has 2 GCCA(s)

Materials and Chemical Engineering Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MA6018	IN	EJBenner	12/31/00	12/31/2000 T	IN: IN 96-09, Sup 2; Damage in Foreign Steam Generator Internals	To inform licensees of steam generator tube degradation in foreign PWRs as a result of sludge lancing activities.

EMCB has 1 GCCA(s)

Mechanical & Civil Engineering Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MA2314	IN	RABenedict	--/--/--	06/30/2000 L	IN: Potential Inadequacies in the Installation of Check Valves Made by Anderson Greenwood and Borg Warner	Discusses maintenance errors that could make the check valves

EMEB has 1 GCCA(s)

DE has a total of 4 GCCA(s)

Division of Inspection Program Management

QA, Vndr Insp, Maint & Alleg Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
M98441	GL	JWShapaker	04/30/99	10/30/2000 T	GL: Quality Assurance of Electronic Records	In view of technological advancements, changes in NRC regulations, a request was made to update the guidance provided in GL 88-18.

IQMB has 1 GCCA(s)

DIPM has a total of 1 GCCA(s)

Open Generic Communication and Compliance Activities
Sorted by Lead Technical Division and Branch

Division of Regulatory Improvement Programs

Events Assmt, Gen Comms & Non-Power Reactor Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MA5938	LT	CDPetrone	--/--/--	06/30/2001 T	LT: Applying AEOD/RES Reliability Studies to NRR Reactor Programs	LT: Applying AEOD/RES Reliability Studies to NRR Reactor
MA6183	LT	CDPetrone	08/16/99 C	06/30/2001 T	LT: Review RES Reliability Study on High Pressure Injection System,	LT: Review RES Reliability Study on High Pressure Injection System
MA6184	LT	CDPetrone	12/31/99 C	06/30/2001 T	LT: Review RES Component Performance Study on Turbine Driven	LT: Review RES Component Performance Study on Turbine Driven
MA6973	RI	JWShapaker	--/--/--	12/29/2000 T	RIS: Guidance on Sale of Site Land	RIS: Guidance on Sale of Site Land
MA6974	RI	JWShapaker	12/30/00	12/29/2000 T	RIS: Staff Position on Control of Hazard Barriers	RIS: Staff Position on Control of Hazard Barriers
MA7819	RI	JWShapaker	12/30/00	10/30/2000 T	RIS: CRMP Description Location	RIS: CRMP Description Location
MA8485	RI	JWShapaker	--/--/--	12/29/2000 T	RIS: Summary of FFD Program Performance Reports for CY 1998	To provide summary/analysis of licensee data re FFD program performance reports for CY 98 and 99.
MA8652	IN	CDPetrone	--/--/--		IN: High Energy Line Break Issues at Cook	RES wants IN to describe how high energy steam line break could cause both trains of CCW to be inoperable.
MA8818	IN	CVHodge	10/30/00		IN: Potential IN On Experience With Safety Injection Relief Valves	Experience with Safety Injection Relief Valves
MA8819	RI	JWShapaker	--/--/--	12/29/2000 T	RIS: SG Tube Integrity - Industry	To document industry initiative on SG Tube Integrity.
MA9204	IN	CVHodge	--/--/--		IN: Potential IN on Rigging Problems	Rigging Problems
MA9310	RI	JWShapaker	12/31/00	11/17/2000 T	RIS: Eligibility of Operator License Applicants	To solicit (voluntary) updated information on operator licensing exam schedule and estimated number of applicants.
MA9474	RI	JWShapaker	--/--/--	11/30/2000 T	RIS: Procedure for Conducting Meetings with Proprietary Content	To inform industry and other stakeholders re the conduct of meetings with proprietary content.
MA9585	RI	JWShapaker	--/--/--	12/29/2000 T	RIS: ADAMS Electronic Submittal	ADAMS Electronic Submittal
MA9795	IN	EJBenner	--/--/--	11/22/2000 T	IN: Bullet Resistant Material	To alert licensees to faulty stell bullet resistant enclosures.

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MA9821	LT	EFGoodwin	--/--/--	08/22/2001 T	LTF: Westinghouse Spent Fuel Pool/Pellet Density	To follow Westinghouse calculations correcting higher than design pellet density affecting pool criticality.
MA9881	RI	EJBenner	--/--/--	12/05/2000 T	RIS: Indian Point 2 - Steam Generator Tube Failure	To alert addresses to lessons learned from IP2 and ANO SG reviews.
MA9968	RI	JWShapaker	--/--/--	11/30/2000 T	RIS: Grid Reliability and Voltage Adequacy	Presents Status of NRC/Industry Effort on Grid Reliability and Voltage Inadequacy Issue.
MA9992	RI	JWShapaker	--/--/--	12/29/2000 T	RIS: Format and Content of No Significant Hazard	Provides informational guidance to licensees.
MA9993	RI	JWShapaker	--/--/--	12/29/2000 T	RIS: Acceptable Format for Oath or Affirmation	Provides informational guidance to licensees.
MB0015	RI	JWShapaker	--/--/--	11/30/2000 T	RIS: Criterion for Review Under 50.80	Inform licensees of threshold for conducting reviews under 10 CFR 50.80 (Transfer of Licenses) for non-owner operator service companies.
MB0077	RI	JWShapaker	--/--/--		RIS: Changes to Performance Indicators	Pilot effort to evaluate changes to performance indicators; licensee participation in the pilot is voluntary.
MB0149	IN	RABenedict	--/--/--		IN: Palisades Check Valves	Check valve operability

REXB has 23 GCCA(s)

License Renewal & Standardization Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MA9571	RI	JWShapaker	--/--/--	11/17/2000 T	RIS: Advance Notice of Intent to Pursue License Renewal	Voluntary reporting of intent to pursue license renewal to support NRC staff planning.

