DISTRIBUTION for NRR Director's Quarterly Status Report

Central File	RGEB R/F	WKane, EDO	SJCollins, NRR
JJohnson, NRR	RWBorchardt, NRR	BWSheron, NRR	DBMatthews, NRR
CCarpenter, NRR	SWest, NRR	MMalloy, NRR	EMMcKenna, NRR
CPetrone, NRR	GFSuber, NRR	BJSweeney, NRR	BABoger, NRR
JAZwolinski, NRR	GMHolahan, NRR	JRStrosnider, NRR	MCase, NRR
RCEmrit, RES	Regional Administrators		

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

Nancy G. Chapman, SERCH Manager
Bechtel Power Corporation
5275 Westview Drive
Frederick, MD 21703-8306

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

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

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

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

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

ADAMS ACCESSION NUMBER: ML011980350

DISTRIBUTION for NRR Director's Quarterly Status Report

Central File	RGEB R/F	WKane, EDO	SJCollins, NRR
JJohnson, NRR	RWBorchardt, NRR	BWSheron, NRR	DBMatthews, NRR
CCarpenter, NRR	SWest, NRR	MMalloy, NRR	EMMcKenna, NRR
CPetrone, NRR	GFSuber, NRR	BJSweeney, NRR	BABoger, NRR
JAZwolinski, NRR	GMHolahan, NRR	JRStrosnider, NRR	MCase, NRR
RCEmrit, RES	Regional Administrators		

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

Nancy G. Chapman, SERCH Manager
Bechtel Power Corporation
5275 Westview Drive.
Frederick, MD 21703-8306

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

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

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

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

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

ADAMS ACCESSION NUMBER: ML011980350

ADAMS DOCUMENT TITLE: Public Version of July 2001 Director's Quarterly Status Report

DOCUMENT NAME: DIST.WPD

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

OFFICE	RGEB:DRIP		RGEB:DRIP		REXB:DRIP		~~RGEB:DRIP~~		RGEB:DRIP		RGEB:DRIP		D:DRIP	
NAME	BSweeney:bs		EMcKenna		CPetrone		~~MMalloy~~		SWest		CCarpenter		FGillespie for DMatthews	
DATE	07/27/01		07/27/01		07/30/01		~~07/ /01~~		07/31/01		08/01/01		08/03/01	

OFFICIAL RECORD COPY

DIRECTOR'S STATUS REPORT

on

GENERIC ACTIVITIES

Action Plans

Generic Communication and Compliance Activities

JULY 2001

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

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.: MA0785, MA0792, MA0793, MA1926, MA1927, MA2326, MA2328, MA3673, MA4203, MA4464, MA4465, MA4467, MA4468, MA5012, MA5140, MA6015, MA7323, MA7356, MA9111, MB0271

Last Update: 06/30/01
Lead NRR Division: DE
Supporting Division: DSSA
GSI: Not Available

MILESTONES	DATE (T/C)[1]
PART I: REVIEW OF GENERIC INSPECTION AND EVALUATION CRITERIA	
1. Issue summary NUREG-1544	03/96 (C)
○ Update NUREG-1544	3Q/02 (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)	12/31/01 (T)

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)	10/01/01 (T)
○ LPCI Coupling Repair Design Criteria (BWRVIP-56)	10/01/01 (T)
○ Instrument Penetrations Repair Design Criteria (BWRVIP-57)	10/01/01 (T)
○ CRD Internal Access Weld Repair (BWRVIP-58)	10/01/01 (T)
○ Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals (BWRVIP-59)	07/31/01 (T)
○ BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants (BWRVIP-60)	07/08/99 (CA)
○ Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)	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)	08/31/01 (T)
○ 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)	08/31/01 (T)
○ BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material (BWRVIP-78)	12/31/01 (T)
○ Evaluation of Crack Growth in BWR Shroud Vertical Welds (BWRVIP-80)	12/31/01 (T)

[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. This schedule change is primarily attributed to BWRVIP-74, and -76, since the staff is waiting for the BWRVIP to supplement its original submittal in accordance with the open items in the staff's initials SE's on these reports.

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

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

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

NRR Technical Contacts: C. E. Carpenter, EMCB, 415-2169
Jai Rajan, EMEB, 415-2788

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

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

Action Plan dated April 1995.

STEAM GENERATORS

TAC Nos.	Description	
M88885	Steam Generator (SG) Integrity Rulemaking	Last Update: 07/02/01
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	

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)	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)	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)	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)	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)	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)	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)	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)	DE L. Lund	DIPM S. Malur 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)	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)	DE C. Khan	DIPM S. Malur 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	07/31/01 (T)	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	09/28/01 (T)	DIPM P. Koltay	DSSA S. Long DE C. Khan
	c) review and revise the training program for inspectors (Attachment 3 to Ref. 1; TG: 5a, 5b, 5c, 5d, 5f, 6c)	09/28/01 (T)	DIPM E. Kleeh	DE C. Khan DSSA S. Long
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)	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)	DLPM R. Ennis	
1.14 (MA4265)	Staff completes review and 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	DE E. Murphy	
1.15 (MB0631)	Hold steam generator workshop with stakeholders (page 2 of Ref. 1; page 2 of Ref. 2)	02/27/01 (C)	DE R. Rothman	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.16 (MA4265)	Staff briefs CRGR on NEI 97-06 (NEI 97-06)	TBD	DE T. Sullivan E. Murphy	
1.17 (MA4265)	Publish SE on NEI 97-06 in FR for public comment (NEI 97-06)	TBD	DE C. Lauron	
1.18 (MA4265)	ACRS review of NEI 97-06 (NEI 97-06)	TBD	DE T. Sullivan E. Murphy	
1.19 (Later)	Issue generic communication related to steam generator operating experience and status of steam generator issues	09/28/01 (T)	DE Z. Fu	
1.20 (MA4265)	Staff briefs Commission on endorsing NEI 97-06 (NEI 97-06, and WITS Item 199400048)	TBD	DE T. Sullivan	
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	DE C. Lauron	
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)	IRO F. Congel	
2.2 (MB0258)	Establish NRC web site for Steam Generator Action Plan	01/16/01 (C)	DLPM R. Ennis	
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)	DLPM R. Ennis	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
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)	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)	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)	TBD	DLPM J. Zimmerman	
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)	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. (Attachment 3 of Ref. 1; TG: 6b, 6d, 6e; page 6 of Ref. 1)	08/31/01 (T)	DLPM J. Zimmerman	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
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	
	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	
	c) Perform additional sensitivity studies as needed.	06/30/03 (T)	RES J. Uhle	
	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

