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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: ML030850023

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Mr. S. Scott
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1700 Rockville Pike, Suite 525
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Ms. Norena G. Robinson, Licensing Technician
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DIRECTOR'S STATUS REPORT

on

GENERIC ACTIVITIES

Action Plans

Generic Communication and
Compliance Activities

APRIL 2003

Office of Nuclear Reactor Regulation

INTRODUCTION

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

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

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

Attachment 2, "Open Generic Communications and Compliance Activities," lists potential generic issues that are safety significant, require technical resolution, and possibly require generic communication or action. The attachment consists of two lists: 1) Open GCCAs and 2) GCCAs closed since the previous report. The generic communications listed in the attachment include bulletins, generic letters, regulatory issue summaries (which replace administrative letters), and information notices. Compliance activities listed in the attachment do not rise to the level of complexity that require an action plan, and a generic communication is not currently scheduled.

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

ATTACHMENT 1

NRR ACTION PLANS

TABLE OF CONTENTS

BOILING WATER REACTOR INTERNALS

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

Last Update: 04/06/03
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/03 (T)
2. Review BWRVIP Re-inspection and Evaluation Criteria	
○ Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03)	. . 07/15/99 (CA)
○ BWRVIP-03, Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components	. . 07/15/99 (CA)
○ BWR Vessel Shell Weld Inspection Recommendations (BWRVIP-05)	. . 07/28/98 (CA)
○ BWR Axial Shell Weld Inspection Recommendations	. . 03/07/00 (CA)
○ Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07)	. . 04/27/98 (CA)
3. Review of generic repair technology, criteria, and guidance	. . . . . . TBD
4. Review generic mitigation guidelines and criteria	. . . . . . TBD
5. Review of generic NDE technologies developed for examinations of BWR internal components and attachments	. . . . . . TBD
6. Other Internals reviews (safety assessments, evaluations, mitigation measures, inspections, and repairs)	
○ Safety Assessment of BWR Reactor Internals (BWRVIP-06)	. . 09/15/98 (CA)
○ Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08 & BWRVIP-46)	. . 03/27/98 (CA)
○ Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14)	. . 12/03/99 (CA)
○ Internal Core Spray Piping and Sparger Replacement Design Criteria (BWRVIP-16)	. . 08/10/00 (CA)
○ Roll/Expansion of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)	. . 03/13/98 (CD)
○ BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)	. . 12/02/99 (CA)
○ BWRVIP-18, Appendix C, BWR Core Spray Internals Demonstration of Compliance With Technical Information Requirements of License Renewal Rule (10 CFR 54.21)	. . 09/06/00 (CA)
○ Internal Core Spray Piping and Sparger Repair Design Criteria (BWRVIP-19)	. . 08/10/00 (CA)
○ Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25)	. . 12/19/99 (CA)
○ Top Guide Inspection and Flaw Evaluation Guideline (BWRVIP-26)	. . 09/29/99 (CA)
○ Standby Liquid Control System / Core Plate ΔP Inspection and Flaw Evaluation Guidelines (BWRVIP-27)	. . 04/27/99 (CA)
○ Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)	. . 04/10/00 (CA)
○ Technical Basis for Part Circumferential Weld Overlay Repair of Vessel Internal Core Spray Piping (BWRVIP-34)	. . . 08/01/03 (CI)
○ 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)