RLSB has 1 GCCA(s)

DRIP has a total of 24 GCCA(s)

There are a total of 29 GCCA(s)

NOTES:

"--/--/--" for a "TR Comp" date means that at least one reviewer is

"11/11/11" for a "TR Comp" date means that at least one reviewer is constant load

DIRECTOR's QUARTERLY STATUS REPORT
October 2000

Generic Communication and Compliance Activities Added
Since July 21, 2000

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Added
MA9571	RI	JWShapaker	License Renewal & Standardization Branch	--/--/--	11/17/2000 T	RIS: Advance Notice of Intent to Pursue License Renewal	7/25/00: TAC approved by C. Petrone
MA9585	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	12/29/2000 T	RIS: ADAMS Electronic Submittal	7/26/00: TAC approved by C. Petrone
MA9795	IN	EJBenner	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	11/22/2000 T	IN: Bullet Resistant Material	8/22/00: TAC approved by C. Petrone
MA9821	LT	EFGoodwin	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	08/22/2001 T	LTF: Westinghouse Spent Fuel Pool/Pellet Density	8/28/00: TAC approved by C. Petrone
MA9881	RI	EJBenner	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	12/05/2000 T	RIS: Indian Point 2 - Steam Generator Tube Failure	8/30/00: TAC approved by C. Petrone
MA9968	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	11/30/2000 T	RIS: Grid Reliability and Voltage Adequacy	9/13/00: TAC approved by C. Petrone
MA9992	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	12/29/2000 T	RIS: Format and Content of No Significant Hazard	9/18/00: TAC approved by C. Petrone
MA9993	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	12/29/2000 T	RIS: Acceptable Format for Oath or Affirmation	9/18/00: TAC approved by C. Petrone
MB0015	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	11/30/2000 T	RIS: Criterion for Review Under 50.80	9/20/00: TAC approved by C. Petrone

Generic Communication and Compliance Activities Added
Since July 21, 2000

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Added
MB0077	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--		RIS: Changes to Performance Indicators	9/27/00: TAC approved by C. Petrone
MB0149	IN	RABenedict	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--		IN: Palisades Check Valves	9/29/00: TAC approved by C. Petrone

NOTES:

Total Number of Records = 11

"--/--/--" for a "TR Comp" date means that at least one reviewer is

"11/11/11" for a "TR Comp" date means that at least one reviewer is constant load

DIRECTOR's QUARTERLY STATUS REPORT
October 2000

Generic Communication and Compliance Activities Closed
Since July 21, 2000

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Closed
MA6220	RI	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/27/00 P	09/27/2000	RIS: Information Notice on Appendix B to Operating License	9/27/00: TAC cancelled.
MA8165	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	07/03/00 P	07/03/2000	GL: NRC Plans to Update Emergency Telecommunications Systems	This proposed GL was made a RIS under MA 9085. This became RIS 2000-11 issued 6/30/00.
MA8303	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/27/00 P	09/27/2000	RIS: Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff	9/21/00: RIS 2000-17 issued
MA8606	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	08/11/00 P	08/11/2000	RIS: Resolution of GSI B-55, Improved Reliability of Target Rock Safety Relief Valves	8/7/00: RIS 2000-12 issued
MA9084	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	08/25/00 P	08/25/2000	RIS: Annual Report on Effectiveness of Training	8/18/00: RIS 2000-13 issued
MA9089	RI	RABenedict	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/11/00 P	09/11/2000	RIS: Preparation and Scheduling of Operator Licensing Exams	9/6/00: RIS 2000-14 issued
MA9466	IN	EFGoodwin	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/27/00 P	09/27/2000	IN: Non-Vital Bus Fault Leads to Fire and Loss of Offsite Power	9/27/00: IN 2000-14 issued
MA9558	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/08/00 P	09/08/2000	RIS: Availability of the Reactor Vessel Integrity Database Version 2.0.1	9/7/00: RIS 2000-16 issued
MA9601	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/08/00 P	09/08/2000	RIS: Recommendations for Ensuring Cont'd Safe Plant Operation & Minimizing Requests for NOEDs	9/7/00: RIS 2000-15 issued

Generic Communication and Compliance Activities Closed
Since July 21, 2000

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Closed
MA9722	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/25/00 P	09/25/2000	IN: Potential Degradation of Firefighter Primary Protective Garments	9/21/00: IN 2000-12 issued
MA9909	IN	EJBenner	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/27/00 P	09/27/2000	IN: Shutdown Risk Operational Perspective	9/27/00: IN 2000-13 issued
MA9910	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	09/18/00 P	09/18/2000	IN: Licensee Responsibility for QA Oversight of Contractor Activities re Fabrication and Use of Spent Fuel Storage Cask	8/7/00: IN 2000-11 issued
MB0052	IN	DLSkeen	Events Assmt, Gen Comms & Non-Power Reactor Branch	10/02/00 P	10/02/2000	IN: Failure of Non-Intrusive Testing to Detect Failed Check Valve	10/02/00: TAC closed (duplicate TAC, see MB0149)

NOTES:

Total Number of Records = 13

"--/--/--" for a "TR Comp" date means that at least one reviewer is

"11/11/11" for a "TR Comp" date means that at least one reviewer is constant