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ADAMS Accession Number for April 2000 DQSR: ML003716395

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## 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 Cracking and Wolf Creek Drindown event), and issues related to regulatory flexibility and improvements (e.g., New Source Term and Probabilistic Risk Assessment (PRA) Implementation Plan). 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, "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 includes 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**

**APRIL 2000**

**Office of Nuclear Reactor Regulation**

**ATTACHMENT 1**

**NRR ACTION PLANS**

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## BOILING WATER REACTOR INTERNALS

TAC Nos. M94975, M95369, M96219, M96539, M97802, M97803, M97815, M98266, M98880, M99638, M99870, M99894, M99897, M99895, MA1102, MA1104, MA1138, MA1926, MA1927, MA2326, MA2328, MA3395, MA4203, MA4464, MA4465, MA4467, MA4468, MA5057, MA5058, MA6015

Last Update: 03/31/2000  
 Lead NRR Division: DE  
 Supporting Division: DSSA  
 GSI: Not Available

MILESTONES	DATE (T/C) <sup>1</sup>
<b>PART I: REVIEW OF GENERIC INSPECTION AND EVALUATION CRITERIA</b>	
1. Issue summary NUREG-1544 <ul style="list-style-type: none"> <li>○ Update NUREG-1544</li> </ul>	03/96 C 3Q/FY00 T
2. Review BWRVIP Re-inspection and Evaluation Criteria <ul style="list-style-type: none"> <li>○ Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03) . . . . .</li> <li>○ BWRVIP-03, Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components . . . . .</li> <li>○ BWR Vessel Shell Weld Inspection Recommendations (BWRVIP-05) . .</li> <li>○ BWR Axial Shell Weld Inspection Recommendations . . . . .</li> <li>○ Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07) . . . . .</li> </ul>	07/15/99 CA  07/15/99 CA 07/28/98 CA 03/07/00 CA 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) <ul style="list-style-type: none"> <li>○ Safety Assessment of BWR Reactor Internals (BWRVIP-06) . . . . .</li> <li>○ Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08 &amp; BWRVIP-46) . . . . .</li> <li>○ Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14) . . . . .</li> <li>○ Internal Core Spray Piping and Sparger Replacement Design Criteria (BWRVIP-16) . . . . .</li> <li>○ Roll/Expansion of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17) . . . . .</li> <li>○ BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18) . . . . .</li> <li>○ BWRVIP-18, Appendix C, BWR Core Spray Internals Demonstration of Compliance With Technical Information Requirements of License Renewal Rule (10 CFR 54.21) . . . . .</li> </ul>	09/15/98 CA  03/27/98 CA  12/03/99 CA  11/16/98 CI  03/13/98 CD  12/02/99 CA  TBD

<sup>1</sup> CA = Complete, Acceptable (i.e., final SER); CI= Complete, Interim (i.e., draft SER); CD = Complete, Denied

MILESTONES	DATE (T/C) <sup>1</sup>
○ Internal Core Spray Piping and Sparger Repair Design Criteria (BWRVIP-19) . . . . .	11/16/98 CI
○ Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25) . . .	12/19/99 CA
○ 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) . . . . .	04/30/00 T
○ 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) . . . . .	09/16/99 CI
○ BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41) . . . . .	04/30/00 T
○ BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42) . . . . .	06/14/99 CI
○ Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46) . . . . .	03/27/98 CA
○ BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47) . . . . .	04/07/99 CI
○ Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (BWRVIP-48) . . . . .	09/29/99 CA
○ Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49) . . . . .	08/04/98 CA
○ Top Guide / Core Plate Repair Design Criteria (BWRVIP-50) . . . . .	06/28/00 T
○ Jet Pump Repair Design Criteria (BWRVIP-51) . . . . .	06/28/00 T
○ Shroud Support and Vessel Repair Design Criteria (BWRVIP-52) . . . . .	06/28/00 T
○ Standby Liquid Control Line Repair Design Criteria (BWRVIP-53) . . . . .	06/28/00 T
○ Lower Plenum Repair Design Criteria (BWRVIP-55) . . . . .	06/28/00 T
○ LPCI Coupling Repair Design Criteria (BWRVIP-56) . . . . .	06/28/00 T
○ Instrument Penetrations Repair Design Criteria (BWRVIP-57) . . . . .	06/28/00 T
○ CRD Internal Access Weld Repair (BWRVIP-58) . . . . .	06/28/00 T
○ Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals (BWRVIP-59) . . . . .	06/28/00 T
○ BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants (BWRVIP-60) . . . . .	07/08/99 CA
○ Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62) . . . . .	06/28/00 T
○ Shroud Vertical Weld Inspection and Evaluation Guidelines (BWRVIP-63) . . . . .	06/28/00 T

<sup>1</sup> 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 will continue to assess the scopes that have yet to be submitted by licensees concerning inspections or re-inspections of their core shrouds. The staff will also continue to assess core shroud reinspection results and any appropriate core shroud repair designs on a case-by-case basis. The staff will issue separate safety evaluations regarding the acceptability of core shroud reinspection results and core shroud repair designs. 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 53 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.