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.1 (continued)	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
	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 E. Thornbury

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
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. (Pgs. 10-11) (See Notes 3 and 5) Specific tasks include: a) Complete tests of jet impingement under MSLB conditions. b) Conduct long duration tests of jet impingement under severe accident conditions. c) Document results from tasks 3.2a and 3.2b.	 12/31/01 (T) 12/31/01 (T) 12/31/01 (T)	 RES J. Muscara RES J. Muscara RES J. Muscara	 DE E. Murphy DE E. Murphy 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. (Pgs. 12-13) (See Notes 3 and 5)	09/30/04 (T) See Note 2	RES R. Lee	
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. (Pgs. 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). b) Re-evaluate existing system level code assumptions and simplifications.	 10/01/01 (T) 12/31/01 (T)	 RES C. Tinkler RES C. Tinkler	 SRXB W. Jensen SRXB W. Jensen

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	c) Examine 1/7 scale data to assess tube to tube temperature variations and estimate variations for plant scale.	08/31/02 (T)	RES C. Tinkler	SRXB W. Jensen
	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 C. Tinkler	SRXB W. Jensen
	e) Examine SG tube severe accident T-H conditions using computational fluid dynamics (CFD) methods. This includes the following:			
	e.1) Benchmark CFD methods against 1/7 scale test data.	08/31/01 (T)	RES C. Boyd	SRXB W. Jensen
	e.2) Perform full scale plant calculations (hot leg and SG) for a 4 loop Westinghouse design. Evaluate scale effects.	12/31/01 (T)	RES C. Boyd	SRXB W. Jensen
	e.3) Perform plant analysis to address the effects on inlet plenum mixing resulting from tube leakage and hot leg orientation (CE design impact).	07/31/02 (T)	RES C. Boyd	SRXB W. Jensen
	f) Examine the uncertainty in the T-H conditions associated with core melt progression.	01/31/03 (T)	RES C. Tinkler	SRXB W. Jensen
	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 C. Tinkler	SRXB W. Jensen

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	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
	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
	h.3) Conduct large scale tests if needed.	11/30/05 (T)	RES J. Muscara	DE E. Murphy
	i) Develop data and analyses for predicting leak rates for degraded tubes in restricted areas under design basis and severe accident conditions.	12/31/03 (T)	RES J. Muscara	DSSA S. Long DE E. Murphy
	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).	06/30/04 (T)	DSSA S. Long	DE E. Murphy RES J. Muscara

Item No. (TAC No.)	**Milestone**	**Date** **(T=Target)** **(C=Complete)**	**Lead**	**Support**
3.4 (continued)	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).	02/28/05 (T)	DSSA S. Long	DE E. Murphy RES J. Muscara C. Tinkler E. Thornbury
3.5	Develop improved methods for assessing the risk associated with SG tubes under accident conditions. (Pgs. 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.	03/29/02 (T)	RES E. Thornbury	DSSA S. Long
	b) Development of improved methods for identifying accident scenarios (including MSLB) that lead to challenges on the reactor coolant pressure boundary.	06/28/03 (T)	RES E. Thornbury	DSSA S. Long
	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)	RES E. Thornbury	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
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. (Pgs. 47, 33)	12/31/01 (T)	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. (Pgs. 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)	01/31/02 (T)	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 radionuclide release to be used in the safety analyses of design basis events. (Pgs. 48, 38-44) (See Note 5) Specific tasks include:		DSSA J. Hayes	
	a) Assess Adams and Atwood and Adams and Sattison spiking data with respect to the ACRS comments.	10/31/01 (T)		
	b) Based upon the assessment performed in task 3.9a, develop a response to the ACRS comments.	12/31/01 (T)		
	c) Publish in the Federal Register for public comment, the response to ACRS' comments.	2/15/02 (T)		
	d) Complete review of public comments.	6/30/02 (T)		
	e) Based upon task 3.9d, determine if additional work needs to be performed.	8/15/02 (T)		

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
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: (Pgs. 20-29) (See last sentence in Note 3) Specific tasks include:			
	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.	12/31/05 (T)	RES J. Muscara	DE E. Murphy
	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.	12/31/06 (T)	RES J. Muscara	DE E. Murphy
	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)	RES J. Muscara	DE E. Murphy

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
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.	12/31/05 (T)	DLPM J. Zimmerman	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 2002.

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).

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 and efficient.