MILESTONES	DATE (T/C)[1]
○ BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42)	. . 05/26/00 (CA)
○ Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)	. . 05/26/00 (CA)
○ BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47)	. . 03/27/98 (CA)
○ Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (BWRVIP-48)	. . 10/13/99 (CA)
○ Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49)	. . 09/29/99 (CA)
○ Top Guide / Core Plate Repair Design Criteria (BWRVIP-50)	. . . 01/29/01 (CI)
○ Jet Pump Repair Design Criteria (BWRVIP-51)	. . . 10/28/00 (CI)
○ Shroud Support and Vessel Repair Design Criteria (BWRVIP-52)	. . . 11/02/00 (CI)
○ Standby Liquid Control Line Repair Design Criteria (BWRVIP-53)	. . . 10/26/00 (CI)
○ Lower Plenum Repair Design Criteria (BWRVIP-55)	. . . 09/28/01 (CI)
○ LPCI Coupling Repair Design Criteria (BWRVIP-56)	. . . 03/22/02 (CI)
○ Instrument Penetrations Repair Design Criteria (BWRVIP-57)	. . . 05/01/02 (CI)
○ CRD Internal Access Weld Repair (BWRVIP-58)	. . . 10/17/01 (CI)
○ Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals (BWRVIP-59)	. . . 07/31/01 (CI)
○ BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants (BWRVIP-60)	. . 07/08/99 (CA)
○ Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)	. . . 01/30/01 (CI)
○ Shroud Vertical Weld Inspection and Evaluation Guidelines (BWRVIP-63)	. . . 04/18/00 (CI)
○ BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)	. . 07/27/01 (CA)
○ Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)	. . . 09/15/00 (CI)
○ BWR Core Shroud Inspection & Flaw Evaluation Guidelines (BWRVIP-76)	. . . 08/20/03 (T)
○ BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material (BWRVIP-78)	. . 02/01/02 (CA)
○ Evaluation of Crack Growth in BWR Shroud Vertical Welds (BWRVIP-80)	. . . 02/19/03 (CI)
○ Evaluation of Guidelines for Selection and Use of Material for Repairs to BWR Internals (BWRVIP-84)	. . . 08/20/03 (CI)
○ Evaluation of Guidelines for Selection and use of Materials for Repairs to BWR Internals (BWRVIP-86)	. . 02/01/02 (CA)
○ Guide for Format and Content of BWRVIP Repair Design Submittals (BWRVIP-95)	. . . 07/20/03 (T)
○ Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals (BWRVIP-96)	. . . 07/20/03 (T)
○ Guidelines for Performing Weld Repairs to Irradiated BWR Internals (BWRVIP-97)	. . . 07/20/03 (T)
○ Crack Growth Rates in Irradiated BWR Stainless Steel Internal Components (BWRVIP-99)	. . . 07/20/03 (T)
○ Updated Assessment of the Fracture Toughness of Irradiated SS for BWR Core Shrouds (BWRVIP-100)	. . . 07/20/03 (T)
○ Evaluation and Recommendations to Address Shroud Support Cracking in BWRS (BWRVIP-104)	. . . 09/01/03 (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. The schedule change for BWRVIP-76 is due to the staff waiting for the BWRVIP to supplement its original submittal in accordance with the open items in the staff's initial SE.

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

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

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

The staff is re-reviewing BWRVIP-17, "BWR Vessel and Internals Project Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels," as a permanent repair.

NRR Technical Contacts: Meena Khanna, EMCB, 415-2150
Jai Rajan, EMEB, 415-2788

NRR Lead PM: Meena Khanna, EMCB, 415-2150

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

Action Plan dated April 1995.

STEAM GENERATORS

TAC Nos.	Description	
M88885	Steam Generator (SG) Integrity Rulemaking	Last Update: 03/31/03
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	
MB2446	SG Risk Communication	
MB3794	SG Communication Plan	
MB7216	SG DPO Followup	

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

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

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

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.15 (MB0631)	Hold steam generator workshop with stakeholders (page 2 of Ref. 1; page 2 of Ref. 2)	02/27/01 (C) ML010820457	DE R. Rothman	
1.16 (MA4265)		Note 12		
1.17 (MA4265)		Note 12		
1.18 (MA4265)		Note 12		
1.19	Issue generic communication related to steam generator operating experience and status of steam generator issues	10/31/01 (C) ML020230299	DE Z. Fu	
1.20 (MA4265)	Staff issues a Commission Paper on regulatory framework (NEI 97-06, and WITS Item 199400048)	04/08/03 (T) Note 12	DE L. Lund	
1.21 (MA4265)	Staff issues: a) safety evaluation on lead plant submittal b) consolidated line item improvement process (CLIIP) item published in *FR*.	Note 12 TBD (T) TBD (T)	DE E. Murphy	
2.1	Evaluate the need for a new communication protocol with the U.S. Secret Service that would cover emergency situations at all NRC licensed facilities (Attachment 3 of Ref. 1)	12/05/00 (C) ML010460485 ML010820457	IRO F. Congel	
2.2 (MB0258)	Establish NRC web site for Steam Generator Action Plan	01/16/01 (C) ML010820457	DLPM R. Ennis	
2.3 (MB0258)	Review and revise, as appropriate, the policy for project manager involvement with the morning call between the resident inspectors and the region. (Attachments 3 and 4 of Ref. 1)	03/23/01 (C) ML011020026	DLPM R. Ennis	