In October 1995, industry's special review group issued a report (BWRVIP-06) which the NRC staff's preliminary review indicated was not comprehensive. The NRC staff requested additional information which the BWRVIP provided in letters dated December 20, 1996, and June 16, 1997. The staff has completed its review of this submittal. The industry group submitted a report on reinspection of repaired and non-repaired core shrouds (BWRVIP-07) in February 1996. The staff completed its review and issued an SER with several open items. The staff met with the industry to resolve these open items, and completed its final SER. The NRC is also reviewing information submitted by GE on the safety significance of and recommended inspections for top guide and core plate ring cracking. Technical review of the "Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03)" is complete and the staff's SE has been issued.

By letter dated September 20, 1996, the BWRVIP informed the staff of its intention to Petition for Rulemaking to change the augmented inspection requirements contained in 10 CFR 50.55a(g)(6)(ii)(A), in accordance with the recommendations of BWRVIP-05, which would change the inspection requirements from "essentially 100%" of all RPV shell welds to 100% of circumferential welds and 0% of longitudinal welds. Information Notice (IN) 97-63, "Status of NRC Staff's Review of BWRVIP-05," was issued August 7, 1997, to inform the industry of both the status of the staff's review and that the staff would consider technically-justified scheduler relief requests from performing augmented inspections of the RPV shell circumferential welds for 40-months or two operating cycles, whichever was longer. The staff's independent assessment of the BWRVIP-05 report was transmitted by letter dated August 14, 1997, to the BWRVIP, along with a request for additional information and information that needed to be addressed for licensees requesting scheduler relief. The staff has granted such relief requests. The staff briefed the ACRS subcommittee on August 26, 1997, and briefed the full committee on September 4, 1997. The NRC staff has completed its evaluation of the BWRVIP-05 report. IN 97-63, Supplement 1, was issued May 7, 1998, to inform the industry that the staff would continue to consider technically-justified scheduler relief requests from performing augmented inspections of the RPV shell circumferential welds for 40-months or two operating cycles, whichever was longer. A proposed GL informing the industry of the staff's SE was published August 7, 1998 (63 FR 42460). No public comments were received, and the staff issued the final GL (GL 98-05) November 10, 1998.

The staff's initial review of the BWRVIP-14, -18, -25, and -47 is complete and the staff's SEs have been issued. The BWRVIP has responded to the open items in the staff's SEs, and the staff is presently completing its final review of these reports. The staff's initial review of BWRVIP-16, -19, -38, and -42 is complete. The staff is awaiting the BWRVIP's response to the open items in the staff's SEs in order to complete the final review.

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 May 2000. This schedule change is primarily attributed to BWRVIP-41 since the staff is waiting for the BWRVIP to supplement its original submittal. This should be the only outstanding license renewal appendix when the first BWR submits its license renewal application.

The BWRVIP submitted BWRVIP-28 to address the safety implications of recent cracking found in BWR jet pump riser elbows. The staff is reviewing the BWRVIP-28 report. 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. The final SE on BWRVIP-28 is presently in concurrence.

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. The BWRVIP has informed the staff that it plans to revise BWRVIP-07 to ensure that the vertical core shroud welds, and the core shroud repair, is adequately inspected. This report was provided for staff review as BWRVIP-63, dated July 1, 1999.

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. The staff is requesting that the BWRVIP have each BWR licensee confirm to the NRC staff that each BWR licensee is aware of the NRC staff's understanding of these commitments, and recognizes their individual commitment to the BWRVIP program and reports.

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   Amy Cabbage, 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: 03/31/00

Lead Division: DE (#394)

MILESTONE	DATE (T/C)
1. SECY Paper delaying issuance of GL while staff works on industry initiative	10/98(C)
2. NEI submits NEI 97-06	02/04/00 (C)
3. Staff completes review of NEI 97-06 initiative	08/00 (T)
4. Commission Paper	03/30/00 (C)
5. ACRS/CRGR review	10/00 (T)
6. Publish draft SE in FR for public comment	09/00 (T)
7. Staff endorses NEI 97-06 initiative	12/00 (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.