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 3, 2000	Issuance of RIS 2000-22, "Issues Stemming from NRC Staff Review of Recent Difficulties Experienced in Maintaining Steam Generator Tube Integrity" (SG Action Plan Item No. 1.1).
- November 15, 2000	Briefing of Commissioner T/As on IP2 Lessons-Learned Report and Steam Generator Action Plan
- November 16, 2000	Issuance of "Steam Generator Action Plan" via memorandum from B. Sheron/J. Johnson to S. Collins (Accession No. ML003770259).
- December 5, 2000	Meeting between NRC (Incident Response Operations) and U.S. Secret Service (SG Action Plan Item No. 2.1).
- December 6-7, 2000	Briefing of Commissioners Diaz and McGaffigan on steam generator issues.
- December 15, 2000	Issuance of letter from NRC to NEI requesting that NEI review the IP2 Lessons-Learned report and work with the staff to resolve the industry recommendations (Accession No. ML003777691).
- December 20, 2000	Meeting with NEI and other stakeholders to discuss the status of steam generator issues and potential resolutions (SG Action Plan Item No. 1.2).
- January 16, 2001	Steam Generator Action Plan website established (SG Action Plan Item No. 2.2).
- January 17, 2001	Briefing for DRP/DRS Counterparts Meeting on IP2 Lessons-Learned Report and SG Action Plan.
- January 18, 2001	Issuance of letter from NRC to NEI notifying NEI that the NRC will resume the review of NEI 97-06 (Accession No. ML010190317).
- February 1, 2001	ACRS Ad Hoc Subcommittee report related to SG DPO issued (NUREG-1740).
- February 13, 2001	Briefing for NRR Leadership Team on resource estimates for FY2001 for activities associated with the SG Action Plan.
- February 15, 2001	NRC (IRO) letter to U.S. Secret Service (Accession No. ML010460485) that provides the protocol for communications between the two organizations during a radiological emergency (SG Action Plan Item No. 2.1).
- February 16, 2001	Senior Management Meeting with NEI and other stakeholders to discuss SG Action Plan and activities associated with NEI 97-06.

- February 27-28, 2001	Steam Generator Workshop held at Bethesda Holiday Inn with NEI and other stakeholders (SG Action Plan Item No. 1.15).
- February 28, 2001	Public meeting between the NRC, NEI, Argonne National Laboratory and the industry to discuss industry actions relative to the SG Action Plan.
- March 5, 2001	Issuance of memorandum from EDO to DPO author stating that DPO is closed based on issuance of the ACRS Ad Hoc Subcommittee report (Accession No. ML010660353).
- March 5, 2001	Issuance of memorandum from EDO to Directors of NRR and RES tasking the staff to develop an action plan to address the conclusions and recommendations in the Ad Hoc Subcommittee report on the SG DPO (Accession No. ML010670217).
- March 23, 2001	Issuance of memorandum providing a revision to the SG Action Plan and documenting completion of Item Nos. 1.1, 1.2, 1.3, 1.4, 1.7, 1.8, 1.15, 2.1, and 2.2 (Accession No. ML010820457).
- March 30, 2001	Issuance of a memorandum documenting completion of Item No. 1.9 (Accession No. ML010920112).
- April 3, 2001	Issuance of a memorandum documenting completion of Item Nos. 2.4 and 2.5 (Accession No. ML010890426).
- April 4, 2001	Briefing for Office of the Inspector General (OIG) on the SG Action Plan.
- April 12, 2001	Issuance of memorandum documenting completion of Item No. 2.3 (Accession No. ML01120026).
- April 24, 2001	Issuance of memorandum from DPO author to Commission concerning issues related to conclusions in ACRS report (Accession No. ML011150011).
- April 26, 2001	Public meeting between the NRC and NEI to discuss the SG Action Plan and the generic change package technical specifications/technical requirements manual wording.
- April 30, 2001	Issuance of memorandum documenting completion of Item Nos. 1.10 and 1.12 (Accession No. ML011220621).
- April 30, 2001	Issuance of memorandum documenting completion of Item No. 1.11a (Accession No. ML011210293).
- May 4, 2001	Issuance of a memorandum from Chairman to EDO requesting a review of the memorandum from DPO author to Commission dated April 24, 2001 (Accession No. ML011290377).
- 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).
- May 17, 2001	Briefing for the EDO on the status of the SG Action Plan.

- May 30, 2001 Public meeting between the NRC and NEI to discuss adequacy of condition monitoring under a NEI 97-06 steam generator regulatory framework.

- June 1, 2001 Issuance of memorandum from EDO to ACRS transmitting SG Action Plan revision dated May 11, 2001 (Accession No. ML011430210).

- June 1, 2001 Issuance of memorandum from EDO to Chairman providing review of the memorandum from DPO author to Commission dated April 24, 2001 (Accession No. ML011370180).

NRR Technical Contacts: April Smith, DE/EMCB, 415-1356
Ed Kleeh, DIPM/IIPB, 415-2964
Steve Long, DSSA/SPSB, 415-1077

NRR Lead PM: Jake Zimmerman, DLPM, 415-2426

RES Contact: Joe Muscara, 415-5844

OKONITE CABLE LOCA TEST FAILURES

TAC Nos. MA8193, MA9199, MA9200, & MA9201

Last Update: 06/30/01
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.	TBD
b. If industry sufficiently addresses the issues and several plants are affected, the staff will publish a Regulatory Issue Summary in accordance with SECY-99-143.	TBD
c. If the industry initiative is inadequate, the staff will issue a generic letter to licensees to obtain information on affected safety-related equipment and plants.	TBD

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

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. NRC is also waiting for a response from Okonite regarding the Part 21 aspects of this issue and on their plans for additional cable testing.

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

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.