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) ML010890426	DIPM T. D'Angelo	
2.5 (MB0737)	Develop, revise, and implement, as appropriate, a process for the timely dissemination of technical information to inspectors for inclusion in the inspection program (TG: 5g)	04/03/01 (C) ML010890426	DIPM G. Klinger	
2.6 (MB2446) (MB3794)	Incorporate experience gained from the IP2 event and the SDP process into planned initiatives on risk communication and outreach to the public (TG: 9) 1. Issue NRR input for incorporation into OEDO initiative 2. Address SRM dated 12/26/01	 01/31/02 (C) ML020590125 12/24/02 (C) ML023440202	PMAS M. Kotzalas	
2.7 (MB0258)	Investigate possibility of establishing protocol with OIG regarding review of draft reports for factual/contextual errors (page 8 of Ref. 1)	06/18/01 (C) ML011720125	DLPM R. Ennis	
2.8 (MB0633)	Review and revise, as appropriate, the amendment review process, including concurrence responsibilities, supervisory oversight, and second-round requests for additional information.			

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
2.8 (continued)	a. Issue OI LIC-101 b. Issue procedure for NRR and RES interactions (Attachment 3 of Ref. 1; TG: 6b, 6d, 6e; page 6 of Ref. 1)	08/31/01 (C) 02/27/02 (C) ML020580484	DLPM M. Banerjee DLPM M. Fields	
3.1 (MB7216)	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 (C) ML023610586	RES J. Uhle	DSSA W. Jensen
	b) Perform T-H assessment of flow-induced vibrations during MSLB. Using the T-H conditions calculated during the transient, generate a conservative estimate of flow-induced vibration displacement and frequency assuming steady state behavior.	12/31/02 (C) ML023610586	RES J. Uhle	DSSA W. Jensen

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.1 (continued)	c) Perform additional sensitivity studies as needed.	06/30/03 (T)	RES J. Uhle	SSA W. Jensen
	d) Obtain information from existing analyses related to loads and displacements (axial, bending, cyclic) experienced by SG structures under MSLB conditions.	12/31/02 (C) ML023610586	RES J. Muscara	
	e) Using information from tasks 3.1a, 3.1b, and 3.1d, estimate upper bound loads and displacements.	12/31/02 (C) ML023610586	RES J. Muscara	DE E. Murphy
	f) Estimate crack growth, if any, for a range of crack types and sizes using bounding loads from task 3.1e in addition to the pressure stresses. Include the effects of TSP movement in these evaluations and any effects from cyclic loads.	12/31/02 (C) ML023610586	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 (C) ML023610586	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 (C) ML023610586	RES J. Muscara	DE E. Murphy

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.1 (continued)	I) Conduct tests of degraded tubes under pressure and with axial and bending loads to validate the analytical results from above tasks.	06/30/03 (T)	RES J. Muscara	DE E. Murphy
	j) Conduct analyses similar to above with refined load estimates if necessary.	06/30/04 (T)	RES J. Muscara	DE E. Murphy
	k) Use information developed in tasks 3.1a through 3.1j to evaluate the conditional probabilities of multiple tube failures for appropriate scenarios in risk assessments for SG tube alternate repair criteria (ARC).	02/28/05 (T)	DSSA S. Long	DE E. Murphy RES J. Muscara H. Woods
3.2	Confirm that damage progression via jet cutting of adjacent tubes is of low enough probability that it can be neglected in accident analyses. (P.s. 10-11) (See Notes 3 and 5) Specific tasks include:			
	a) Complete tests of jet impingement under MSB conditions.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
	b) Conduct long duration tests of jet impingement under severe accident conditions.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
	c) Document results from tasks 3.2a and 3.2b.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
3.3 (MB7216)	When available, use data from the ARTIST program (planned in Switzerland) to develop a better model of the natural mitigation of the radionuclide release that could occur in the secondary side of the SGs. (P.s. 12-13) (See Notes 3 and 5)	09/30/04 (T) See Note 2	RES R. Lee	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (MB7216)	In order to address ACRS criticism of current risk assessments, develop a better understanding of RCS conditions and the corresponding component behavior (including tubes) under severe accident conditions in which the RCS remains pressurized. (P.s. 46-47, 12-15) (See Notes 3 and 5) Specific tasks include:			
	a) Perform system level analyses to assess the impact of plant sequence variations (e.g., pump seal leakage and SG tube leakage).	09/28/01 (C) ML012720004	RES C. Tinkler	DSSA W. Jensen S. Long
	b.1) Re-evaluate existing system level code assumptions and simplifications.	04/12/02 (C)	RES D. Bessett	DSSA W. Jensen S. Long
	b.2) Following the results from 3.4.a and 3.4.b.1, perform additional analysis to: include modeling of heat transfer enhancement from radiation heat transfer in the hot leg and steam generator; suppress unphysical numerically driven flows in the calculations; and investigate the sensitivity of calculated results to bypass flows.	10/31/03 (T) See Note 14	RES D. Bessett	DSSA W. Jensen
	c) Examine 1/7 scale data to assess tube to tube temperature variations and estimate variations for plant scale.	08/31/02 (C)	RES D. Bessett	DSSA W. Jensen S. Long
	d) Perform more rigorous uncertainty analyses with system level code to address inlet plenum mixing by developing distribution functions for mixing parameters based on available data. Peer review.	06/30/03 (T) See Note 13	RES D. Bessett	DSSA W. Jensen S. Long
	e) Examine SG tube severe accident T-H conditions using computational fluid dynamics (CFD) methods. This			