NRR Technical Contacts: Ted Sullivan, EMCB, 415-2796  
James Andersen, EMCB, 415-1437

RES Contact: N/A

## ENVIRONMENTAL SRP REVISION ACTION PLAN Final Update

TAC No. MA0837  
GSI: Not Available

Last Update: 03/31/00  
Lead NRR Division: DRIP

MILESTONES	DATE (T/C)
1. Reflect Potential Impacts and Integrated Impacts in Options for Resolution	
a. Identification of potential impacts	03/96C
b. Identification of integrated impacts	06/96C
c. Proposed options for resolution and develop initial draft of revised ESRP	10/96C
d. Staff/contractor meeting to resolve format and content of revised ESRP	11/96C
2. Prepare Final Draft of ESRP Sections for Public Comment	
a. Draft updated ESRP for staff review	01/97C
b. ACRS and/or CRGR review, if necessary	06/97C
c. Publish (electronic) for public comment	09/97C
3. Disposition Public Comments	02/98C
4. Publish Final NUREG-1555	2/00C
5. Maintenance of program data	Ongoing

**Brief Description:** The Environmental Standard Review Plan (ESRP) Revision Action Plan deals with the revision to NUREG-0555 to reflect changes in the statutory and regulatory arena, to incorporate emerging environmental protection issues (e.g., SAMDA and environmental justice) since originally published in 1979, and to support the review of license renewal applications.

**Regulatory Assessment:** NRR has established the ESRP Update Program for use in the life cycle review of environmental protection issues for nuclear power plants, especially license renewal applications, but also operating reactors, and future reactor site approval applications. The ESRP will reflect current NRC requirements and guidance, consider other statutory and regulatory requirements (e.g., the National Environmental Policy Act, Presidential Executive Orders), and incorporate the generic environmental impact work and plant-specific requirements developed during amending of Part 51 for license renewal reviews.

**Current Status:** The PNNL/NRC staff workshop on the restructured and revised ESRP was held during November 13-14, 1996. Now that the Part 51 rule for license renewal is final, particular emphasis is being placed on assuring that license renewal needs are being addressed in a schedule consistent with the RES regulatory guide and pilot plant application. The results of the November workshop were provided by PNNL in January 1997; followup discussions were held with the contractor through August 1997. The June 1997 draft of the ESRP was forwarded to ACRS for its consideration. In light of the current ACRS schedule, ACRS staff indicated that the ACRS will have no objection to publishing the draft ESRP; the ACRS may request a briefing during the public comment period. The June draft was provided to CRGR for information; the CRGR declined to consider it. Technical editor, legal (OGC), and technical (lead technical branches) comments were received on the July draft in early August and were included in the final draft. The FR notice of availability of Draft NUREG-1555 was published on October 3, 1997; the electronic version (CD and diskette) is available in the PDR and will be made available to the public at no cost. Approximately 300 CDS and 500 hardcopies of the Draft NUREG

were distributed for comment. ACRS discussed the NUREG at its May 1, 1998, meeting; in subsequent interactions with ACRS staff, the Committee determined that it no longer needed a subsequent SAMDA/SAMA briefing on the ESRP or any environmental document prepared by the staff for license renewal unless staff practice changes. During the week of February 9, 1998, the staff developed the comment binning and disposition plan; subsequently, a PNNL/NRC staff workshop was held during February 24-25, 1998, to disposition technical comments and make decisions regarding the organizational structure of the ESRP. A primary concern raised by the public was the consolidation of guidance for the technology area across disparate licensing frameworks (i.e., Parts 50, 52, and 54); the staff restructured the document to segregate guidance into a Part 50/52 ("greenfield"-type review) and that for Part 54 (renewal of a license for an existing facility). This segregation took the form of a supplement to the ESRP and was completed in draft form on July 3, 1998. During December 1-3, 1998, the final PNNL/NRC staff workshop was held to consider how and whether comments raised on the companion RG for license renewal should be dispositioned for the ESRP. The final draft of the ESRP for NRC concurrence was provided by the contractor in February 1999. Based on the reviews done to support license renewal at Calvert Cliffs and Oconee, the staff has finalized its positions on SAMA and the scope of transmission line review and the ESRP will be published in a time frame consistent with Regulatory Guide 4.2, Supplement 1, on the format and content of the environmental report for license renewal. The final ESRP was issued in February 2000. Consequently, this is the last update for the DQSR.