EMERGENCY ACTION LEVEL GUIDANCE DEVELOPMENT

TAC No.: MA3695	Revision to NESP-007	Last Update: 06/29/01
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 (On hold due to spent fuel pool study impact)	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.

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

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

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

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

References:
1. NUREG-0654/FEMA-REP-1, "Criteria for the Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, November 1980.
2. NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990," June 1990.
3. NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," September 1993.
4. NUMARC/NESP-007, Revision 2, "Methodology for Development of Emergency Action Levels," January 1992.

5. Regulatory Guide 1.101, Rev. 3, “Emergency Planning and Preparedness for Nuclear Power Reactors,” August 1992.
6. Letter from A. Nelson to J. Roe, September 16, 1997.
7. Memorandum from J. Taylor to T. Murley, June 21, 1990.
8. Letter from B. Zalcman to A. Nelson, March 13, 1998.
9. Memorandum from S. Magruder to T. Essig, June 26, 1998.
10. Letter from C. Miller to A. Nelson, August 3, 1998.
11. Letter from A. Nelson to C. Miller, August 13, 1998.
12. Letter from A. Nelson to T. Essig, January 11, 1999.
13. Letter from T. Essig to A. Nelson, May 11, 1999.
14. Memorandum from J. Larkins to W. Travers, June 3, 1999.
15. Memorandum from J. Larkins to W. Travers, September 10, 1999.
16. Letter from J. Birmingham to A. Nelson, August 8, 2000.
17. Memorandum from J. Larkins to W. Travers, September 7, 2000.
18. Email from M. Federline to J. Birmingham, September 18, 2000.

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

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

ECCS SUCTION BLOCKAGE

TAC Nos. MA6454, M98500, MA0704, MA2452, MA4014, MA6204, and MA0698

Last Update: 07/01/01
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"	
○ Complete review of licensee responses	03/01 (C)
○ Complete audits of 4-6 plants	08/00 (C)
○ Complete hydrodynamic load review	09/01 (T)
○ Evaluate impact of coatings research on BWR resolution	03/01 (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 RG 1.1/RG 1.82 (DG-1107)	9/02 (T)
PART III: **CONTAINMENT COATINGS**	
1. GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"	
○ Complete review of licensee responses	12/99 (C)
○ Complete revision of RG 1.54	07/00 (C)
2. NRC-sponsored research program on the potential for coatings to fail during an accident	
○ Phase I analytical evaluation/coating degradation model	12/98 (C)
○ Phase II test program to validate model and test key parameters	03/01 (C)
○ Evaluate need for regulatory action based on research results	03/01 (C)
PART IV: **GSI 191, "ASSESSMENT OF DEBRIS ACCUMULATION ON PRESSURIZED WATER REACTOR SUMP PERFORMANCE"**	
1. NRC-sponsored research program on the potential for loss of ECCS NPSH during a LOCA due to clogging by debris	
○ Preliminary (qualitative) risk assessment (NRR)	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)	12/01 (T)
○ Evaluate need for regulatory action based on research program results (NRR)	03/02 (T)

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

This plan has four parts. First, for boiling-water reactors (BWRs), this issue has been addressed by licensee responses to NRCB 96-03. 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. Second, the adequacy of licensee (both PWR and BWR) net positive suction head (NPSH) calculations was evaluated through NRR review of licensee responses to Generic Letter (GL) 97-04, “Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps,” dated October 7, 1997. The third part of the plan consists of two efforts by the staff. The first effort assessed the adequacy of the implementation and maintenance of current licensee coating programs through NRR review of licensee responses to GL 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,” dated July 14, 1998. The second effort is a research program to assess the potential for coatings to become debris, including the timing of any failures that might occur, and the cause and the characteristics of the debris. These two efforts combined will provide NRR the necessary technical bases on which to assess the potential threat to the ECCS by coating debris and the adequacy of current coating licensing bases (both PWR and BWR). The results of these two programs will also feed into the fourth part of the action plan: an evaluation of the potential for clogging of PWR ECCS recirculation sumps during a LOCA. As with the coating research discussed above, this part of the plan is being conducted by the Office of Nuclear Regulatory Research (RES). RES is evaluating the potential for PWR sumps to become clogged during an accident based on new information learned during the development of NRCB 96-03 for the BWRs.

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

Since the resolution of USI A-43, new information has arisen which challenged the adequacy of the NRC’s conclusion that no new requirements were needed to prevent clogging of ECCS strainers in boiling-water reactors (BWRs). On July 28, 1992, an event occurred at Barsebäck Unit 2, a Swedish boiling-water reactor (BWR), which involved the plugging of two containment vessel spray system (CVSS) suction strainers. The strainers were plugged by mineral wool insulation that had been dislodged by steam from a pilot-operated relief valve that spuriously opened while the reactor was at

435 psig. Two of the three strainers on the suction side of the CVSS pumps that were in service became partially plugged with mineral wool. Following an indication of high differential pressure across both suction strainers 70 minutes into the event, the operators shut down the CVSS pumps and backflushed the strainers. The Barsebäck event demonstrated that the potential exists for a pipe break to generate insulation debris and transport a sufficient amount of the debris to the suppression pool to clog the ECCS strainers.