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	e.1) Benchmark CFD methods against 1/7 scale test data.	08/31/01 (C) ML012750061	RES C. Boyd	DSSA W. Jensen S. Long
	e.2) Perform full scale plant calculations (hot leg and SG) for a 4 loop Westinghouse design. Evaluate scale effects.	03/28/02 (C)	RES C. Boyd	DSSA W. Jensen S. Long
	e.3) Perform plant analysis to address the effects on inlet plenum mixing resulting from tube leakage and hot leg orientation (CE design impact).	12/30/02 (C)	RES C. Boyd	DSSA W. Jensen S. Long
	f) Examine the uncertainty in the T-H conditions associated with core melt progression.	10/31/03 (T) See Note 14	RES D. Bessett	DSSA W. Jensen S. Long
	g) Perform experiments to develop data on inlet plenum mixing impacts due to SG tube leakage and hot leg/ inlet plenum configuration.	03/31/03 (C) See Note 15	RES D. Bessett	DSSA W. Jensen S. Long
	h) Perform a systematic examination of the alternate vulnerable locations in the RCS that are subject to failure due to severe accident conditions. This includes the following:			
	h.1) Evaluate the creep failure of primary system passive components such as pressurizer surge line and the hot leg taking into account the material properties of the base metal, welds, and heat affected zones in the presence of residual and applied stresses, in addition to the pressure stress, and the presence of flaws.	11/30/03 (T)	RES J. Muscara	DE E. Murphy DSSA S. Long
	h.2) Evaluate the failure of active components such as PORVs, safety valves, and bolted seals based on operability and “weakest link” considerations for these components.	11/30/03 (T)	RES J. Muscara	DE E. Murphy DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	h.3) Conduct large scale tests if needed.	11/30/05 (T)	RES J. Muscara	DE E. Murphy DSSA S. Long
	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
	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 D. Bessett H. Woods
3.5 (MB7216)	Develop improved methods for assessing the risk associated with SG tubes under accident conditions. (P.s. 47, 16-20) (See Note 5) Specific tasks include: a) Development of an integrated framework for assessing the risk for the high-temperature/high-pressure accident scenarios of interest.	 04/01/02 (C) ML020910624	 RES H. Woods	 DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.5 (continued)	b) Development of improved methods for identifying accident scenarios (including MSB) that lead to challenges on the reactor coolant pressure boundary.	06/28/03 (T)	RES H Woods	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 H. Woods	DSSA S. Long
3.6	To address an ACRS report conclusion that improvements can be made over the current use of a constant probability of detection (POD) for flaws in SG tubes, RES has recently completed an eddy current round robin inspection exercise on a SG mock-up as part of NRC's research to independently evaluate and quantify the inservice inspection reliability for SG tubes. This research has produced results that relate the POD to crack size, voltage, and other flaw severity parameters for stress corrosion cracks at different tube locations using industry qualified teams and procedures. Complete analysis of research results and prepare topical report to document the results. (P.s. 47, 33)	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
3.7 (MB7216)	Assess the need for better leakage correlations as a function of voltage for 7/8" SG tubes. (P.s. 48, 28-29) (See Note 5)	04/30/03 (T)	DE J. Tsao	RES J. Muscara
3.8 (MB0258)	Develop a program to monitor the prediction of flaw growth for systematic deviations from expectations. (Pg. 48) (See Note 5)	01/03/02 (C)	DE J. Tsao	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.9 (MB7216)	Develop a more technically defensible position on the treatment of radio nuclide release to be used in the safety analyses of design basis events. (P.s. 48, 38-44) (See Note 5) Specific tasks include: a) Assess Adams and Atwood and Adams and Sattison spiking data with respect to the ACRS comments. b) Based upon the assessment performed in task 3.9a, develop a response to the ACRS comments. c) Publish in the *Federal Register* for public comment, the response to ACRS' comments. d) Complete review of public comments. e) Based upon task 3.9d, determine if additional work needs to be performed.	 08/09/01 (C) 06/30/03 (T) Note 11 08/31/03 (T) Note 11 12/31/03 (T) Note 11 03/15/04 (T) Note 11	DSSA J. Hayes	
3.10 (MB7216)	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:			