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## EMERGENCY ACTION LEVEL GUIDANCE DEVELOPMENT

TAC No.: MA3695  
M98020

Revision to NESP-007  
Shutdown EAL Guidance

Last Update: 04/17/00  
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	1/28/99C
2. NEI to provide new shutdown EAL guidance (NEI-99-01) for NRC review	4/07/99C
3. NRC provides comments to NEI on NEI-99-01	5/11/99C
4. Meet with NEI to discuss comments	5/13/99C
5. Comments resolved and final draft of NEI-99-01 submitted for endorsement	7/99C
6. Draft guide developed endorsing NEI-99-01 developed in form of a draft guide for CRGR/ACRS review.	3/6/00C
7. Determination made on whether to issue a Generic Letter on plant-specific implementation of shutdown EALs	TBD
8. CRGR/ACRS meeting on draft guide and generic letter	TBD
9. Draft Guide issued for public comment	03/22/00C
10. Public comments addressed (NEI-99-01 revised as needed)	6/00T
11. CRGR/ACRS meeting on final guide and generic letter	8/00T
12. Regulatory Guide and generic letter issued	10/00T

**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 provided comments to NEI on the proposed EAL guidance in a letter dated March 13, 1998 [8]. In meetings held in March and June 1998 [9] the proposed EALs were discussed and the industry provided proposed modifications to the shutdown EALs. The NRC provided comments on the proposed modifications in a letter dated August 3, 1998 [10].

In a letter dated August 13, 1998, NEI proposed an adjustment to the approach for the development of industry EAL guidance [11]. A two-phase approach was proposed. The first phase would focus on incorporating clarifications to the existing guidance. The second phase of the project will produce a new document to be numbered NEI 99-01, "Methodology for Development of Emergency Action Levels for Cold Shutdown, Refueling, and Long Term Fuel Storage," and will provide EAL guidance for cold shutdown and refueling conditions, defueled plants, and dry-cask fuel storage users.

The industry provided the final draft of NEI 97-03 Revision 3 for NRC review and approval in a letter dated January 11, 1999 [12]. The industry provided a draft of NEI 99-01 to the NRC for its review by April 7, 1999. NRC provided comments on NEI 99-01 to NEI in a letter dated May 11, 1999 [13]. In a memorandum dated June 3, 1999, ACRS informed the EDO that it decided not to review NEI 97-03 but would like a briefing on NEI 99-01 [14]. In a letter dated July 27, 1999, NEI submitted NEI 99-01 for NRC endorsement. In a memorandum dated September 10, 1999, ACRS informed the EDO that it decided not to review NEI 97-03 but would like a briefing on NEI 99-01 [15].

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, being 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 was issued for public comment on March 22, 2000, in Federal Register, Vol. 65, No. 56 pages 15397 and 15398. Comments are requested by May 22, 2000.

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.

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## ECCS SUCTION BLOCKAGE

TAC Nos. MA6454, MA0704, MA2452, and MA0698

Last Update: 3/31/2000

Lead NRR Division: DSSA

Supporting Divisions: DE, DRCH, and  
DET (RES)

GSI: 191

MILESTONES	DATE (T/C)
<b>PART I: BWR ECCS SUCTION STRAINER CLOGGING ISSUE</b>	
1. NRCB 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors" <ul style="list-style-type: none"> <li>○ Complete review of licensee responses</li> <li>○ Complete audits of 4-6 plants</li> <li>○ Complete hydrodynamic load review</li> <li>○ Evaluate impact of coatings research on BWR resolution</li> </ul>	10/00 T 7/00 T 3/01 T 1/00 T
<b>PART II: NPSH EVALUATIONS</b>	
1. GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps" <ul style="list-style-type: none"> <li>○ Complete review of licensee responses</li> <li>○ Complete revision of RG 1.1/RG 1.82</li> </ul>	3/00 C 12/01 T
<b>PART III: CONTAINMENT COATINGS</b>	
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" <ul style="list-style-type: none"> <li>○ Complete review of licensee responses</li> <li>○ Complete revision of RG 1.54</li> <li>○ Publish summary of GL responses</li> </ul>	12/99 C 6/00 T 6/00 T
2. NRC-sponsored research program on the potential for coatings to fail during an accident <ul style="list-style-type: none"> <li>○ Phase I analytical evaluation/coating degradation model</li> <li>○ Phase II test program to validate model and test key parameters</li> <li>○ Evaluate need for regulatory action based on research results</li> </ul>	12/98 C 12/00 T 1/01 T
<b>PART IV: GSI 191, "ASSESSMENT OF DEBRIS ACCUMULATION ON PRESSURIZED WATER REACTOR SUMP PERFORMANCE"</b>	
1. NRC-sponsored research program on the potential for loss of ECCS NPSH during a LOCA due to clogging by debris <ul style="list-style-type: none"> <li>○ Preliminary (qualitative) risk assessment (NRR)</li> <li>○ Complete collection of plant data to support research program</li> <li>○ Integrate industry activities into this Action Plan</li> <li>○ Complete research program on PWR sump blockage (including final risk assessment)</li> <li>○ Evaluate need for regulatory action based on research program results (NRR)</li> </ul>	3/99 C 6/99 C 4/00 C 12/01 T 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 is being 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 assesses 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 and issued for comment as DG-1076 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.

