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DIRECTOR'S STATUS REPORT

on

GENERIC ACTIVITIES

Action Plans

Generic Communication and Compliance Activities

OCTOBER 2002

Office of Nuclear Reactor Regulation

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 three attachments: 1) action plans, 2) generic communications under development and other generic compliance activities, and 3) risk-informed initiatives table.

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," lists potential generic issues that are safety significant, require technical resolution, and possibly require generic communication or action. The attachment consists of two lists: 1) Open GCCAs and 2) 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.

Attachment 3, “Risk-Informed Initiatives,” contains a table of risk-informed initiatives that the NRR staff are currently working on. The table provides a summary of recent, current, and future activities for each initiative.

ATTACHMENT 1

NRR ACTION PLANS

TABLE OF CONTENTS

BOILING WATER REACTOR INTERNALS

Open TAC Nos.: MA0792, MA1926, MA1927, MA2326, MA2328, MA3673, MA4203, MA4464, MA4465, MA4467, MA4468, MA5012, MA5140, MA7356, MA9111, MB0271

Last Update: 10/09/02
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	2Q/03 (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)
○ 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)	04/10/00 (CA)
○ Technical Basis for Part Circumferential Weld Overlay Repair of Vessel Internal Core Spray Piping (BWRVIP-34)	02/01/02 (CI)

MILESTONES	DATE (T/C)[1]
○ Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)	. . 07/24/00 (CA)
○ BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)	. . 07/24/00 (CA)
○ BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42)	. . 05/26/00 (CA)
○ Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)	. . 05/26/00 (CA)
○ BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47)	. . 03/27/98 (CA)
○ Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (BWRVIP-48)	. . 10/13/99 (CA)
○ Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49)	. . 09/29/99 (CA)
○ Top Guide / Core Plate Repair Design Criteria (BWRVIP-50)	. . . 01/29/01 (CI)
○ Jet Pump Repair Design Criteria (BWRVIP-51)	. . . 10/28/00 (CI)
○ Shroud Support and Vessel Repair Design Criteria (BWRVIP-52)	. . . 11/02/00 (CI)
○ Standby Liquid Control Line Repair Design Criteria (BWRVIP-53)	. . . 10/26/00 (CI)
○ Lower Plenum Repair Design Criteria (BWRVIP-55)	. . . 09/28/01 (CI)
○ LPCI Coupling Repair Design Criteria (BWRVIP-56)	. . . 03/22/02 (CI)
○ Instrument Penetrations Repair Design Criteria (BWRVIP-57)	. . . 05/01/02 (CI)
○ CRD Internal Access Weld Repair (BWRVIP-58)	. . . 10/17/01 (CI)
○ Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals (BWRVIP-59)	. . . 07/31/01 (CI)
○ 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)	. . . 01/30/01 (CI)
○ Shroud Vertical Weld Inspection and Evaluation Guidelines (BWRVIP-63)	. . . 04/18/00 (CI)
○ BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)	. . 07/27/01 (CA)
○ Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)	. . . 09/15/00 (CI)
○ BWR Core Shroud Inspection & Flaw Evaluation Guidelines (BWRVIP-76)	. . . 12/31/02 (T)
○ BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material (BWRVIP-78)	. . 02/01/02 (CA)
○ Evaluation of Crack Growth in BWR Shroud Vertical Welds (BWRVIP-80)	. . . 12/31/02 (T)
○ Evaluation of Guidelines for Selection and Use of Material for Repairs to BWR Internals (BWRVIP-84)	. . . 10/30/02 (T)
○ Evaluation of Guidelines for Selection and use of Materials for Repairs to BWR Internals (BWRVIP-86)	. . 02/01/02 (CA)

[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 over 50 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 2002. The schedule change for BWRVIP-76 is due to the staff waiting for the BWRVIP to supplement its original submittal in accordance with the open items in the staff's initial SE.

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:	Meena Khanna, EMCB, 415-2150 Jai Rajan, EMEB, 415-2788
NRR Lead PM:	Meena Khanna, EMCB, 415-2150
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.	Description	
M88885	Steam Generator (SG) Integrity Rulemaking	Last Update: 10/04/02
M99432	GL: SG Tube Integrity	Lead Division: DLPM
MA4265	NEI 97-06	Supporting Divisions: DE, DIPM, DSSA
MA5037	SG Action Plan	Supporting Office: RES
MA5260	DPO on SG Issues	
MA7147	GSI-163	
MA9881	Regulatory Issue Summary - IP2 SG Tube Failure	
MB0258	SG Action Plan Administration	
MB0553	SG Inspection Program	
MB0576	Licensee SG Inspection Results Summary Reports & SG Tube Integrity Amendment Review Guidance	
MB0631	SG Workshop	
MB0633	OL No. 803 Revisions per SG Action Plan	
MB0737	IIPB SG Action Plan Activities	
MB3794	SG Communication Plan	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.1 (MA9881)	Issue Regulatory Information Summary on SG Lessons Learned (TG: 8; page 2 of Ref. 2)	11/03/00 (C) ML010820457	DE E. Murphy	
1.2 (MA4265)	Discuss steam generator action plan and IP2 lessons learned with industry and other external stakeholders (TG: 2a-2o, 3a, 3b, 4a, 4b , 4c, 8)	12/20/00 (C) ML010820457	DE T. Sullivan R. Rothman	
1.3 (MB0258)	Subsequent to item 2, identify technical and management leads for each item and develop initial resource estimates	12/27/00 (C) ML010820457	DLPM R. Ennis	DE K. Karwoski DIPM D. Coe
1.4 (MB0258)	Brief management on resource estimates and invoke PBPM process as appropriate	12/27/00 (C) ML010820457	DLPM R. Ennis	DE K. Karwoski DIPM D. Coe

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.5 (MA5260)	Staff review of ACRS recommendations on DPO and develop detailed milestones and evaluate impact on other action plan milestones. Invoke PBPM process, as appropriate. (GSI-163 and DPO)	05/11/01 (C) ML011720125 ML011300073	DLPM R. Ennis	DE S. Coffin E. Murphy DSSA S. Long RES J. Muscara
1.6 (MA7147)	Determine GSI-163 resolution strategy and revise steam generator action plan milestones, as appropriate (GSI-163)	05/11/01 (C)	DE E. Murphy	
1.7 (MB0553)	Determine need to incorporate new steam generator performance indicators into Reactor Oversight Process (page 2 of Ref. 2; TG: 5e, 5f)	01/24/01 (C) ML010820457	DIPM D. Hickman	DE C. Khan E. Murphy DSSA S. Long
1.8 (MA4265)	Recommence work on NEI 97-06 (page 3 of Ref. 2; TG: 7)	01/31/01 (C) ML010820457	DE E. Murphy	
1.9 (MB0553)	Review NRC inspection program and, if necessary, revise guidance to inspectors on overseeing facilities with known steam generator tube leakage. (Attachment 3 to Ref. 1)	03/30/01 (C) ML010920112	DE L. Lund	DIPM DSSA S. Long
1.10 (MB0576)	Reassess the NRC treatment of licensee steam generator inspection results summary reports and conference calls during outages. Evaluate need for review guidance. (Attachment 3 to Ref. 1; TG: 6c; page 4 and 5 (top and bottom) of Ref. 1)	04/30/01 (C) ML011220621 ML013020093	DE S. Coffin	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.11 (MB0553)	Review the NRC inspection program and, if necessary, revise guidance to inspectors on overseeing facility eddy current inspection of steam generators. This involves the following major substeps: a) review and revise the baseline inspection program.	 04/30/01 (C) ML011210293	 DE C. Khan	 DIPM DSSA S. Long
	b.1) review how ISI results/degraded conditions should be assessed for significance by a risk-informed SDP and define needed revisions to the SDP	09/21/01 (C) ML012680252	DSSA S. Long	DE C. Khan DIPM P. Koltay
	b.2) develop and issue draft revision of risk-informed SDP using information identified in b.1 above	02/21/02 (C) ML020730318	DIPM P. Koltay	DSSA S. Long DE C. Khan
	c) review and revise the training program for inspectors	ML020560366 ML012970361	DIPM E. Kleeh	DE C. Khan
	c.1) Provide IP training material to Regions	10/11/01 (C)		
	c.2) Formal training to inspectors	02/01/02 (C)		
	(Attachment 3 to Ref. 1; TG: 5a, 5b, 5c, 5d, 5f, 6c)			
1.12 (MB0576)	Determine need for formal written guidance for technical reviewers to utilize in performing steam generator tube integrity license amendment reviews (TG: 5c, 6a)	04/30/01 (C) ML011220621	DE S. Coffin	
1.13 (MB0258)	Staff provides EDO with update on status of action plan (page 8 of Ref. 1)	05/17/01 (C) ML011720125	DLPM R. Ennis	
1.14 (MA4265)	Staff completes review and prepares draft safety evaluation of NEI 97-06 including addressing issues raised in OIG report and IP2 lessons learned report (NEI 97-06, TG: 2, 3, 4, 7)	TBD (T) Note 12	DE E. Murphy	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.15 (MB0631)	Hold steam generator workshop with stakeholders (page 2 of Ref. 1; page 2 of Ref. 2)	02/27/01 (C) ML010820457	DE R. Rothman	
1.16 (MA4265)	Staff briefs CRGR on NEI 97-06 (NEI 97-06)	TBD (T) Note 12	DE E. Murphy	
1.17 (MA4265)	Publish SE on NEI 97-06 in FR for public comment (NEI 97-06)	TBD (T) Note 12	DLPM M. Banerjee	
1.18 (MA4265)	ACRS review of NEI 97-06 (NEI 97-06)	TBD (T) Note 12	DE E. Murphy	
1.19 (Later)	Issue generic communication related to steam generator operating experience and status of steam generator issues	10/31/01 (C) ML020230299	DE Z. Fu	
1.20 (MA4265)	Staff briefs Commission on endorsing NEI 97-06 (NEI 97-06, and WITS Item 199400048)	TBD (T) Note 12	DE L. Lund	
1.21 (MA4265)	Staff issues endorsement package on NEI 97-06 in a safety evaluation and includes the approval of the generic technical specification change in a Regulatory Issue Summary	TBD (T) Note 12	DE E. Murphy	
2.1	Evaluate the need for a new communication protocol with the U.S. Secret Service that would cover emergency situations at all NRC licensed facilities (Attachment 3 of Ref. 1)	12/05/00 (C) ML010460485 ML010820457	IRO F. Congel	
2.2 (MB0258)	Establish NRC web site for Steam Generator Action Plan	01/16/01 (C) ML010820457	DLPM R. Ennis	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
2.3 (MB0258)	Review and revise, as appropriate, the policy for project manager involvement with the morning call between the resident inspectors and the region. (Attachments 3 and 4 of Ref. 1)	03/23/01 (C) ML011020026	DLPM R. Ennis	
2.4 (MB0737)	Review program requirements for routine communications between the resident inspectors and local officials based on public interest. Based on weighing current resident inspector responsibilities (e.g., inspection requirements, following up on plant events) against this review, revise program requirements if needed. (Attachment 3 of Ref. 1)	04/03/01 (C) ML010890426	DIPM T. D'Angelo	
2.5 (MB0737)	Develop, revise, and implement, as appropriate, a process for the timely dissemination of technical information to inspectors for inclusion in the inspection program (TG: 5g)	04/03/01 (C) ML010890426	DIPM G. Klinger	
2.6 (MB0258)	Incorporate experience gained from the IP2 event and the SDP process into planned initiatives on risk communication and outreach to the public (TG: 9) 3. Issue NRR input for incorporation into OEDO initiative 4. Address SRM dated 12/26/01	 01/31/02 (C) ML020590125 12/31/02 (T)	PMAS M. Kotzalas	
2.7 (MB0258)	Investigate possibility of establishing protocol with OIG regarding review of draft reports for factual/contextual errors (page 8 of Ref. 1)	06/18/01 (C) ML011720125	DLPM R. Ennis	
2.8 (MB0633)	Review and revise, as appropriate, the amendment review process, including concurrence responsibilities, supervisory oversight, and second-round requests for additional information.			