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.10 (continued)	a) Conduct tests to evaluate crack initiation, evolution, and growth. Tests to be conducted under prototypic field conditions with respect to stresses, temperatures and environments. Some tests will be conducted using tubular specimens.	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
3.11	In order to resolve GSI 163, it is necessary to complete the work associated with tasks 3.1 through 3.5 and 3.7 through 3.9. Upon completion of those tasks, develop detailed milestones associated with preparing a GSI resolution document and obtaining the necessary approvals for closing the GSI, including ACRS acceptance of the resolution. (See Note 9)	12/31/05 (T)	DLPM J. Zimmerman	DE E. Murphy DSSA S. Long
3.12	Develop outline and a detailed schedule for completing DG 1073, “Plant Specific Risk-Informed Decision Making: Induced SG Tube Rupture (See Note 9)	12/31/05 (T)	DE E. Murphy	DSSA S. Long

<u>Notes</u>:

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

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

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

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

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

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

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

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

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

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

11. Limited staff resources has necessitated that the focus be placed upon those activities associated with preparation of DSAR for AP 1000, safety evaluations for power uprates, and finalization of draft regulatory guides associated with control room habitability. This re-prioritization of work necessitated delaying the affected SGAP milestones.

12. The NRC received the steam generator license amendment submittal for a lead plant (Catawba) on February 25, 2003, and the generic submittal as a Technical Specification Task Force (TSTF) Traveler on March 14, 2003. The staff is currently performing an initial review of the submittal before a schedule can be established (e.g., to review the quality and level of detail in the submittal).

The staff met with Duke Power on March 27, 2003, and discussed these issues and the results of its acceptance review. Significant progress was made at resolving these issues. As discussed at this meeting, the staff has transmitted a request for additional information (RAI) to Duke Power, and is preparing a second one addressing the TSTF, to summarize the results of its acceptance review. Additionally, the staff is preparing a Commission Paper, to be submitted by 4/8/03, to inform the Commission about the basis for staff's conclusion that there is reasonable assurance of tube integrity, discuss the progress made in revising the regulatory framework, and outline the future plans in this area. This negative consent Commission Paper is proposing to revise the staff's efforts associated with the regulatory framework by canceling the RIS and a separate safety evaluation on NEI 97-06 effort and the GLCP in lieu of the application of the plant specific (lead plant) license amendment and the TSTF processes. Accordingly the milestones 1.14, 1.16, 1.17, and 1.18 are deleted and 1.21 is revised.

13. Problems discovered in the computer code resulted in a change of the scheduled completion date for this item.

14. Several errors and deficiencies were observed in the RELAP/SCDAP code that must be resolved before valid calculations can be performed. The code and input model modifications are in progress, but this has resulted in a delay in the task.

15. This milestone was not performed as evaluation of the cost to perform experiments that would improve upon the Westinghouse experiments showed the cost to be prohibited. CFD analysis provided better information than possible experiments at a very small fraction of the cost. Hence, the objective was satisfied by the completion of milestone 3.4.e.2.