Proposed Actions: 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 will audit 4-6 plants to determine if any significant issues exist. If inspection effort is needed to evaluate licensee resolutions, a Temporary Instruction will be prepared. Otherwise, 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 is for the staff to complete its review of GL 97-04 responses. This will ensure 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.

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 will evaluate 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. 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 is also 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) has conducted a survey of PWR licensees to provide 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.

Originating Document: Not Applicable.

Regulatory Assessment: 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 have been identified in the review of licensee responses. Any issues identified with an individual licensee's response will be handled on a case-by-case basis.

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.

**Current Status:** The review of NRCB 96-03 responses is nearly complete. The staff has completed four audits and is currently preparing the audit reports for Grand Gulf and Dresden. The schedule for the audits has been extended four months to provide additional time for the audit reports. The additional time is needed for report preparation because resources were redirected from the audit reports to licensing actions during the last quarter. The revised schedule is considered acceptable because no safety issues were identified during the audits. GE has provided a second supplement to its strainer topical report addressing the staff's hydrodynamic load concerns. However, one open issue remains

relating to determining the hydrodynamic mass of the strainers. The staff is currently evaluating ways to resolve this issue. The hydrodynamic load review schedule has been revised to allow adequate review time for the Mark III hydrodynamic load test program. The report was originally scheduled to be provided to the staff in January/February 2000, but has been delayed until November 2000.

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 conducting its Phase 2 coatings research program. The Phase 2 coatings program was previously extended to December 2000 due to the number of coating systems to be tested and the difficulty in obtaining representative samples of several of the coating systems. The difficulty in obtaining samples arose out of the fact that many of the older coating systems contained asbestos fibers. The coating manufacturers discontinued this practice in the late 1970's/early 1980's, so there is no readily available source for representative samples of these coating systems. Also, the asbestos fibers are not produced in this country anymore. However, a source for asbestos was recently found, and the coatings program is proceeding with no other difficulties at the present time. 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.

**NEW SOURCE TERM FOR OPERATING REACTORS  
Final Update**

TAC No. M89586  
GSI No. 155.1  
CTL I.I

Last Update: 03/31/00  
Lead NRR Division: DSSA  
Supporting Division: DE

MILESTONES	DATE (T/C)
1. NEI Letter	07/94C
2. Commission Memo	09/94C
3. NEI Response	09/94C
4. NEI/NRC Meeting	10/94C
5. Publication of NUREG-1465	02/95C
6. NEI/NRC Meetings	10/94C, 06/95C, 10/95C, 01/96C, 02/96C, 05/96C, 08/96C, 10/96C, 04/97C
7. Submittal of Generic Framework Document (from NEI)	11/95C
8. First Pilot Plant Submittal	12/95C
9. Issue Memo to Commission, Updating Status	08/96C
10. Present Commission Paper in E-Team Briefing	09/96C
11. Brief CRGR on Commission Paper	10/96C
12. Send Commission Paper to EDO/Commission	11/96C
13. Brief ACRS on Commission Paper	11/96C
14. Response to NEI Framework Document	02/97C
15. Begin Pilot Plant Reviews	02/97C
16. Begin Rebaselining	02/97C
17. Brief E-Team on Status of Rebaselining	07/97C
18. Issue User Need for Rulemaking	08/97C
19. Finish Rebaselining	06/98C
20. Finish Rulemaking Plan	06/98C
21. Finish First Pilot Plant Review (Perry)	02/99C
22. Finish Second Pilot Plant Review (Grand Gulf)	09/99C
23. Finish Third Pilot Plant Review (Indian Point Unit 2)	04/29/00T

Description: More than a decade of research has led to an enhanced understanding of the timing, magnitude and chemical form of fission product releases following nuclear accidents. The results of this work has been summarized in NUREG-1465 and in a number of related research reports. Application of this new knowledge to operating reactors could result in cost savings without sacrificing real safety margin. In addition, safety enhancements may also be achieved.

Historical Background: In 1962, the U.S. Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors." Since then licensees and the NRC have used the accident source term presented in TID-14844 in the evaluation of the dose consequences of design basis accidents (DBA).

After examining years of additional research and operating reactor experience, NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," in February 1995. The NUREG describes the accident source term as a series of five release phases. The first three phases (coolant, gap, and early in-vessel) are applicable to DBA evaluations, and all five phases are applicable to severe accident evaluations. The DBA source term from the NUREG is comparable to the TID source term; however, it includes a more realistic description of release timing and composition. Since the NUREG source term results in lower calculated DBA dose consequences, NRC decided not to require current plants to revise their DBA analyses using the new source term. However, many licensees want to use the new source term to perform DBA dose evaluations in support of plant, technical specification, and procedure modifications.