Item No. (TAC No.)	Milestone	Date **(T=Target) (C=Complete)**	Lead	Support
2.8 (continued)	a. Issue OI LIC-101 b. Issue procedure for NRR and RES interactions (Attachment 3 of Ref. 1; TG: 6b, 6d, 6e; page 6 of Ref. 1)	08/31/01 (C) 02/27/02 (C) ML020580484	DLPM M. Banerjee DLPM M. Fields	
3.1	In order to address ACRS comments on current risk assessments, develop a better understanding of the potential for damage progression of multiple steam generator (SG) tubes due to depressurization of the SGs (e.g., during a main steam line break (MSLB) or other type of secondary side design basis accident). (Pgs. 46, 8-12) (See Notes 4, 5, and 6) Specific tasks include:			
	a) Perform thermal-hydraulic (T-H) calculations and sensitivity studies using the 3-D hydraulic component of TRAC-M to assess the loads on the tube support plate and SG tubes during main steam line break (MSLB). Perform sensitivity studies on code and model parameters including numerics. Develop conservative estimate of loads and evaluate against similar analyses.	12/31/02 (T)	RES J. Uhle	DSSA W. Jensen
	b) Perform T-H assessment of flow-induced vibrations during MSLB. Using the T-H conditions calculated during the transient, generate a conservative estimate of flow-induced vibration displacement and frequency assuming steady state behavior.	12/31/02 (T)	RES J. Uhle	DSSA W. Jensen

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.1 (continued)	c) Perform additional sensitivity studies as needed.	06/30/03 (T)	RES J. Uhle	SSA W. Jensen
	d) Obtain information from existing analyses related to loads and displacements (axial, bending, cyclic) experienced by SG structures under MSLB conditions.	12/31/02 (T)	RES J. Muscara	
	e) Using information from tasks 3.1a, 3.1b, and 3.1d, estimate upper bound loads and displacements.	12/31/02 (T)	RES J. Muscara	DE E. Murphy
	f) Estimate crack growth, if any, for a range of crack types and sizes using bounding loads from task 3.1e in addition to the pressure stresses. Include the effects of TSP movement in these evaluations and any effects from cyclic loads.	12/31/02 (T)	RES J. Muscara	DE E. Murphy
	g) Estimate the margins to crack propagation for a range of crack sizes for MSLB types loads and displacements in addition to the pressure stress.	12/31/02 (T)	RES J. Muscara	DE E. Murphy
	h) Based on the margins calculated in task 3.1g over and above the bounding loads, decide if more refined TH analyses need to be conducted to obtain forces and displacements of structures under MSLB conditions.	12/31/02 (T)	RES J. Muscara	DE E. Murphy

Item No. (TAC No.)	Milestone	Date **(T=Target) (C=Complete)**	Lead	Support
3.1 (continued)	I) Conduct tests of degraded tubes under pressure and with axial and bending loads to validate the analytical results from above tasks.	06/30/03 (T)	RES J. Muscara	DE E. Murphy
	j) Conduct analyses similar to above with refined load estimates if necessary.	06/30/04 (T)	RES J. Muscara	DE E. Murphy
	k) Use information developed in tasks 3.1a through 3.1j to evaluate the conditional probabilities of multiple tube failures for appropriate scenarios in risk assessments for SG tube alternate repair criteria (ARC).	02/28/05 (T)	DSSA S. Long	DE E. Murphy RES J. Muscara H. Woods
3.2	Confirm that damage progression via jet cutting of adjacent tubes is of low enough probability that it can be neglected in accident analyses. (P.s. 10-11) (See Notes 3 and 5) Specific tasks include:			
	a) Complete tests of jet impingement under MSB conditions.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
	b) Conduct long duration tests of jet impingement under severe accident conditions.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
	c) Document results from tasks 3.2a and 3.2b.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
3.3	When available, use data from the ARTIST program (planned in Switzerland) to develop a better model of the natural mitigation of the radionuclide release that could occur in the secondary side of the SGs. (P.s. 12-13) (See Notes 3 and 5)	09/30/04 (T) See Note 2	RES R. Lee	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4	In order to address ACRS criticism of current risk assessments, develop a better understanding of RCS conditions and the corresponding component behavior (including tubes) under severe accident conditions in which the RCS remains pressurized. (P.s. 46-47, 12-15) (See Notes 3 and 5) Specific tasks include:			
	a) Perform system level analyses to assess the impact of plant sequence variations (e.g., pump seal leakage and SG tube leakage).	09/28/01 (C) ML012720004	RES C. Tinkler	DSSA W. Jensen S. Long
	b.1) Re-evaluate existing system level code assumptions and simplifications.	04/12/02 (C)	RES D. Bessett	DSSA W. Jensen S. Long
	b.2) Following the results from 3.4.a and 3.4.b.1, perform additional analysis to: include modeling of heat transfer enhancement from radiation heat transfer in the hot leg and steam generator; suppress unphysical numerically driven flows in the calculations; and investigate the sensitivity of calculated results to bypass flows.	03/31/03 (T)	RES D. Bessett	DSSA W. Jensen
	c) Examine 1/7 scale data to assess tube to tube temperature variations and estimate variations for plant scale.	08/31/02 (C)	RES D. Bessett	DSSA W. Jensen S. Long
	d) Perform more rigorous uncertainty analyses with system level code to address inlet plenum mixing by developing distribution functions for mixing parameters based on available data. Peer review.	12/31/02 (T)	RES D. Bessett	DSSA W. Jensen S. Long
	e) Examine SG tube severe accident T-H conditions using computational fluid dynamics (CFD) methods. This			