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

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

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

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

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

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

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

The third effort involves the evaluation of coatings as a potential debris source. Concerns raised in this area are due to recent events where qualified coatings have failed during normal operation at a number of sites. The failure of qualified coatings during normal operation has led to two specific staff concerns. The first concern is whether the qualification of coatings is adequate to ensure that coatings do not pose a potential threat to the ECCS. Accordingly, the staff has begun a research effort led by RES to evaluate the potential for coatings to become debris during an accident and consequently, become a threat to the ECCS performing its safety function. The second concern relates to the adequacy of licensee programs to apply and maintain coatings consistent with their licensing bases. This concern 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 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 has begun a program to evaluate PWR sump designs and their susceptibility to blockage by debris. This evaluation will include a detailed risk assessment. Risk insights will be used to support any conclusions drawn relative to the need for licensees to address the potential for ECCS suction clogging.

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

Originating Document: Not Applicable.

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

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

The staff believes that there is sufficient new information and concerns raised relative to the potential for debris clogging in PWRs that part of this action plan has been prepared to address PWR sump blockage concerns. However, it is not clear whether a significant threat to PWR ECCS operation exists. The staff believes that continued operation of PWRs is justified because of PWR design features which would tend to prevent blockage of the ECCS sumps during a LOCA. These features would tend to be effective for insulation and coating debris. For instance, the containments in PWRs tend to be very compartmentalized making the transport of debris to the sump screens difficult. In addition, PWRs typically do not need to switchover to recirculation from the sump during a LOCA until 20-30 minutes after the accident initiation allowing time for much of the debris to settle in other places within the containment. Coating debris, in particular, would have plenty of time to settle. Clearly, the results of the staff's research program are needed before a final conclusion regarding the potential to clog the ECCS sump can be reached. In addition to these design considerations, the staff considers continued operation of PWRs to be justified because the probability of the initiating event (i.e., large break LOCA) is extremely low. More probable (although still low probability) LOCAs (small, intermediate) will require less ECCS flow, take more time to use up the water inventory in the refueling water storage tank (RWST), and in some cases may not even require the use of recirculation from the ECCS sump because the flow through the break would be small enough that the operator will have sufficient time to safely shut the plant down. In addition, all PWRs have received approval by the staff for leak-before-break (LBB) credit on their largest RCS primary coolant piping. While LBB is not acceptable for demonstrating compliance with 10 CFR 50.46, it does demonstrate that LBB-qualified piping is of sufficient toughness that it will most likely leak (even under safe shutdown earthquake conditions) rather than rupture. This, in turn, would allow operators adequate opportunity to shut the plant down safely (although debris generation and transport for an LBB size through-wall flow will still be investigated). Additionally, the staff notes that there are sources of margin in PWR designs which may not be credited in the licensing basis for each plant. For instance, NPSH analyses for most PWRs do not credit containment overpressure (which would likely be present during a LOCA). Any containment pressure greater than assumed in the NPSH analysis provides additional margin for ECCS operability during an accident. Another example of margin would be that it has been shown, in many cases, that ECCS pumps would be able to continue operating for some period of time under cavitation conditions. Some licensees have vendor data demonstrating this. Design margins such as these examples may prevent complete loss of ECCS recirculation flow or increase the time available for operator action (e.g., refilling the RWST) prior to loss of flow.

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

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

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

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

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

These conclusions clearly support this action plan as outlined herein.

<u>Current Status</u>: The review of NRCB 96-03 responses is complete. The staff has completed four audits No safety issues have been identified in the audits. A summary report has been prepared documenting the staff's review of the strainer issue, and a memo citing the report as the basis for closing the multiplant action (MPA) is being prepared. Upon the issuance of the memo, the MPA will be closed.

The staff has also completed its review of the hydrodynamic load test program for the Mark III strainer design. No safety issues were identified in the staff's review of the Mark III hydrodynamic load test program. The only remaining issue in the hydrodynamic loads area is in the staff's review of GE's Licensing Topical Report on their strainer design. One open issue remains relating to the need for additional testing to adequately determine the hydrodynamic mass of the strainers. The staff sent GE a letter requesting that they perform additional testing to resolve the issue, and GE declined to perform additional tests. The staff obtained the services of a hydrodynamic load expert (contractor) to assist in resolving the issue with GE. Based on the contractor's report, the staff has sent another letter to GE (dated June 21, 2001) requesting additional testing. The letter further states that in the absence of a plan to complete additional testing, the staff plans to issue its final evaluation of the GE Licensing Topical Report NEDC-32721P, by September 30, 2001. In the absence of an approved topical report, the NRC would contact affected licensees to request appropriate action to resolve the issue on a plant-specific basis. The staff is currently awaiting GE's response to this letter.

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. This consideration, in addition to the low probability of the initiating event and the difficulty of transporting the debris to the sump given the circuitous geometry of a containment flow path, leads to the conclusion that the overall schedule for resolving the sump/strainer clogging issue relative to coatings is appropriate.

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 ongoing. 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. RES has shared the results of the parametric evaluation with NRR staff and management. In addition, as part of the resolution of GSI-191, RES is assessing the risk associated with sump blockage based on the parametric evaluations. This risk assessment is ongoing and scheduled for completion by the end of July 2001. Once RES has completed its evaluation of GSI-191, RES and NRR will work together to determine the appropriate course of action.

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 will be presenting the results of the GSI-191 parametric evaluation to the ACRS on July 12, 2001. Also, a public meeting between the NRC, the Nuclear Energy Institute, and the three Pressurized Water Reactor Owners' Groups has been scheduled by NRR for July 26 and 27, 2001, to discuss the parametric evaluation. The staff will continue to hold regular public meetings with the three PWR owners groups and NEI to keep them informed on the progress of the GSI-191 research program.