NRC and NEI met several times to discuss the industry's plans to use the new source term. To make efficient use of NRC's review resources, NRC encouraged the industry to approach the issue on a generic basis. The Nuclear Energy Institute (NEI) unveiled its plans for the use of the new source term at operating plants at the Regulatory Information Conference in May 1995. NEI, Polestar (EPRI's consultant), and pilot plant (Grand Gulf, Beaver Valley, Browns Ferry, Perry, and Indian Point) representatives met with NRC staff in June and October 1995 to discuss more detailed plans.

Proposed Actions: The staff proposes to close this action plan item since the staff issued the final rule on the use of alternative source terms at operating reactors in the Federal Register on December 23, 1999, and the rule became effective on January 24, 2000. Included with the Federal Register notice of the final rule were a draft regulatory guide, DG-1081, entitled "Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors," and a draft Standard Review Plan, 15.0.1, entitled "Radiological Consequence Analyses Using Alternative Source Terms." Public comment period on this draft regulatory guide and standard review plan was closed on March 31, 2000. The knowledge gained from the pilot plant application review has been used in developing the draft regulatory guide and draft standard review plan. The staff is in the process of issuing the final regulatory guide and standard review plan (see page 24, NRR Rulemaking: Implementation of Revised Source Term at Operating Reactors.)

Originating Document: EPRI Technical Report TR-105909, "Generic Framework Document for Application of Revised Accident Source Term to Operating Plants," transmitted by letter dated November 15, 1995.

Regulatory Assessment: There will be no mandatory backfit of the new source term for operating reactors. The design-basis accident analyses for current reactors based on the TID-14844 source term are still valid. Therefore, non-urgent regulatory action and continued facility operation are justified.

Current Status: The staff completed its review of the two pilot plant applications in February 1999 for Perry and in August 1999 for Grand Gulf (selective implementation). Issued safety evaluations and license amendments for the Perry plant in March 1999 and for Grand Gulf in September 1999. The staff is currently reviewing the Indian Point Unit 2 pilot plant applications and expects to complete its review by April 29, 2000.

NRR Technical Contact: J. Lee, SPSB, 415-1080

References:

NUREG-1465, "Accident Source Term for Light Water Nuclear Power Plants," February 1995.

July 27, 1994, letter to A. Marion, NEI, from D. Crutchfield, NRC, "Application of New Source Term to Operating Reactors".

September 6, 1994, memorandum to the Commission from NRC staff, "Use of NUREG-1465 Source Term at Operating Reactors".

July 21, 1995, memorandum to the Commission from NRC staff, "Use of NUREG-1465 Source Term at Operating Reactors".

December 22, 1995, pilot plant submittal, letter to Document Control Desk from Tennessee Valley Authority, "Brown's Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specifications (TS) No. 356 and Cost Beneficial Licensing Action (CBLA) 08 - Increase in Allowable Main Steam Isolation Valve (MSIV) Leakage Rate and Request for Exemption from 10 CFR 50, Appendix J... and 10 CFR 100, Appendix A...".

August 9, 1996, memorandum to the Commission from NRC staff, "Use of NUREG-1465 Source Term at Operating Reactors".

November 25, 1996, SECY-96-242, "Use of the NUREG-1465 Source Term at Operating Reactors." February 12, 1997, Staff Requirements Memorandum to SECY-96-242.

February 26, 1997, letter to T. Tipton, NEI, from J. Callan, NRC, responding to the NEI Framework Document.

August 1, 1997, memorandum from D:NRR to D:RES to request that rulemaking be initiated for operating reactor use of updated source term insights.

September 9, 1997, memorandum to the Commission from NRC staff, "Use of NUREG-1465 Source Term at Operating Reactors."

June 30, 1998, memorandum to the Commission from NRC staff, "Rulemaking Plan for Implementation of Revised Source Term at Operating Reactors," SECY-98-158.

June 30, 1998, memorandum to the Commission from NRC staff, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," SECY-98-154.

Summaries of public meetings:

- dated November 10, 1994, for public meeting with NEI held on October 6, 1994;
- dated July 26, 1995, for public meeting with NEI held on June 1, 1995;
- dated November 17, 1995, for public meeting with NEI held on October 12, 1995;
- dated February 1, 1996, for public meeting with NEI held on January 23, 1996;
- dated February 27, 1996, for public meeting with Browns Ferry held on February 7, 1996;
- dated September 27, 1996, for public meeting with Grand Gulf held on August 29, 1996;
- dated October 11, 1996, for public meeting with NEI held on October 2, 1996;
- dated January 24, 1997, for public meeting with Surry held on November 26, 1996;
- dated April 24, 1997, for public meeting with PWR (Surry) held on March 25, 1997;
- dated April 24, 1997, for public meeting with BWR (Grand Gulf) held on March 27, 1997;
- dated May 8, 1997, for public meeting with NEI held on April 2, 1997;
- dated July 28, 1997, for public meeting with PWR (Surry) held on June 18, 1997.