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	e.1) Benchmark CFD methods against 1/7 scale test data.	08/31/01 (C) ML012750061	RES C. Boyd	DSSA W. Jensen S. Long
	e.2) Perform full scale plant calculations (hot leg and SG) for a 4 loop Westinghouse design. Evaluate scale effects.	03/28/02 (C)	RES C. Boyd	DSSA W. Jensen S. Long
	e.3) Perform plant analysis to address the effects on inlet plenum mixing resulting from tube leakage and hot leg orientation (CE design impact).	12/30/02 (T)	RES C. Boyd	DSSA W. Jensen S. Long
	f) Examine the uncertainty in the T-H conditions associated with core melt progression.	01/31/03 (T)	RES D. Bessett	DSSA W. Jensen S. Long
	g) Perform experiments to develop data on inlet plenum mixing impacts due to SG tube leakage and hot leg/ inlet plenum configuration.	03/31/03 (T)	RES D. Bessett	DSSA W. Jensen S. Long
	h) Perform a systematic examination of the alternate vulnerable locations in the RCS that are subject to failure due to severe accident conditions. This includes the following:			
	h.1) Evaluate the creep failure of primary system passive components such as pressurizer surge line and the hot leg taking into account the material properties of the base metal, welds, and heat affected zones in the presence of residual and applied stresses, in addition to the pressure stress, and the presence of flaws.	11/30/03 (T)	RES J. Muscara	DE E. Murphy DSSA S. Long
	h.2) Evaluate the failure of active components such as PORVs, safety valves, and bolted seals based on operability and "weakest link" considerations for these components.	11/30/03 (T)	RES J. Muscara	DE E. Murphy DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	h.3) Conduct large scale tests if needed. i) Develop data and analyses for predicting leak rates for degraded tubes in restricted areas under design basis and severe accident conditions. j) Put the information developed in task 3.4i into a probability distribution for the rate of tube leakage during severe accident sequences, based on the measured and regulated parameters for ARCs applied to flaws in restricted places (e.g., drilled-hole TSPs and the unexpanded sections of tubes in tube sheets). k) Integrate information provided by tasks 3.4a through 3.4j and 3.5 to address ACRS criticisms of risk assessments for ARCs that go beyond the scope and criteria of GL 95-05 (e.g., ARCs that credit "indications restricted against burst") as well as dealing with other SG tube integrity and licensing issues (e.g., relaxation of SG tube inspection requirements).	11/30/05 (T) 12/31/03 (T) 06/30/04 (T) 02/28/05 (T)	RES J. Muscara RES J. Muscara DSSA S. Long DSSA S. Long	DE E. Murphy DSSA S. Long DSSA S. Long DE E. Murphy DE E. Murphy RES J. Muscara DE E. Murphy RES J. Muscara D. Bessett H. Woods
3.5	Develop improved methods for assessing the risk associated with SG tubes under accident conditions. (P.s. 47, 16-20) (See Note 5) Specific tasks include: a) Development of an integrated framework for assessing the risk for the high-temperature/high-pressure accident scenarios of interest.	04/01/02 (C) ML020910624	RES H. Woods	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.5 (continued)	b) Development of improved methods for identifying accident scenarios (including MSB) that lead to challenges on the reactor coolant pressure boundary. c) Development of improved PRA models of the scenarios identified above, including the impact of operator actions and appropriate treatment of uncertainty.	06/28/03 (T) 06/28/03 (T)	RES H Woods RES H. Woods	DSSA S. Long DSSA S. Long
3.6	To address an ACRS report conclusion that improvements can be made over the current use of a constant probability of detection (POD) for flaws in SG tubes, RES has recently completed an eddy current round robin inspection exercise on a SG mock-up as part of NRC's research to independently evaluate and quantify the inservice inspection reliability for SG tubes. This research has produced results that relate the POD to crack size, voltage, and other flaw severity parameters for stress corrosion cracks at different tube locations using industry qualified teams and procedures. Complete analysis of research results and prepare topical report to document the results. (P.s. 47, 33)	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
3.7	Assess the need for better leakage correlations as a function of voltage for 7/8" SG tubes. (P.s. 48, 28-29) (See Note 5)	04/30/03 (T)	DE E. Murphy	RES J. Muscara
3.8	Develop a program to monitor the prediction of flaw growth for systematic deviations from expectations. (Pg. 48) (See Note 5)	1/3/02 (C)	DE J. Tsao	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.9	Develop a more technically defensible position on the treatment of radio nuclide release to be used in the safety analyses of design basis events. (P.s. 48, 38-44) (See Note 5) Specific tasks include: a) Assess Adams and Atwood and Adams and Sattison spiking data with respect to the ACRS comments. b) Based upon the assessment performed in task 3.9a, develop a response to the ACRS comments. c) Publish in the *Federal Register* for public comment, the response to ACRS' comments. d) Complete review of public comments. e) Based upon task 3.9d, determine if additional work needs to be performed.	 08/09/01 (C) 11/30/02 (T) Note 11 12/31/02 (T) Note 11 03/15/03 (T) Note 11 05/15/03 (T) Note 11	DSSA J. Hayes	
3.10	To address concerns in the ACRS report regarding our current level of understanding of stress corrosion cracking, the limitations of current laboratory data, the difficulties with using the current laboratory data for predicting field experience (crack initiation, crack growth rates), and the notion that crack growth should not be linear with time while voltage growth is, the following tasks will be performed: (P.s. 20-29) (See last sentence in Note 3) Specific tasks include:			

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.10 (continued)	a) Conduct tests to evaluate crack initiation, evolution, and growth. Tests to be conducted under prototypic field conditions with respect to stresses, temperatures and environments. Some tests will be conducted using tubular specimens. b) Using the extensive experience on stress corrosion cracking in operating SGs, and results from laboratory testing under prototypic conditions, develop models for predicting the cracking behavior of SG tubing in the operating environment. c) Based on the knowledge accumulated on stress corrosion cracking behavior and the properties of eddy current testing, attempt to explain the observed relationship between changes in eddy current signal voltage response and crack growth.	12/31/05 (T) 12/31/06 (T) 12/31/05 (T)	RES J. Muscara RES J. Muscara RES J. Muscara	DE E. Murphy DE E. Murphy DE E. Murphy
3.11	In order to resolve GSI 163, it is necessary to complete the work associated with tasks 3.1 through 3.5 and 3.7 through 3.9. Upon completion of those tasks, develop detailed milestones associated with preparing a GSI resolution document and obtaining the necessary approvals for closing the GSI, including ACRS acceptance of the resolution. (See note 9)	12/31/05 (T)	DLPM J. Zimmerman	DE E. Murphy DSSA S. Long
3.12	Develop outline and a detailed schedule for completing DG 1073, “Plant Specific Risk-Informed Decision Making: Induced SG Tube Rupture (See note 9)	12/31/05 (T)	DE E. Murphy	DSSA S. Long

Notes:

1. For SG Action Plan milestones associated with the SG DPO (i.e., Item Nos. 3.1 - 3.11), the page numbers referenced in the milestone description indicate the source of the milestone as described in ACRS Report NUREG-1740, "Voltage-Based Alternative Repair Criteria." The ACRS report was included as an enclosure to a memorandum from D. Powers to W. Travers dated February 1, 2001 (Accession No. ML010780125).

2. With respect to milestone Item No. 3.3, the ARTIST program plan is being finalized for implementation. A firm testing schedule is not currently available but testing is expected to commence in 2003. Finalizing the agreement with the participants (including NRC) is taking longer than expected.

3. The work described in this milestone is related, in part, to previously planned work associated with an NRR User Need request dated February 8, 2000 (Accession No. ML003682135), and the associated RES response to the request dated September 7, 2000 (Accession No. ML003714399). In addition, portions of this work were undertaken on an anticipatory basis by RES.

4. The work described in this milestone is related, in part, to previously planned work associated with GSI 188, "Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass."

5. The work described in this milestone is related, in part, to previously planned work associated with GSI 163, "Multiple Steam Generator Tube Leakage."

6. The thermal-hydraulic analyses (items 3.1a through 3.1c) will provide input into the tube integrity analyses (items 3.1d through 3.1j) on an on-going basis. The end dates for these two areas coincide because of the close integration between these two RES efforts. Also, the end dates reflect the target date for the final report documenting the RES findings.

7. Item Nos. 1.1 through 2.8 in the above table were developed from Attachment 1 of a memorandum from J. Zwolinski, J. Strosnider, B. Boger and G. Holahan to B. Sheron and R. Borchardt dated March 23, 2001 (Accession No. ML010820457). That memorandum provided a revision to the Steam Generator Action Plan that was originally issued via a memorandum from B. Sheron and J. Johnson to S. Collins dated November 16, 2000 (Accession No. ML003770259).

8. Item Nos. 3.1 through 3.11 in the above table were developed from Attachment 1 of a memorandum from S. Collins and A. Thadani to W. Travers dated May 11, 2001 (Accession No. ML011300073). That memorandum provided a revision to the Steam Generator Action Plan as requested by a memorandum from W. Travers to S. Collins and A. Thadani dated March 5, 2001 (Accession No. ML010670217).

9. The completion date assumes need for large scale test.

10. The ADAMS accession no. listed under "Date" is the closure document.

11. Limited staff resources has necessitated that the focus be placed upon those activities associated with implementation of the alternate source term, power uprates, amendments involving containment equipment hatches being open during refueling operations, the draft generic letter on control room habitability, and the four draft regulatory guides associated with control room habitability. This re-prioritization of work necessitated delaying the affected SGAP milestones.

12. The target dates assume industry submittal of an acceptable generic license change package for NRC review and approval by 6/30/02. However, the industry submittal schedule was impacted after staff clarified the NRC policy and applicability of the license amendment process to NEI 97-06. The 6/30/02 schedule was not met by the industry. At a 9/11/02 public meeting with the staff, the industry indicated that they intended to make a steam generator license amendment submittal for a lead plant by the end of the year thus putting the resolution of the remaining issues through a formal regulatory process. The staff expects a minimum of six months slide in the schedule, however needs to ensure that the lead plant submittal is of acceptable quality before this schedule can be established.

Description: Steam generator tube integrity issues continue to arise. As a result, many organizations within the NRC have evaluated portions of the regulatory process associated with steam generator tube integrity and have made some insightful observations and/or recommendations. To ensure safety from a steam generator tube integrity standpoint is maintained, that public confidence in the steam generator tube integrity area is improved, and the NRC and stakeholder resources are effectively and efficiently utilized, the steam generator action plan was developed. The action plan is intended to direct and monitor the NRC's effort in this area and to ensure the issues are appropriately tracked and dispositioned. The action plan is also intended to ensure the NRC's efforts result in an integrated steam generator regulatory framework (license review, inspection and oversight, research, etc.) which is effective, efficient, and realistic.