NRR Lead PMs:	Donna Skay, 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)
NRR Lead Technical Reviewer:	Rob Elliott, SPLB, 415-1397
NRR Technical Contacts:	Jim Davis, EMCB, 415-2713 Rich Lobel, SPLB, 415-2865 Kerri Kavanagh, RTSB, 415-3743 Nicholas Saltos, SPSB, 415-1072
RES Technical Contacts:	Michael Marshall, ERAB, 415-5895 John Boardman, ERAB, 415-6534

References:

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

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

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

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

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

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

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

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

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

GL 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,” dated July 14, 1998.

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

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

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 Draft Technical Report, entitled, "GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," Dated July 2001 (Accession Number ML011860039).

CONTROL ROOM HABITABILITY
(NEW)

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

Last Update: Initial Update
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, Control Room Envelope Integrity Testing, Design Basis Accident Radiological Analyses, and Meteorology for Control Room Habitability Assessment	07/01/01 (C)
3.	Office review of draft Regulatory Guides and Generic Letter	08/01/01 (T)
4.	Brief ACRS on draft Regulatory Guides and Generic Letter	10/01 (T)
5.	Brief CRGR on draft Regulatory Guides and Generic Letter	11/01 (T)
6.	Issue draft Regulatory Guides and Generic Letter for public comment	12/01 (T)
7.	Public meeting on Regulatory Guides and Generic Letter	01/02 (T)
8.	Resolve public comments on Regulatory Guides	02/02 (T)
9.	Office review of final Regulatory Guides and Generic Letter	03/02 (T)
10.	Brief ACRS on final Regulatory Guides and Generic Letter	05/02 (T)
11.	Brief CRGR on final Regulatory Guides and Generic Letter	06/02 (T)
12.	Issue final Regulatory Guides and Generic Letter	07/02 (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 25 percent of the operating plants' control rooms. In all of the tests performed to date, 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 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.

<u>Historical Background</u>: 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.

<u>Proposed Actions</u>: 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 25 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.

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).

<u>Originating Document</u>: None.

<u>Regulatory Assessment</u>: 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: The staff has completed preparation for office review of the draft generic letter and 4 draft regulatory guides on control room habitability, control room envelope integrity testing, meteorology for control room habitability assessment, and design basis accident radiological analyses.

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, 35
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

JULY 2001 DIRECTOR's QUARTERLY STATUS REPORT

Open Generic Communications and Compliance Activities

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

July 12, 2001

DIRECTOR's MONTHLY STATUS REPORT
July 2001

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

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
Division of Engineering						
Materials and Chemical Engineering Branch						
MA6018	IN	EJBenner	--/--/--	12/31/2001 T	IN: IN 96-09, Sup 2; Damage in Foreign Steam Generator Internals	To inform licensees of steam generator tube degradation in foreign PWRs as a result of sludge lancing activities.
						EMCB has 1 GCCA(s)
Mechanical & Civil Engineering Branch						
MA2314	IN	RABenedict	--/--/--	09/30/2001 T	IN: Potential Inadequacies in the Installation of Check Valves Made by Anderson Greenwood and Borg Warner	Discusses maintenance errors that could make the check valves
						EMEB has 1 GCCA(s)
						DE has a total of 2 GCCA(s)
Division of Regulatory Improvement Programs						
Events Assmt, Gen Comms & Non-Power Reactor Branch						
MA8819	RI	JWShapaker	--/--/--	09/30/2001 T	RIS: SG Tube Integrity - Industry	To document industry initiative on SG Tube Integrity.
MA9204	IN	CVHodge	07/25/01	07/30/2001 T	IN: Potential IN on Rigging Problems	Rigging Problems
MA9474	RI	JWShapaker	--/--/--	09/30/2001 T	RIS: Procedure for Conducting Meetings with Proprietary Content	To inform industry and other stakeholders re the conduct of meetings with proprietary content.
MA9821	LT	EFGoodwin	--/--/--	08/22/2001 T	LTF: Westinghouse Spent Fuel Pool/Pellet Density	To follow Westinghouse calculations correcting higher than design pellet density affecting pool criticality.
MA9992	RI	JWShapaker	--/--/--	09/30/2001 T	RIS: Format and Content of No Significant Hazard	Provides informational guidance to licensees.
MA9993	RI	JWShapaker	--/--/--	09/30/2001 T	RIS: Acceptable Format for Oath or Affirmation	Provides informational guidance to licensees.