**ATTACHMENT 2**

**GENERIC COMMUNICATION AND COMPLIANCE  
ACTIVITIES**

**APRIL 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

April 27, 2000

# DIRECTOR's MONTHLY STATUS REPORT

April 2000

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

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
<b>Division of Engineering</b>						
<b>Electrical &amp; Instrumentation &amp; Controls Branch</b>						
MA8193	IN	DLskeen	11/11/11	06/30/2000	T IN: LOCA Test Failures of Single Conductor Control Cable	IN: LOCA Test Failures of Single Conductor Control Cable
						<b>EEIB has 1 GCCA(s)</b>
<b>Materials and Chemical Engineering Branch</b>						
M99432	GL	JWShapaker	06/30/00	08/31/2000	T GL: Steam Generator Tube Integrity	Informs licensees that actions beyond current TS requirements may be necessary to ensure steam generator tube integrity.
MA6018	IN	EJBenner	12/31/00	06/30/2000	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.
						<b>EMCB has 2 GCCA(s)</b>
<b>Mechanical &amp; Civil Engineering Branch</b>						
MA2314	IN	RABenedict	--/--/--	06/30/2000	T 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
						<b>EMEB has 1 GCCA(s)</b>
						<b>DE has a total of 4 GCCA(s)</b>
<b>Division of Inspection Program Management</b>						
<b>QA, Vndr Insp, Maint &amp; Alleg Branch</b>						
M98441	GL	JWShapaker	04/30/99	06/30/2000	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.
						<b>IQMB has 1 GCCA(s)</b>
						<b>DIPM has a total of 1 GCCA(s)</b>

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

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
<b>Division of Regulatory Improvement Programs</b>						
<b>Events Assmt, Gen Comms &amp; Non-Power Reactor Branch</b>						
MA5938	LT	RABenedict	--/--	06/30/2000	T LT: Applying AEOD/RES Reliability Studies to NRR Reactor Programs	LT: Applying AEOD/RES Reliability Studies to NRR Reactor
MA6183	LT	RABenedict	08/16/99	C 06/30/2000	T LT: Review RES Reliability Study on High Pressure Injection System,	LT: Review RES Reliability Study on High Pressure Injection System
MA6184	LT	RABenedict	12/31/99	C 06/30/2000	T LT: Review RES Component Performance Study on Turbine Driven	LT: Review RES Component Performance Study on Turbine Driven
MA6220	IN	CDPetrone	--/--	12/30/2000	T IN: Information Notice on Appendix B to Operating License	To inform licensees of their responsibility to meet Federal environmental statutes, regardless of whether or not the requirements are contained in their environmental protection plan (EPP-Appendix B to the operating license).
MA6834	LT	EFGoodwin	--/--		LTF - Sticking Rods In B&W High Burnup Fuel Assemblies	LTF: Sticking Rods in B&W High Burnup Fuel Assemblies
MA6973	RI	JWShapaker	--/--	06/30/2000	T RIS: Guidance on Sale of Site Land	RIS: Guidance on Sale of Site Land
MA6974	RI	JWShapaker	--/--	06/30/2000	T RIS: Staff Position on Control of Hazard Barriers	RIS: Staff Position on Control of Hazard Barriers
MA7819	RI	JWShapaker	05/30/00	06/30/2000	T RIS: CRMP Description Location	RIS: CRMP Description Location
MA8165	RI	JWShapaker	--/--	06/30/2000	T RIS: NRC Plans to Update Emergency Telecommunications Systems	RIS: NRC Plans to Update Emergency Telecommunications Systems

**REXB has 9 GCCA(s)**

**DRIP has a total of 9 GCCA(s)**

**There are a total of 14 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 MONTHLY STATUS REPORT

April 2000

## Generic Communication and Compliance Activities Added Since January 10, 2000

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Added
MA7819	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	05/30/00	06/30/2000	T RIS: CRMP Description Location	1/4/00: TAC approved by C. Petrone
MA7871	IN	DLSkeen	Electrical & Instrumentation & Controls Branch	--/--/--		IN: Potential Aging Degradation of Medium Voltage Power Cables	1/14/00: TAC approved by C. Petrone.
MA8165	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	06/30/2000	T RIS: NRC Plans to Update Emergency Telecommunications Systems	2/10/00: TAC approved by C. Petrone
MA8193	IN	DLSkeen	Electrical & Instrumentation & Controls Branch	11/11/11	06/30/2000	T IN: LOCA Test Failures of Single Conductor Control Cable	2/14/00: TAC approved by C. Petrone