This plan consolidates numerous activities related to steam generators including: 1) the NRC's review of the industry initiative related to steam generator tube integrity (i.e., NEI 97-06); 2) GSI-163 (Multiple Steam Generator Tube Leakage); 3) the NRC's Indian Point 2 (IP2) Lessons Learned Task Group recommendations; 4) the Office of the Inspector General (OIG) report on the IP2 steam generator tube failure event; and 5) the differing professional opinion (DPO) on steam generator issues. The plan does not address plant-specific reviews or industry proposed modifications to the Generic Letter 95-05 (voltage-based tube repair criteria) methodology. The plan also includes non-steam generator related issues that arose out of recent steam generator related activities (e.g., Emergency Preparedness issues from the OIG report). The milestone table shown above is organized as follows:
- Item Nos. 1.1 through 1.21: SG-related issues (not including the DPO-related issues);
- Item Nos. 2.1 through 2.8: Non-SG related issues; and
- Item Nos. 3.1 through 3.11: DPO-related issues.

Historical Background: 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. SECY-00-0078 outlines the staff's proposed review process associated with the revised steam generator tube integrity regulatory framework described in NEI 97-06.

Originating Document: Memorandum from B. Sheron/J. Johnson to S. Collins dated November 16, 2000, "Steam Generator Action Plan" (Accession No. ML003770259).

Regulatory Assessment: The current regulatory framework provides reasonable assurance that operating PWRs are safe. Improvements to the regulatory framework are being pursued through the NEI 97-06 initiative.

Current Status:

- November 1, 2000	Issuance of "Indian Point 2 Steam Generator Tube Failure Lessons-Learned Report" via memorandum from W. Travers to the Commission (Accession No. ML003765272).
- November 3, 2000	Issuance of "Staff Review of OIG Report on the NRC's Response to the Steam Generator Tube Failure at Indian Point 2 and Related Issues" via memorandum from W. Travers to the Commission (Accession No. ML003753067).
- November 16, 2000	Issuance of "Steam Generator Action Plan" via memorandum from B. Sheron/J. Johnson to S. Collins (Accession No. ML003770259).
- February 1, 2001	ACRS Ad Hoc Subcommittee report related to SG DPO issued (NUREG-1740).
- May 11, 2001	Issuance of a memorandum providing a revision to the SG Action Plan to address the issues related to the DPO on SG tube integrity issues (Accession No. ML011300073).
- August 2, 2001	Issuance of a letter to NEI transmitting a draft NRC paper on NEI 97-06 SG generic change package (Accession No. ML012200349).
- September 26, 2001	Staff briefing of ACRS subcommittee on Materials and Metallurgy regarding SG action plan status.
- September 26, 2001	Staff briefing of ACRS Subcommittee on Materials and Metallurgy on SG action plan.
- October 4, 2001	Staff briefing of ACRS full-committee on SG action plan status.
- October 18, 2001	ACRS letter to the Chairman documenting their comment on staff action plan to address the SG DPO (ML012960166).
- November 28, 2001	Public meeting between NRC and NEI management to discuss NEI 97-06 and TMI tube severance issues.
- November 29, 2001	Staff briefing of ACRS Subcommittee on Materials and Metallurgy on NEI 97-06.
- December 3, 2001	Staff briefing of the Commission on the status of SG action plan.
- December 06, 2001	Staff briefing of ACRS on NEI 97-06.
- June 13, 2002	Public meeting between NRC and NEI on SG issues.
- September 11, 2002	Public meeting between NRC and NEI on SG issues.

NRR Technical Contacts:	Louise Lund, DE/EMCB, 415-3248 Doug Coe, DIPM/IIPB, 415-2040 Steve Long, DSSA/SPSB, 415-1077
NRR Lead PM:	Maitri Banerjee, DLPM, 415-2277
RES Contact:	Joe Muscara, RES, 415-5844

OKONITE CABLE LOCA TEST FAILURES
(FINAL UPDATE)

TAC Nos. MA8193, MA9199, MA9200, & MA9201

Last Update: 10/08/02
Lead Division: DE

MILESTONES	DATE (T/C)
1. Meet with Okonite to discuss LOCA test #5 cable failure results	02/08/00 (C)
2. Meet with nuclear industry to discuss LOCA test #5 cable failure results	02/16/00 (C)
3. Issue letter to Okonite with BNL test report	05/17/00 (C)
4. Issue letter to NEI with BNL test report	05/18/00 (C)
5. Meet with NEI and Okonite to discuss impact on operating reactors and responses being considered by NRC and industry	06/22/00 (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.	N/A
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.	08/09/02 (C)
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.	N/A

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.”

Historical 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 were presented to the staff in a meeting on October 12, 2000. NRC is waiting for a response from NEI on the February 7, 2001, letter to NEI. By letter dated July 26, 2001, Okonite provided the staff with the test protocol for EQ testing of Okonite Okalon cables. The EQ test at

Wyle Laboratories, including the test results, were provided to the staff from Okonite by letter dated December 20, 2001. NRC Regulatory Issue Summary (RIS) 2002-11 (ADAMS Accession number ML022190099) was issued on August 9, 2002. Issuance of RIS 2002-11 closes out this issue.

NRR Technical Contact: P. Shemanski, DE/EEIB, 415-1377

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

References:

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.
10. Minutes of NRC public meeting on October 12, 2000.
11. NRC Regulatory Issue Summary 2000-25, December 26, 2000.
12. Letter from Jack Strosnider to NEI, February 7, 2001.
13. Letter from Okonite to Samuel Collins, May 2, 2001.
14. Letter from NEI to Jack Strosnider, July 17, 2001.
15. Letter from Okonite to Samuel Collins, July 26, 2001.
16. Letter from Jack Strosnider to Okonite, August 23, 2001.
17. Letter from Okonite to Samuel Collins, December 20, 2001.
18. Letter from Okonite to Satish Aggarwal, January 25, 2002.
19. Letter from Okonite to Samuel Collins, April 12, 2002.
20. Letter from Samuel Collins to Okonite, April 26, 2002.
21. Letter from Okonite to Jose Calvo, May 29, 2002.
22. RIS 2002-11, August 9, 2002.

EMERGENCY ACTION LEVEL GUIDANCE DEVELOPMENT

TAC No.: MA3695 Revision to NESP-007 Last Update: 09/24/02
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	01/28/99 (C)
2.	NEI to provide new shutdown EAL guidance (NEI-99-01) for NRC review	04/07/99 (C)
3.	NRC provides comments to NEI on NEI-99-01	05/11/99 (C)
4.	Meet with NEI to discuss comments	05/13/99 (C)
5.	Comments resolved and final draft of NEI-99-01 submitted for endorsement	07/99 (C)
6.	Draft guide developed endorsing NEI-99-01 developed in form of a draft guide for CRGR/ACRS review.	03/06/00 (C)
7.	Determination made on whether to issue a Generic Letter on plant-specific implementation of shutdown EALs - no GL to be issued	08/30/00 (C)
8.	CRGR/ACRS meeting on generic letter - canceled	08/30/00 (C)
9.	Draft Guide issued for public comment	03/22/00 (C)
10.	Public comments addressed (NEI-99-01 revised as needed)	07/14/00 (C)
11.	CRGR/ACRS meeting on final guide NEI 99-01 (meeting waived)	11/01/00 (C)
12.	Regulatory Guide issued	TBD

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.

CRGR waived formal review of NEI 99-01 and the final Reg Guide. After discussion with NEI, issuance of the Reg Guide was placed on hold pending final evaluation of the impact of the spent fuel pool study on EALs for decommissioned reactors.

<u>Proposed Actions</u>: 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.

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

<u>Regulatory Assessment</u>: 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.

<u>Current Status</u>: In September 2002, NEI submitted a request that NRC endorse NEI-99-01 regardless of issues with EALs for Defueled Stations and Independent Spent Fuel Storage Installations. The review is on-going.

<u>References</u>:

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.
19. Letter from L. Hendricks to T. Quay, September 23, 2002.

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ECCS SUCTION BLOCKAGE

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

Last Update: 09/30/02
Lead NRR Division: DSSA
Supporting Divisions: DE, DRCH, 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"	10/01 (C)
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	03/00 (C)
○ Complete revision of Draft RG 1.1/RG 1.82 (DG-1107)	09/03 (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"	07/00 (C)
2. NRC-sponsored research program on the potential for coatings to fail during an accident	03/01 (C)
PART IV: **GSI 191, "ASSESSMENT OF DEBRIS ACCUMULATION ON PRESSURIZED WATER REACTOR (PWR) 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)	03/99 (C)
○ Complete collection of plant data to support research program	06/99 (C)
○ Integrate industry activities into this Action Plan	04/00 (C)
○ Complete research program on PWR sump blockage (including final risk assessment)	09/01 (C)
○ Evaluate need for regulatory action based on research program results (NRR)	03/02 (C)

MILESTONES	DATE (T/C)
2. Resolve ECCS suction clogging issue for PWRs (Regulation/Guidance Development and Issuance Stages of GSI process in MD 6.4 (Stages 4 and 5))	
○ Update ECCS Suction Clogging Action Plan to include resolution of the issue for PWRs	01/02 (C)
○ Brief NRR ET to obtain approval to prepare a generic letter (GL)	02/02 (C)
○ Public meeting with NEI, WOG, B&WOG, CEOG	03/02 (C)
○ Proposed Draft GL to CRGR for review	10/02 (T)
○ CRGR Briefing on proposed draft GL	11/02 (T)
○ Proposed draft GL issued for Public Comment	12/02 (T)
○ Public meeting with NEI, WOG, B&WOG, CEOG during Public Comment period	01/03 (T)
○ Public Comment period ends	02/03 (T)
○ Resolution of Public Comments and revisions to proposed GL made, as necessary	03/03 (T)
○ CRGR Briefing on proposed final GL	03/03 (T)
○ ACRS Briefing on proposed final GL	03/03 (T)
○ Information Paper sent to Commission, issue GL	04/03 (T)

Description: This action plan was originally 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 is risk informed.