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

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MB0371	IN	EFGoodwin	--/--/--	08/30/2001 T	IN: Debris in Standby Liquid Control System Storage Tanks	Fragments of plastic bags used for chemicals were left in SLIC tanks and might disable SLIC pumps.
MB0420	RI	JWShapaker	--/--/--	09/30/2001 T	RIS: Cumulative Effects of Operator Workarounds	RIS to address staff policy/procedure regarding operator workarounds (based on findings of TI).
MB0703	RI	CVHodge	--/--/--	09/30/2001 T	RIS: On Improvements in Distribution of Generic Communications (GC)	Staff's proposal to use email messages with hyperlinks to disseminate GCs and to ask addressees to voluntarily inform NRC of their willingness to accept electronic msgs linked to new generic comms on the NRC web, instead of paper or electronic copies.
MB0858	RI	JWShapaker	--/--/--	09/30/2001 T	RIS: Submitting Security Plan Changes	Proposed RIS clarifying the correct regulatory process for submitting security plan changes.
MB1037	IN	CDPetrone	07/31/01	08/30/2001 T	IN: Hydrogen Fire & Hydrogen Storage at Nuclear Power	To inform addressees to potential hazards associated with hydrogen storage facilities. Notice discusses the results of an NRC survey of hydrogen storage facilities at nuclear power plants.
MB1120	IN	EFGoodwin	--/--/--	09/30/2001 T	IN: Deficiencies in Work Packages Under Sec. 11, ASME Code	Level II examiner had not reviewed and signed work packages as required by ASME Code, Section 11.
MB1167	RI	CDPetrone	12/31/01	09/30/2001 T	RIS: BWROG DC Powered MOV Output Issue	To inform addressees of the closure of the dc-powered MOV output issue by referencing the availability of the BWROG methodology.
MB1340	IN	CVHodge	09/12/01	09/30/2001 T	IN: Holtec Part 21 on Excess Weight Found in Spent Fuel Racks	Holtec International reported finding that weights in spent fuel racks removed from the Byron station were 50 percent greater than those used in design analyses.
MB1382	IN	EFGoodwin	--/--/--	09/12/2001 T	IN: Occupational Exposure Due to Hot Particles	Inadequacies in health physics resulted in higher than expected deep dose from cobalt-60 particles.
MB1505	IN	ENFields	--/--/--	09/30/2001 T	IN: Susquehanna SLC System Inop Due to Power Upgrade	Describes the inoperability of the standby liquid control system as a result of a recent power upgrade at the Susquehanna plant.
MB1537	IN	ENFields	--/--/--	09/30/2001 T	IN: Fitness-For-Duty Performance Data - Year 2000	Summarizing fitness-for-duty program performance reports for CY 2000
MB1573	RI	ENFields	--/--/--	09/30/2001 T	RIS: Proposed Changes to Performance Indicators	To inform addressees of a 6-month pilot program to evaluate changes to the performance indicator that tracks "unplanned power changes per 7,000 hours".

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

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MB1622	IN	EFGoodwin	--/--/--	10/03/2001 T	IN: Guide Tube Failures In Westinghouse Lopar Fuel Assemblies	Guide tube failures in Westinghouse lopar fuel assemblies.
MB1758	RI	ENFields	--/--/--	09/30/2001 T	RIS: RCIC Systems Failures	Position on reportability requirements for reactor core isolation cooling system failures
MB1952	RI	ENFields	--/--/--	07/31/2001 T	RIS: Deficiencies in DB Radiological Analysis Documentation	Deficiencies in design basis radiological analysis documentation submitted in conjunction with license amendment requests.
MB1978	RI	ENFields	--/--/--	09/22/2001 T	RIS: Attributes of a Proposed NSHC Determination	Provides licensees with guidance on preparing a no significant hazards consideration analysis for a license amendment request.
MB2060	BL	JWShapaker	12/31/02	12/30/2001 T	BL: Urgent Generic Communication - CRDM Nozzle Cracking	Circumferential cracking of reactor pressure vessel head penetration nozzles
MB2092	RI	ENFields	--/--/--	09/05/2001 T	RIS: Update of Evacuation Estimates	Alerts addressees to the possible need to update emergency planning evacuation time estimates as the results of the year 2000 census become available.
MB2112	RI	ENFields	--/--/--	08/31/2001 T	RIS: Lessons Learned - Decommissioning/License Termination	Provides licensees with information that may help them develop more complete decommissioning plans and license termination plans.

REXB has 25 GCCA(s)

DRIP has a total of 25 GCCA(s)

There are a total of 27 GCCA(s)

NOTES:

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

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

DIRECTOR's MONTHLY STATUS REPORT
July 2001

Generic Communication and Compliance Activities Added
Since April 19, 2001

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Added
MB1758	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	09/30/2001 T	RIS: RCIC Systems Failures	4/19/01: TAC approved by C. Petrone.
MB1952	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	07/31/2001 T	RIS: Deficiencies in DB Radiological Analysis Documentation	5/8/01: TAC approved by C. Petrone
MB1978	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	09/22/2001 T	RIS: Attributes of a Proposed NSHC Determination	5/22/01: TAC approved by C. Petrone
MB2060	BL	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	12/31/02	12/30/2001 T	BL: Urgent Generic Communication - CRDM Nozzle Cracking	5/30/01: TAC approved by C. Petrone.
MB2092	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	09/05/2001 T	RIS: Update of Evacuation Estimates	6/5/01: TAC approved by C. Petrone
MB2112	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--/--	08/31/2001 T	RIS: Lessons Learned - Decommissioning/License Termination	6/6/01: TAC approved by C. Petrone
MB2265	BL	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch			BL- Circumferential Cracking of RPV Head Penetration Nozzles	6/28/01: TAC withdrawn (duplicated, see TAC

NOTES:

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

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

Total Number of Records = 7

DIRECTOR's MONTHLY STATUS REPORT
July 2001

Generic Communication and Compliance Activities Closed
Since April 19, 2001

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Closed
MA7871	IN	DLSkeen	Electrical & Instrumentation & Controls Branch	05/10/01 P	05/10/2001	IN: Potential Aging Degradation of Medium Voltage Power Cables	5/10/01: IN cancelled.
MB0414	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	05/11/01 C	05/11/2001	IN: Unescorted Access Granted on the Basis of Incomplete and/or Inaccurate Information	5/11/01: IN 2001-07 issued.
MB0419	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	05/23/01 P	05/23/2001	RIS: Non-Conservatisms in PWR SFP Storage Reactivity Equivalencing Calculations	5/18/01: RIS 2001-12 issued.
MB1087	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	05/19/01 P	05/19/2001	IN: Centrifugal Charging Pump Thrust Bearing Damage Not Detected Due to Inadequate Assessment of Oil Analysis Results…	5/11/01: IN 2001-06 issued.
MB1149	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	04/20/01 P	04/20/2001	RIS: Revising Approved LTPs w/o NRC Review of Approval NMSS	4/16/01: TAC withdrawn
MB1346	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	04/20/01 P	04/20/2001	RIS: Emergency Response Correspondence to IRO not AEOD	4/18/01: TAC withdrawn
MB1737	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	05/23/01 P	05/23/2001	RIS: Voluntary Submission of Performance Indicator Data	5/11/01: RIS 2001-11 issued.
MB1897	IN	RDTelson	Events Assmt, Gen Comms & Non-Power Reactor Branch	06/20/01 P	06/20/2001	IN: Main Feedwater System Degradation in Safety-Related ASME Code Class 2 Piping Inside the Containment of a PWR	6/12/01: IN 2001-09 issued
MB1943	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	06/28/01 P	06/28/2001	IN: Failure of Central Sprinkler Company Model GB Series Fire Sprinkler Heads	6/28/01: IN 2001-10 issued.

Generic Communication and Compliance Activities Closed
Since April 18, 2001

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Closed
MB1947	IN	ICJung	Events Assmt, Gen Comms & Non-Power Reactor Branch	05/12/01 C	05/11/2001	IN: Through-Wall Circum. Cracking of Reactor Pressure Vessel Head CRDM Penetration Nozzles at Oconee	4/30/01: IN 2001-05 issued.

NOTES:

Total Number of Records = 10

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

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

ATTACHMENT 3

RISK-INFORMED INITIATIVES

RISK-INFORMED INITIATIVES

INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
1. Revised Oversight Process			
- Performance indicators (PIs)	- analysis of PIs	- development of risk-based PIs - analysis/trending of PIs	- continue development and possible implementation of risk-based PIs
- Plant & system reliability studies	- developed databases to track LERs and common-cause failures (CCFs)	- analyzing data on reliability and CCFs	- work with industry to develop consistent approach for safety system unavailability reporting
- Significance determination process (SDP)	- developed SDP - ROP action matrix	- implementing/improving SDP	- issue plant specific SDP notebooks - revise ALARA, physical protection, SDP - evaluate fire protection, shutdown, external events, concurrent deficiences
2. Risk-informed Licensing Actions	Developed guidance documents - general guidance (RG 1.174 and SRP chapter 19) - IST (RG 1.175 and SRP section 3.9.7) - GQA (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	Updating guidance documents Reviewing increasing number of risk-informed amendments	Publish revisions to guidance documents Evaluate additional industry proposals (e.g., eliminate PASS requirements, extend ILRT interval)

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	Reviewing submittals for initiatives 1-3	Continue reviews of initiatives
4. Maintenance Rule	- New section (a)(4) effective 11/28/00 - RG 1.182 endorses industry guidance document for managing risk during maintenance activities	Coordinating implementation with risk-informed technical specifications	Effectiveness review
5. Fire protection	- NFPA-805 issued in April 2001 - alternative performance-based risk-informed fire protection standard for nuclear power plants. - Circuit Analysis Resolution Program	- staff working on proposed rulemaking that would endorse NFPA 805 as a voluntary alternative to NRC existing fire protection regulations - staff working with industry to develop risk-informed post-fire safe shutdown methodology document	

INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
6. Safeguards	- Proposed revisions to 10 CFR 73.55 sent to Commission 6/4/01 - Requires that licensees' security programs be based on risk-informed target sets of equipment necessary to prevent core damage and/or spent fuel sabotage		
7. 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	
8. Alternate source term	- New rule (10 CFR 50.67) published 12/23/99 - Allows for application of improved knowledge of fission product releases and plant performance	- Evaluating license amendments that take advantage of new rule	
9. RIP50/Option 2 (risk-informing scope of special treatment requirements)	- Published ANPR 3/00 - STPNOC exemptions reviewed	- Reviewing industry guidance documents - Pilot plants starting process	- Complete review of industry guidance documents - Review pilot plants results - Publish proposed and final rules (10 CFR 50.69)

INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
10. RIP50/Option 3 (risk-informing technical requirements)	- Developed framework document to guide future efforts - Completed detailed technical review and proposed changes to 10 CFR 50.44 - Completed feasibility study of risk-informed changes to 10 CFR 50.46	- Developing proposed rule changes for 50.44 - Developing technical basis for proposed changes to 50.46 and associated rules - Developing technical basis for risk-informed changes to 10 CFR 50.61	- Publish final revisions to 50.44 - Publish proposed and final rule changes to 50.46 - Publish proposed and final rule changes to 50.61
11. PRA standards	- Working with ASME on internal events standard - Working with ANS on low power and shutdown standard - Industry developing guidance on peer reviews	- Continuing work with ASME and ANS - Reviewing industry guidance on peer reviews	- Endorse industry standards generically or for specific applications (e.g., Option 2)
12. Creating a risk-informed environment	- Began effort within NRR to create environment in which risk-informed methods are fully integrated into staff activities	- Evaluating current environment	- Establish target environment - Implement target environment - Assess effectiveness
13. New regulatory licensing approach	- Exelon submitted risk-informed, top-down approach for licensing pebble bed modular reactors (PBMR) similar to General Atomics MHTGR approach in early 1990's.	- RES/NRR working group evaluating Exelon proposal - Ongoing meetings with Exelon	- NEI working with Exelon and developing a more generic approach for any new plant (framework modeled on ROP) -Commission paper planned in November on Exelon's approach

INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
14. Construction Inspection Program reactivation	- Use of risk insights in the Construction Inspection Program are being proposed by NEI.	- Ongoing meetings with NEI	