### 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

**Total Number of Records = 4**

# DIRECTOR's MONTHLY STATUS REPORT

**April 2000**

## Generic Communication and Compliance Activities Closed Since January 10, 2000

<b>TAC</b>	<b>Type</b>	<b>Contact</b>	<b>Lead Tech Branch</b>	<b>TR Comp</b>	<b>LA Comp</b>	<b>Title</b>	<b>Reason Closed</b>
MA1538	IN	RABenedict	Reactor Systems Branch	02/15/00 P	02/15/2000	IN: Ice Condenser Operability Assurance Problems	2/15/00: IN cancelled because of its limited application.
MA2124	GL	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	01/01/00 P	01/01/2000	AL: ADAMS Electronic Submittal Capability	1/1/2000: Closed. Issuance date cannot be established.
MA4932	IN	WFBurton		03/31/99 P	03/31/1999	IN: 1998 Enforcement Sanctions as a Result of Deliberate Violations of NRC Employee Protection Requirements	3/19/99: IN 99-06 issued.
MA5050	IN	EJBenner	Reactor Systems Branch	10/27/99 C	02/14/2000	IN: FSAR Documentation regarding fuel leakage	2/14/00: Cancelled - low safety significance of issue
MA5413	IN	JWShapaker	Materials and Chemical Engineering Branch	03/28/00 P	03/28/2000	IN: RCS Chemistry Effects on Flaw Growth Estimates	03/28/00: TAC closed.
MA5734	LT	DLSkeen	Mechanical & Civil Engineering Branch	10/16/99 P	10/16/1999	LT: Review of Draft RES Study on	10/16/99: TAC closed. Memo sent to RES 8/99. Comments were extensive. RES is working on report.
MA6194	IN	TKoshy	Events Assmt, Gen Comms & Non-Power Reactor Branch	03/27/00 C	01/14/2000	IN: Beaver Valley Electrical Transient and Forced Outage	1/14/00: TAC closed.
MA6272	RI	EJBenner	Events Assmt, Gen Comms & Non-Power Reactor Branch	03/24/00 P	03/24/2000	RIS: Generic Standard Technical Specification Adoption	3/20/2000: RIS 2000-06 issued.
MA6972	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	03/24/00 P	03/24/2000	RIS - Resolution of GSI 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions	3/15/2000: RIS 200-03 issued.
MA7110	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	04/22/00 C	02/25/2000	RIS: Closure of GI-23 - RCP Seal Failure	2/15/00: RIS 2000-02 issued.



## Generic Communication and Compliance Activities Closed Since January 10, 2000

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Closed
MA7239	RI	TKoshy	Events Assmt, Gen Comms & Non-Power Reactor Branch	02/24/00 C	02/11/2000	Waterford 3 - RCS Draindown While Shutting Down	2/11/00: TAC closed.
MA7838	RI	JWShapaker	Mechanical & Civil Engineering Branch	03/24/00 P	03/24/2000	RIS: Closeout of GSI 165, Spring Actuated Safety & Relief Valve Reliability	3/16/2000: RIS 2000-05 issued
MA7840	RI	JWShapaker	Gen Issues, Env Finan & Rulemaking Branch	02/11/00 P	02/11/2000	RIS: Change in Status of Foreign Ownership, Control or Domination of Nuclear Reactor Licensees	2/1/2000: RIS 2000-01 issued.
MA7841	RI	JWShapaker	Probabilistic Safety Assessment Branch	04/07/00 P	04/07/2000	RIS: Use of Risk-Informed Decisionmaking in License Amendment Reviews	3/28/2000: RIS 2000-07 issued.
MA8006	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	02/25/00 P	02/25/2000	IN: 1999 Enforcement Sanctions As a Result of Deliberate Violations of NRC Prot Rqmts	2/25/2000: IN 2000-04 issued.
MA8032	IN			03/10/00 C	02/11/2000	IN: Operational Issues Identified in Boiling Water Reactor Trip and Transient	2/11/00: IN 2001-01 issued.
MA8046	RI	JWShapaker	Program Mgmt, Policy Development and Analysis Staff	04/06/00 C	03/24/2000	RIS: OR Licensing Action Estimates	3/16/00: RIS 2000-04 issued.
MA8080	RI	JWShapaker	Inspection Program Branch	04/07/00 P	04/07/2000	RIS: Voluntary Submission of PI Data	3/29/2000: RIS 2000-08 issued.
MA8208	IN	CDPetrone	Electrical & Instrumentation & Controls Branch	04/15/00 C	03/22/2000	IN: Grid Reliability Concerns Regarding Offsite Power Sources	3/27/2000: IN 2000-06 issued.

**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

**Total Number of Records = 19**