This plan has four parts. First, for boiling-water reactors (BWRs), this issue has been addressed by licensee responses to NRCB 96-03. At the time this action plan was developed, the staff was in the process of confirming the adequacy of the licensee solutions implemented in response to the bulletin; therefore, the staff's confirmatory effort included in this action plan for completeness. The staff's activities related to NRCB 96-03 are complete. 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 staff's activities related to GL 97-04 are complete. 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 staff's activities related to GL 98-04 and the coatings research program are complete. The results of these two programs 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. RES has recently completed its assessment of the potential for debris clogging of PWR ECCS sumps during a LOCA. The study was performed to support the resolution of generic safety issue (GSI) -191, "Assessment of Debris Accumulation on PWR Sump Performance." RES performed a parametric evaluation to demonstrate whether sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill suited for making a determination that sump blockage will impede or prevent long-term recirculation at a specific plant. By memorandum dated September 28, 2001, RES transferred the lead for GSI-191 to NRR consistent with Management Directive 6.4. The parametric evaluation forms the basis for concluding the Technical Assessment phase of the GSI. RES also recommended in the memorandum that plant-specific analyses be performed by licensees to determine if debris will impede ECCS operation during recirculation, and that appropriate corrective action be taken, if the analyses demonstrate that ECCS operation will be impeded. This plan has been updated to include NRR activities to resolve GSI-191.

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 BWRs. On July 28, 1992, an event occurred at Barsebäck Unit 2, a Swedish 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 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.

RES conducted 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 included 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 also conducted 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 evaluated 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 was 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 were 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 was initially divided into four parallel efforts. Three of these efforts are complete. The action plan has been updated to provide additional NRR actions necessary to respond to RES findings related to GSI-191. The first effort was 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. The issue has been closed based on the audit findings and the findings of the staff's review of coatings related issues (discussed below). A summary of the review results is provided in a memorandum from R. Elliott to G. Holahan, "Completion of Staff Reviews of NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-water Reactors," and NRC Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode'" dated October 18, 2001.

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 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 involved the evaluation of coatings as a potential debris source. Concerns raised in this area were due to events where qualified coatings have failed during normal operation at a number of sites. The failure of qualified coatings during normal operation 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 conducted 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. This research program is complete and the findings are discussed below under "Current Status." The second concern relates to the adequacy of licensee programs to apply and maintain coatings consistent with their licensing bases. This concern was 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. The staff review of the responses to GL 98-04 is complete and identified no significant issues. 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 conducted a program to evaluate PWR sump designs and their susceptibility to blockage by debris. This evaluation included a 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. RES's PWR sump study is complete.

RES conducted a parametric evaluation was performed to demonstrate whether sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill suited for making a determination that sump blockage will impede or prevent long-term recirculation at a specific plant. By memorandum dated September 28, 2001, RES transferred the lead for GSI-191 to NRR consistent with Management Directive 6.4. The parametric evaluation forms the basis for concluding the Technical Assessment phase of the GSI. RES also recommended in the memorandum that plant-specific analyses be performed by licensees to determine if debris will impede ECCS operation during recirculation, and that appropriate corrective action be taken, if the analyses demonstrate that ECCS operation will be impeded. As noted above, this action plan has been updated to include NRR actions necessary to address RES's findings.

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.

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 this action plan has been updated to address PWR sump blockage concerns. As noted above, the results of RES's parametric evaluation demonstrated that sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill suited for making a determination that sump blockage will impede or prevent long-term recirculation at a specific plant. Therefore, it is not clear how significant a threat to PWR ECCS operation exists. The staff considers continued operation of PWRs during the implementation of this action plan to be acceptable 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 generate smaller quantities of debris, 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 flaw will still need to be considered). 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. And finally, the staff believes that continued operation of PWRs is also acceptable because of PWR design features which may minimize potential blockage of the ECCS sumps during a LOCA. The RES study on sump blockage attempted to capture many of the PWR design features parametrically, however, it is not possible for a generic study of this nature to capture all the variations in plant-specific features that could affect the potential for ECCS sump blockage (e.g., piping layouts, insulation location within containment, etc.). Therefore, evaluation on a plant-specific basis is necessary to determine the potential for ECCS sump clogging in each plant.

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

As part of the GSI-191 study, RES's contractor, Los Alamos National Laboratory (LANL), performed a generic risk assessment to determine how much core damage frequency (CDF) is changed by the findings of the parametric analysis. Utilizing initiating event frequencies that consider LBB credit consistent with NUREG/CR-5750, LANL an calculated an overall CDF of 3.3E-06 when debris clogging as a failure mechanism is not considered, and an overall CDF of 1.5E-04 when debris clogging is considered. However, these CDFs were calculated without giving any credit for operator action, and without consideration to whether the ECCS or containment spray pumps would be able to continue operating after the headloss across the sump screen exceeds the calculated licensing basis NPSH margin. The change in CDF is also dominated by the small and very small break LOCAs which are events where there are significant operator actions that can be taken to prevent core damage. Accordingly, it's expected that the actual core damage frequency when accounting for potential operator actions would likely be an order of magnitude lower (e.g., 10E-5). On this basis, the schedule for issuing a generic communication to address the PWR sump clogging issue outlined above is considered to be appropriate.

These conclusions clearly support this action plan as outlined herein.

<u>Current Status</u>: The review of NRCB 96-03 responses is complete.

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 has completed its coating research program and has incorporated the results of this program into the PWR sump study. Available evidence from limited industry tests of the transport of coating debris indicates that coating debris (chips) may not transport very well under conditions approximating those of containment sump flow. In fact, only very small amounts of debris actually reached the screens in these tests.

RES did identify a potential new mechanism for generation of coating (particulate) debris. Specifically, some qualified coatings irradiated to 10^9 Rads and placed in 200° Fahrenheit water did generate debris. However, this coating debris appears to have been caused by irradiating the coatings to the bounding levels specified in the ASTM standards for coating qualification. When the coatings were irradiated to a more realistic level consistent with conditions expected in operating reactors (i.e., calculated levels consistent with a 60 year plant life followed by a LOCA or approximately 10^7 Rads), coating debris was not generated. As a result, the staff concluded that no regulatory action based on the results of the coatings program is required at this point.

RES's PWR sump study is complete. To date, the industry has monitored the NRC's activities in this area rather than conduct any testing or research of their own. As part of the generic safety issue (GSI) -191, "Assessment of Debris Accumulation on PWR Sump Performance," a parametric evaluation was performed to demonstrate whether sump blockage is a plausible concern for operating pressurized water reactors (PWRs). The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill suited for making a determination that sump blockage will impede or prevent long-term recirculation at a specific plant. By memorandum dated September 28, 2001, RES transferred the lead for GSI-191 to NRR consistent with Management Directive 6.4. The parametric evaluation forms the basis for concluding the Technical Assessment phase of the GSI. RES also recommended in the memorandum that plant-specific analyses be performed to determine if debris will impede ECCS operation during recirculation, and that appropriate corrective action be taken, if the analyses demonstrate that ECCS operation will be impeded. This action plan has been updated to address the concerns identified in the RES GSI-191 study.

On July 3, 2001, RES has made available to the public the draft Los Alamos National Laboratory report entitled, "GSI-191: Parametric Evaluation for Pressurized Water Reactor Recirculation Sump Performance," dated July 2001. This report documents the parametric evaluation. The draft report was made publicly available to facilitate discussions with external stakeholders. RES presented the results of the GSI-191 parametric evaluation to the ACRS on July 12 and September 5, 2001. Also, a public meeting between the NRC, the Nuclear Energy Institute, and the three Pressurized Water Reactor Owners' Groups was held on July 26 and 27, 2001, to discuss the parametric evaluation with interested stakeholders. The staff continues to hold regular public meetings with the three PWR owners groups and NEI on the progress toward resolving GSI-191.

<u>NRR Lead PMs</u>:	Donna Skay, LPD I-1, 415-1322 (NRCB 96-03, GL 97-04) John Lamb, LPD III-1, 415-1446 (PWR Sumps) Bob Pulsifer, PD I-2, 415-3016 (Containment Coatings, GL 98-04, GE Topical Report)
<u>NRR Lead Technical Reviewers</u>:	Ralph Architzel, SPLB, 415-2804 John Lehning, SPLB, 415-3285
<u>NRR Technical Contacts</u>:	Rich Lobel, SPLB, 415-2865 Nicholas Saltos, SPSB, 415-1072
<u>RES Technical Contact</u>:	B. P. Jain, ERAB, 415-6778

<u>References</u>:

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.

Letter from Gary M. Holahan to James F. Klapproth, “NRC Staff Review of GE Licensing Topical Report NEDC-32721P, ‘Application Methodology for the General Electric Stacked Disk ECCS Suction Strainers,’ TAC Number M98500,” dated June 21, 2001.

Los Alamos Technical Report, LA-UR-01-4083, "GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2001.

Memorandum from Ashok C. Thadani to Samuel J. Collins, “RES Proposed Recommendation for Resolution of GSI-191, ‘Assessment of Debris Accumulation on PWR Sump Performance,’” dated September 28, 2001 (Accession Number ML012750149).

Memorandum from Robert B. Elliott to Gary M. Holahan, “Completion of Staff Reviews of NRC Bulletin 96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-water Reactors,” and NRC Bulletin 95-02, “Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode’” dated October 18, 2001 (Accession Number ML012970261).

CONTROL ROOM HABITABILITY

TAC Nos.: MB0449, MB0450
GSI No.: N/A
CTL: N/A

Last Update: 10/04/02
Lead NRR Division: DSSA
Supporting Division: TBD

	MILESTONES	DATE (T/C)
1.	Staff review of NEI 99-03 and redline and strikeout version provided to NEI Control Room Habitability task force	04/17/01 (C)
2.	Staff prepare Generic Letter and develop draft Regulatory Guides on Control Room Habitability at Nuclear Power Reactors (DG-1114), Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (DG-1115), Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light Water Nuclear Power Reactors (DG-1113), and Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (DG-1111)	07/01/01 (C)
3.	Office review of draft Regulatory Guides DG-1111 and DG-1113	12/31/01 (C)
4.	Office review of draft Regulatory Guides DG-1114 and DG-1115 and draft Generic Letter	03/01/02 (C)
5.	Brief CRGR on draft Regulatory Guides DG-1111 and DG-1113	12/31/01 (C)
6.	Brief CRGR on draft Regulatory Guides DG-1114 and DG-1115 and draft Generic Letter	draft GL: 04/29/02 (C) DG-1114, DG-1115: 03/11/02 (C)
7.	Issue draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115 and draft Generic Letter for public comment	draft GL: 05/09/02 (C) DG-1111: 12/31/01 (C) DG-1113: 01/31/02 (C) DG-1114: 03/28/02 (C) DG-1115: 03/28/02 (C)
8.	Public meeting on draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115 and draft Generic Letter	RI: 07/11/02 (C) RII: 07/16/02 (C) RIII: 08/06/02 (C) RIV: 07/18/02 (C)
9.	Resolve public comments on draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115	11/29/02 (T)
10.	Office review and concurrence of final Regulatory Guides and Generic Letter	12/31/02 (T)
11.	Brief ACRS on final Regulatory Guides and Generic Letter	02/03 (T)
12.	Brief CRGR on final Regulatory Guides and Generic Letter	02/03 (T)
13.	Commission Information Paper on Generic Letter	03/03 (T)
14.	Issue final Regulatory Guides and Generic Letter	04/03 (T)

Description: General Design Criterion (GDC-19), “Control Room,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, establishes criteria for a control room. It requires that a control room be provided which allows operators to take actions under normal conditions to operate the reactor safely and to maintain the reactor in a safe condition under accident conditions. GDC-19 also requires that equipment be provided at locations outside the control room with the design capability for hot shutdown of the reactor, including the necessary instrumentation and controls that both maintain the reactor in a safe condition during hot shutdown and possess the capability for the cold shutdown of the reactor through the use of suitable procedures. GDC-19 also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures more than 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Applicants to build or license a new plant under Part 50 after January 10, 1997, applicants for design certification under Part 52 after January 10, 1997, applicants to build a new plant under Part 52 who don’t reference a standard design certification, or current licensees who want to use an alternative source term as allowed by 50.67, are required by GDC-19 to use as the control room dose criterion 0.05 Sv (5 rem) total effective dose equivalent (TEDE).

In its review of license amendment submittals over the past several years, the staff has identified numerous problems associated with the assessment of control room habitability. These problems have included the overall integrity of the control room envelope and the manner in which licensees have demonstrated the ability of their control room designs to meet GDC-19. Licensees have failed to: (1) assess the impact of proposed changes to plant design, operation, and performance on control room habitability, (2) identify the limiting accident, (3) appropriately credit the performance of control room isolation and emergency ventilation systems in a manner consistent with system design and operation, and (4) substantiate assumptions regarding control room unfiltered inleakage. In response to this latter concern, several utilities performed testing of their control room unfiltered inleakage using methods from ASTM E741-93, “Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution.” The tests performed represent about 30 percent of the operating plants’ control rooms. In all of the tests performed to date except one, the measured unfiltered inleakage exceeded the design basis analysis assumptions; in several cases by over an order of magnitude. Also, in all of the cases to date, the licensees have been able to ultimately demonstrate compliance to GDC-19 through corrective action and retesting or by re-analysis. The nearly 100 percent failure rate of such a large fraction of the operating plant control rooms creates a large uncertainty in the ability of the remaining untested facilities to meet control room habitability requirements.

These control room habitability issues adversely affect the timely review of many current license amendment requests. Licensee and staff expend extensive resources to resolve differences of opinion regarding licensing and design basis issues and to resolve weaknesses in analysis assumptions, inputs and methods.

While the capability of untested control rooms to meet their design basis is in question, the staff has reasonable assurance that continued operation is safe for the following reasons: Events that would impact control room habitability are of fairly low probability. Compensatory measures; e.g., use of self contained breathing apparatus and potassium iodide, although not ideal, are available. The staff has been working with industry to address the issues. There are analytical conservatisms.

Historical Background: In March 1998, the staff briefed the Office of Nuclear Reactor Regulation Executive Team (ET) on its concerns related to the infiltration testing results and other aspects of control room habitability. The ET directed the staff to work with the Nuclear Energy Institute (NEI) to resolve the issues. Pursuant to this direction, the staff co-hosted, with NEI and the Nuclear Heating Ventilation and

Air Conditioning Users Group (NHUG), a workshop on control room habitability in July 1998. Following this workshop, NEI agreed to form a task force to address control room habitability. In August 1999, NEI submitted for staff review and comment a draft of a proposed NEI document intended to address this issue. This document, NEI 99-03, entitled, “Control Room Habitability Assessment Guidance,” did not adequately address the staff’s concerns. In response to the staff concerns, NEI agreed in December 1999 to restructure NEI 99-03. During the period January 2000 through June 2000, the NEI task force met with the NRC staff in public meetings on nearly a monthly basis to resolve outstanding issues and to discuss the appropriate content of NEI 99-03. The latest NEI 99-03 revision was sent to the staff on October 13, 2000. The staff reviewed the October 13, 2000, revision and determined that, while there was much agreement on positions taken in the document, areas remained where the staff and industry were in disagreement. The staff has now determined and NEI agrees that the staff should reflect its position in formal regulatory guidance, and the issues should be resolved through the public comment process. NEI issued in June 2001 the final version of NEI 99-03, “Control Room Habitability Assessment Guidance,” which is substantially the same as the October 13, 2000, draft reviewed by the NRC staff.

Proposed Actions: This action plan provides for staff activities toward a generic resolution to the issues of control room habitability. The NRC staff has been pursuing a technically correct, optimum solution to the control room habitability issue with the NEI issue task force. The staff has indicated its willingness to step forward and to incorporate up-to-date information into its assessment of radiological analyses. The staff is considering possible changes in the radiological dose acceptance criteria and possible reductions in the conservatisms in control room habitability analyses. Such steps could result in the reduction of unnecessary regulatory burden. Presently, NEI has not committed to making this industry initiative binding on individual utilities. The staff believes that a voluntary approach may not adequately resolve the staff concerns and that some generic approach may still be needed. A Generic Letter will request licensees to take action to evaluate, in light of the ASTM E741 testing results to date, how they meet the requirements of GDC-19 with respect to unfiltered inleakage to their control room envelopes.

During staff interaction with the NEI issue task force, many issues were discussed and it is necessary that proper attention be applied to these issues. The staff feels that additional regulatory guidance is necessary in order that these control room habitability issues are addressed in a complete and thorough manner. In addition, it is necessary that the regulatory information associated in this area be updated to reflect current knowledge. In meetings with the NEI Task Force on Control Room Habitability, changes to design basis accident radiological analysis assumptions were discussed. The staff and industry believe it is necessary to update the analysis guidance contained in numerous current regulatory guides and consolidate it into one regulatory guide on design basis accident radiological analyses using the plant’s original design and licensing source term, which in most cases is taken from TID-14844. For those licensees that implement an alternative source term as allowed by 10 CFR 50.67, Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” currently provides guidance for performing control room radiological analyses. The staff also believes that creating regulatory guidance on meteorology for control room habitability assessment is necessary and appropriate. These regulatory guides would be vehicles to present to the industry and public more realistic assumptions based on current knowledge that are acceptable to the staff. In addition, it has been almost 20 years since the staff updated its information on control room habitability. Various staff and industry studies have been conducted in those 20 years. These studies have uncovered issues which were addressed to only a limited extent in the previous guidance on control room habitability. A regulatory guide on control room habitability would assist licensees to determine the present state of their control room envelope integrity. Along with the control room habitability regulatory guide, an additional regulatory guide on control room envelope integrity testing would provide guidance to the industry on how plants may determine control room envelope integrity and continually demonstrate that integrity. Such regulatory guidance would utilize the information gleaned from testing 30 percent of the control room envelopes.

The initial deliverables for this action plan are the Generic Letter mentioned above and new Regulatory Guides on: (1) control room habitability, (2) control room envelope integrity testing, (3) meteorology for control room habitability assessments, and (4) design basis accident radiological analyses. The latter would revise and consolidate the suite of Regulatory Guides for design basis accident radiological analyses.

Additionally, to support licensees that begin testing the integrity of the control room envelope by measuring unfiltered inleakage, the staff is proposing to the Technical Specifications Task Force changes to standard technical specifications on control room emergency ventilation systems. The staff's position that changes may have to be made to technical specifications had been discussed previously during interaction with the NEI control room habitability task force.

Resolution of this issue is supportive of the NRR pillars of maintaining safety, increasing public confidence (both by restoring control room integrity to the level assumed in the facility's licensing basis), increasing effectiveness and efficiency of key NRC processes (via a generic approach to resolution rather than the current plant-by-plant approach), and may reduce unnecessary regulatory burden and increase realism (due to possible relaxation in certain analysis assumptions and acceptance criteria, based on current information).

Originating Document: None.

Regulatory Assessment: The staff believes that the potential deficiencies in the control room habitability designs, operations, and analyses represent safety issues that warrant resolution. It is important to recognize that the objective of control room habitability requirements, such as those in GDC-19, is not to minimize operator exposure for the purposes of ALARA (which is controlled under 10 CFR Part 20), but to provide a habitable environment in which to take action to operate the reactor safely under normal conditions and to maintain it in a safe condition under accident conditions, thereby to provide protection to the public. The numeric criterion of 5 rem whole body was selected as it was believed that operations personnel would not be distracted from necessary plant operations and would not unnecessarily evacuate the controls area due to concerns for their personal safety, thereby potentially affecting the protection of the public health and safety.

Protection against smoke and other toxic gases is also necessary since these hazards could cause, in some cases, immediate physical impairment or incapacitation of control room operators. While toxic gases are considered in control room habitability analyses in accordance with the guidance in Regulatory Guide 1.78, the potentially toxic byproducts of fires and their impacts on control room habitability were not considered a problem in the past because of the presumed control room envelope integrity. In the past, a fire outside the control room was considered to have no impact upon the operators because smoke and toxic fire gases were never presumed to enter the control room envelope. If a fire occurred in the control room, the operators had the remote shutdown areas for controlling the reactor. Testing of the control room envelope's integrity has demonstrated that the perceived integrity does not exist. Consequently, some portions of the smoke issue may be covered under this action plan while other aspects may not.

The staff considered the risk impacts of control room habitability and made a preliminary determination that control room habitability has not been addressed in current PRAs because: (1) it has been assumed that the design basis was being met, and (2) quantification of the risk associated with failure to meet the design basis for control room habitability is not addressed by current metrics, methods, and risk experience data.

Current Status: Four draft regulatory guides, numbered DG-1111, DG-1113, DG-1114 and DG-1115, have been issued for public comment. Proposed Generic Letter 2002-XX, “Control Room Envelope Habitability,” (ADAMS accession number ML021430317) was published on May 9, 2002, at 67 FR 31385. The joint public comment period for the four draft guides and the proposed generic letter expired on August 7, 2002. The staff held four public meetings, one in each region, on July 11, 16, and 18 and August 6, 2002, to solicit NRC stakeholder feedback on the proposed Generic Letter and four draft regulatory guides. NEI requested a public meeting with the staff to discuss the comments they had sent, and this public meeting took place September 10, 2002. The staff is currently dispositioning the comments received during the public comment period and making the necessary revisions to the draft guides and proposed generic letter.

The staff’s proposed changes to technical specifications for control room emergency ventilation systems have been presented to the Technical Specifications Task Force, and industry has said that they will prepare a package to address the staff’s proposal. No action by the Technical Specifications Task Force has occurred on this since the last update.

NRR Contacts: J. J. Hayes, SPSB/DSSA/NRR, 415-3167
M. Hart, SPSB/DSSA/NRR, 415-1265

References:

USNRC, Title 10 Code of Federal Regulations Part 50, Appendix A.

USNRC, “Clarification of TMI Action Plan Requirements,” NUREG-0737, 1980.

USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800.

L. Soffer, et al, “Accident Source terms for Light Water Nuclear Power Plants,” NUREG-1465, 1995.

Murphy, K.G. and Campe, K. W., “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19,” published in proceedings of 13th AEC Air Cleaning Conference.

Driscoll, J. W., “Control Room Habitability Survey of Licensed Commercial Nuclear Power Generating Stations,” NUREG/CR-4960, 1988.

DiNunno, et al, “Calculation of Distance Factors for Power and Test Reactor Sites,” Technical Information Document TID-14844, USAEC, 1962.

USNRC, Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” 2000.

American Society for Testing and Materials ASTM E741, “Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution,” 1993.

ATTACHMENT 2

GENERIC COMMUNICATION AND COMPLIANCE ACTIVITIES

Open Generic Communication TACs (PA No. 101122CA/B)

Summary Report (07/15/2002 - 10/04/2002)

TAC NO.	TAC TITLE	AGE	LEAD ORG
MB0703	RIS on Improvements in Distribution of Generic Communications	22	DRIP
MB1537	IN-FFD PERFORMANCE DATA IN	19	DIPM
MB2530	RIS: Part 9900 Revision	15	DRIP
MB2788	GL - Revision to NEI 99-03, 5 rem TEDE - Hayes	14	DSSA
MB4070	IN: 4160 Volt Switchgear Problem at Columbia Generating Station	8	DRIP
MB4864	GL: Potential Clogging of Containment Recirculation Sump Screens by Debris Accumulation at PWRs	6	DSSA
MB5109	IN: Recent Design Problems in Safety Functions of Pneumatic Systems (Hodge)	5	DRIP
MB5184	RIS: NRC Threat Advisory Conditions (Shapaker)	5	NSIR
MB5261	IN: Electromigration Issue (Shapaker)	4	DE
MB5372	Prepare a RIS on Inspection of Steam Generator Tubes in the Tube Sheet Region	4	DE
MB5490	BL 2002-01, Sup 1 - Reactor Pressure Vessel Head Degradation & Reactor Coolant Pressure Boundary Integrity (ALee/JWS)	4	DE
MB5626	RIS: Tech Assessment of GSI 168, Environmental Qual of Low-Voltage Instrumentation & Control Cables (Shemanski/JWS)	3	DE
MB5683	RIS: Issuance of NRC MD 8.17, "Licensee Complaints Against NRC Employees"	3	DIPM
MB5738	RIS: The Use of EPRI TR-102348, Revision 1, (NEI 01-01), Guideline on Licensing Digital Upgrades (Mortensen/Shapaker)	2	DE
MB5762	RIS: National Guard and Other Emergency Responders Located in the Licensees' Owner Controlled Area (Blount/Shapaker)	2	DIPM
MB5869	GL: Inventories of Government-Owned Nuclear Materials at Licensee Sites (NSIR/Shapaker)	2	NSIR
MB6076	RIS: Orders Regarding Transportation of Spent Nuclear Fuel (PBrochman (NSIR)/Shapaker)	2	NSIR
MB6186	IN: IN 89-69, Supplement 1 - Loss of Thermal Margin Caused by Channel Box Bow (Jimenez (SRXB/DSSA)/Dozier)	2	DSSA
MB6278	RIS: Manual Actions to Satisfy Appendix R (Qualls (DSSA)/Shapaker)	1	DSSA
MB6294	RIS: High Security Protected and Vital Area Barrier/Equipment Penetration Manual (Vanden Berghe	1	NSIR
MB6308	RIS: RIS 2002-14, Sup 1: Proposed Changes to the Safety Sys Unavailability Perform Indicators (Sanders(DIPM)/Shapaker)	1	DIPM

Closed Generic Communication TACs (PA No. 101122CA/B)

Summary Report (07/15/2002 - 10/04/2002)

TAC NO.	TAC TITLE	AGE	TAC CLOSED	LEAD ORG
MB3216	RIS-Changes to safety system unavailability - Sanders	12	09/13/2002	DIPM
MB3554	IN: POTENTIAL PROBLEMS WITH THE USE OF HEAT COLLECTORS ON FIRE PROTECTION SPRINKLER HEADS	7	07/29/2002	DSSA
MB3555	IN: Recent Fires at Nuclear Power Plants	10	10/04/2002	DSSA
MB4522	RIS-NRC's Incident Response Program Issues (Fields)	7	09/25/2002	NSIR
MB4779	IN: EP Public Notification Issues - ANS, EAS, Telezapper (Kahler/Petrone)	5	09/13/2002	DIPM
MB4964	IN: IN 2002-02 Sup 1 - Recent Experience with Plugged Steam Generator Tubes (RCaldwell)	2	07/29/2002	DE
MB5017	RIS: Electronic Transmission of Sensitive Unclassified Safeguards Information (Shapaker/NSIR)	4	09/13/2002	NSIR
MB5117	RIS: Revision of the Skin Dose Limit in 10 CFR Part 20 (Shapaker)	2	07/24/2002	NMSS
MB5466	RIS: Requalification Program Test Results of Okonite Okolon Single Conductor Bonded Jacket Cable (Shemanski/Shapaker)	2	08/22/2002	DE
MB5468	BL: Vessel Head and Vessel Head Penetration Inspection (Marshall/Shapaker)	2	08/22/2002	DE
MB5618	IN: Failure of Steam Dryer Cover Plate Related to Recent Power Uprate (Caldwell/Lanyi)	2	09/25/2002	DRIP
MB6133	RIS: Licensee Guidance Regarding Verification of Military Service (NSIR/Shapaker)	1	10/04/2002	NSIR
MB6134	RIS: Guidance Regarding Confirmation of Employment Eligibility (NSIR/Shapaker)	1	09/13/2002	NSIR

ATTACHMENT 3

RISK-INFORMED INITIATIVES

RISK-INFORMED INITIATIVES

A. CURRENT INITIATIVES

INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
1. Revised Oversight Process			
- Enhanced performance indicators (PIs)	-Piloted replacement scram and loss of normal heat removal PIs (3/02) - Joint NRC/industry working group met periodically to develop consistent approach for safety system unavailability reporting - Briefed Commission and ACRS - Conducted public workshop for MSPI Pilot Program (7/02)	- Developing mitigating systems performance index (MSPI) for unavailability and unreliability of plant systems - Testing MSPI concept (6-month pilot began Sept. 1)	- Brief ACRS on MSPI pilot (7/03) - Assess feasibility of enhanced (risk-based) PIs for containment using LERF models - Improve current set of PIs
- Industry-level performance Indicators in the Industry Trends Program (ITP)	- Issued SECY-02-0058, "Results of the ITP and Status of Ongoing Development" (4/02) - Briefed Com. and ACRS (5/02) - Issued 72 plant specific SDP notebooks, Rev. 0	- Developing Initiating Events index (IEI) based on relative contribution to risk - Developing risk-informed thresholds for ex-AEOD PIs - Developing risk-informed thresholds for ROP PIs	- Update data and develop thresholds for operating experience information, including system reliability - Brief ACRS on IEPI and threshold development
- Significance determination process (SDP)	- Issued revised ALARA SDP (3/02) - Issued revised concurrent deficiency guidance for reactor safety SDP (3/02)	- Implementing/improving SDP notebooks, Rev. 1 - Evaluate fire protection, shutdown, external events - SDP Task Group conducting comprehensive review of SDP	- Implement all elements of the SDP improvement plan - Develop S/G Tube Integrity and Risk Assessment SDPs

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
2. Risk-informed Licensing Actions	Updated guidance documents - General guidance (RG 1.174 and SRP chapter 19) Developed guidance documents - IST (RG 1.175 and SRP section 3.9.7) - Graded QA (RG 1.176 and GQA inspection guidance) - TS (RG 1.177 and SRP section 16.1) - ISI (RG 1.178 and SRP section 3.9.8) Issued hundreds of risk-informed amendments over last few years	Publish revisions to guidance documents - General guidance (RG 1.174 and SRP chapter 19) Updating guidance - For ISI, staff is reviewing ASME code cases associated with existing guidance and methodology and draft Appendix X to Section 11 of ASME Code - For IST, staff is awaiting final 10 CFR 50.69 rule language to determine appropriate revisions Reviewing increasing number of risk-informed amendments	Publish revisions to guidance documents - ISI (RG 1.178 and SRP section 3.9.8) Evaluate RG 1.177 and SRP section 16.1 to determine if revision is needed Evaluate additional industry proposals (e.g., eliminate PASS requirements, extend ILRT interval)

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
3. Risk-informed technical specifications	- Working with NSSS owners groups and NEI to coordinate submittals - Goal is to reflect safety significance of the condition or requirement - Eight industry initiatives 1. modified end states 2. missed surveillance 3. flexible mode restraints 4. risk-informed AOTs with a backstop 5. optimize surveillance frequencies 6. modify LCO 3.0.3 to about 24 hours 7. define actions to be taken when equipment is not operable but functional 8. risk-inform the scope of the TS rule	- Initiative 2 complete and available using a Consolidated Line Item Improvement Process - Reviewing industry concepts for initiatives 4, 5, and 7. - Safety evaluations written for CE and BWR topical reports on initiative 1 - First CLIIP *Federal Register* notice on initiative 3 was published on August 2, 2002. Public comments are being addressed. - Writing safety evaluation on CE topical report on initiative 6.	- Define “pilot” efforts to support initiative 4 and 5 - Issue FRN announcing availability of initiative 3 for licensee adoption - Continue reviews of other initiatives

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
4. Fire protection	- NFPA-805 national standard was issued in April 2001. (NFPA-805 is an alternative performance-based risk-informed fire protection standard for nuclear power plants.)	- Staff working on proposed rulemaking that would endorse NFPA 805 as a voluntary alternative to NRC existing fire protection regulations. Draft rule language was posted on the NRC Regulatory Forum web site for public comment in December 2001 and draft proposed rule language was placed on the Regulatory Forum web site for public information twice during spring 2002. The staff provided the proposed rule to the Commission in SECY-02-132 on July 15, 2002. NEI is interacting with the staff regarding its effort to separately develop implementation guidance for NFPA-805. NRC plans to endorse the implementation guidance via Regulatory Guide.	- Publish proposed and final rules (10 CFR 50.48) - Publish RG.

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
4. Fire protection (cont.)	- Circuit Analysis Resolution Program (CARP)	- Staff has revised the CARP and is obtaining technical assistance from BNL and SNL to develop a risk-informed approach to resolve the circuit analysis issues.	- Following the issuance of a risk-informed Generic Letter in September 2003, the staff will withdraw the enforcement guidance memorandum that halted inspection activities in this area.
5. Safeguards NOTE: This effort is now the responsibility of the Office of Nuclear Security and Incident Response	- Proposed revisions to 10 CFR 73.55 sent to Commission 6/4/01. Proposal requires that licensees' security programs employ risk insights in identifying target sets of equipment necessary to prevent core damage and/or spent fuel sabotage and create a more performance oriented basis for security regulations. - Proposed 73.55 returned by Commission to staff for rework to reflect lessons learned from September 11, 2001, events.	- Subsumed by staff efforts on post-September 11, 2001, Response to Terrorist Activities.	- Subsumed by staff efforts on post-September 11, 2001, Response to Terrorist Activities.

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
6. 10 CFR 50.69 rulemaking - risk-informing scope of special treatment requirements	- Pilot plants completed IDP review of categorization, with staff observation - Draft rule language made available for public comment on NRC web site. (Notice of Availability published in November 29, 2001, Federal Register); revised drafts posted April 5 and August 2, 2002.	- Sent proposed rule package to Commission in paper dated September 30, 2002	- Post revised draft rule language on NRC web site. - Complete review of industry guidance documents - Publish proposed and final rules (10 CFR 50.69)
7. RIP50/Option 3 (risk-informing technical requirements) - Combustible Gas Control (10 CFR 50.44) - Fracture Toughness Requirements(10 CFR 50.61) - Emergency Core Cooling System (ECCS) requirements (10 CFR 50.46)	- Developed framework document to guide Option 3 efforts - Published proposed rule changes to 10 CFR 50.44 on August 2, 2002. - Technical reports delivered by RES to NRR	- Awaiting completion of public comment period for proposed rule changes to 10 CFR 50.44. - Developing technical basis for risk-informed changes to 10 CFR 50.61 - Staff is reviewing the RES recommendations and is continuing to develop technical basis for rulemaking	- Publish final rule changes to 50.44 - Publish proposed and final rule changes to 50.61 - Publish proposed and final rule changes to 50.46

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
8. PRA standards	- ASME standard completed on Level 1 and Level 2 LERF PRA (full power) - Staff prepared SECY paper informing Commission of intent to write Reg Guide addressing use of PRA standards (including ASME PRA standard) and industry peer review process for regulatory applications - Reviewed industry guidance on peer reviews	- Continuing work with ANS on external events, low power and shutdown, and internal fires - Developing regulatory guidance which addresses industry standards and industry guidance on peer review	- Publish regulatory guidance for public review and comment - Provide ASME with comments for future revision of standard
9. Creating a risk-informed environment	- Report on current environment issued to ET and Deputy Regional Administrators on August 30, 2002 - Presentation on risk-informed environment initiative given at ANS Utility Working Conference - Paper on results of focus groups and interviews with staff presented at ANS Topical Meeting on Probabilistic Safety Assessment	- Preparing for NRR/Regional Division meetings to discuss initiative - Revised action plan prepared; currently under review by management	- Conduct NRR/Regional Division meetings - Implement revised action plan

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
10. Licensing issues associated with non-LWRs	- Letter containing staff's preliminary assessment sent to Exelon March 2002. - In April 2002, Exelon canceled the PBMR project.	- Preparing identification of policy issues Commission Paper (Summer 2002)	- RES/NRR staff will continue to formulate policy issues associated with licensing non-LWRs and engage the Commission as appropriate. - Upcoming Commission paper on Policy Recommendations (Fall 2002).
11. Advanced Reactor Regulatory Framework	- Staff met internally to discuss options for an advanced reactor risk-informed regulatory framework. Focus on how framework for new reactors is integrated with ongoing risk-informed initiatives. - NEI submitted a white paper on May 7, 2002, (Accession #: ML021350406)	- RES staff will review NEI white paper as part of their efforts to develop an advanced reactor regulatory framework - NRR/DRIP staff will ensure that efforts for item 13, Improving Coherence Among Risk Informed Activities, are coordinated and integrated to the extent possible with advanced reactor framework development.	.
12. Construction Inspection Program reactivation	- Use of risk insights in the Construction Inspection Program is being proposed by NEI.	- Ongoing meetings with NEI	

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
13. Improving Coherence Among Risk Informed Activities	- Public meeting held on 9/20/02 - Staff developed outline for coherence plan	- Developing detailed coherence plan - Staff plans and activities discussed at ANS conference (PSA '02) in Detroit, Michigan	- Hold public meetings and workshops to gather stakeholder input (including review of NEI white paper) - Commission paper in fall
14. Risk-Informed Regulation Implementation Plan (RIRIP)	- Last published July 12, 2002 (SECY-02-0131)		- Publish semiannual updates

B. COMPLETED INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
1. Maintenance Rule	- New section (a)(4) effective 11/28/00 - RG 1.182 endorses industry guidance document for managing risk during maintenance activities	- Participating in risk-informed technical specifications initiatives, including licensee use of programs and processes developed to implement 10 CFR 50.65(a)(4) - Developing "Efficacy of 10 CFR 50.65, The Maintenance Rule, memorandum to the Commission from the EDO	
2. Reporting Rules	- Revised 10 CFR 50.72 and 50.73 effective 1/23/01 - Focuses on reporting only events that are risk-significant	- Evaluating reports to determine effectiveness of new rules	
3. Alternate source term	- New rule (10 CFR 50.67) published 12/23/99; RG1.183 issued 7/2000 - Allows for application of improved knowledge of fission product releases and plant performance	- Evaluating license amendments that take advantage of new rule. Several have been approved to date.	- Continue processing applications received from licensees. Consideration is being given to possible revision of RG 1.183 to reflect some lessons learned.