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

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

- Item Nos. 1.1 through 1.21: SG-related issues (not including the DPO-related issues);
- Item Nos. 2.1 through 2.8: Non-SG related issues; and
- Item Nos. 3.1 through 3.11: DPO-related issues.

Historical Background: The NRC originally planned to develop a rule pertaining to steam generator tube integrity. The proposed rule was to implement a more flexible regulatory framework for steam generator surveillance and maintenance activities that allows a degradation specific management approach. The results of the regulatory analysis suggested that the more optimal regulatory approach was to utilize a generic letter. The NRC staff suggested, and the Commission subsequently approved, a revision to the regulatory approach to utilize a generic letter. In SECY-98-248, the staff recommended to the Commission that the proposed GL be put on hold for 3 months while the staff works with NEI on their NEI 97-06 initiative. In the staff requirements memorandum dated December 21, 1998, the Commission did not object to the staff's recommendation. In late 1998 and 1999 the NRC and industry addressed NRC technical and regulatory concerns with the NEI 97-06 initiative, and on February 4, 2000, NEI submitted the generic licensing change package for NRC review. The generic licensing change package included NEI 97-06, Revision 1, proposed generic technical specifications, and a model technical requirements manual section. SECY-00-0078 outlines the staff's proposed review process associated with the revised steam generator tube integrity regulatory framework described in NEI 97-06.

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

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

Current Status:

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

- December 3, 2001	Staff briefing of the Commission on the status of SG action plan.
- December 06, 2001	Staff briefing of ACRS on NEI 97-06.
- June 13, 2002	Public meeting between NRC and NEI on SG issues.
- September 9, 2002	Issuance of a letter to NEI transmitting staff comments on the draft generic license change package.
- September 11, 2002	Public meeting between NRC and NEI on SG issues.
- February 25, 2003	Duke Power submits lead plant (Catawba) SG technical specification amendment application.
- March 14, 2003	NEI submits TSTF-449, Revision 0, SG Program Generic License Change Package.
- March 27, 2002	Public meeting between NRC, Duke Power, and NEI on lead plant submittal.

<u>NRR Technical Contacts</u>: Louise Lund, DE/EMCB, 415-3248
Doug Coe, DIPM/IIPB, 415-2040
Steve Long, DSSA/SPSB, 415-1077

<u>NRR Lead PM</u>: Maitri Banerjee, DLPM, 415-2277

<u>RES Contacts</u>: Joe Muscara, RES, 415-5844
James Davis, 415-6987

EMERGENCY ACTION LEVEL GUIDANCE DEVELOPMENT

TAC No.: MA3695 Revision to NESP-007 Last Update: 03/31/03
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. Document placed on hold pending outcome of spent fuel pool issues.	9/30/01 (C)
13. NEI resubmitted request for endorsement regardless of SPF issues.	10/18/02 (C)
14. Public meeting with NEI to address latest staff comments.	11/21/02 (C)
15. Comments resolved.	02/13/03 (C)
16. DIPM Concurrence (SC, BC, DD, OD)	04/11/03 (T)
17. Obtain Office (NRR/OGC/NMSS/RES) concurrence and or waiver.	04/11/03 (T)
18. Obtain Committee (CRGR/ACRS) concurrence and or waiver.	05/15/03 (T)
19. DIPM Submit to EDO.	05/30/03 (T)
20. ADM publish final Regulatory Guide in FR.	06/16/03 (T)

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

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

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

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

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

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

On June 4, 2001, SECY-01-0100 was sent to the Commission regarding policy issues related to Safeguards, Insurance, and Emergency Preparedness regulations at decommissioning nuclear power plants storing fuel in spent fuel pools. In this document, the staff sought guidance on the appropriate level of emergency preparedness for decommissioning plants. Following the events of September 11, 2001, this paper was recommended for withdrawal on October 25, 2001, and the request was granted on October 30, 2001.

In a memorandum to the Commission on the "Status of Regulatory Exemptions for Decommissioning Plants", dated August 16, 2002, the staff indicated that based on the security measures put into effect since September 11, 2001, together with the time available to take mitigative actions due to the age of the spent fuel, the staff considers the likelihood of an act of radiological sabotage resulting in significant offsite release to be very low. To support future decommissioning regulation, the staff will revise and re-submit a policy options paper on decommissioning regulatory issues, superceding SECY-01-0100, 3 months after Commission direction is received on staff rulemaking recommendations for decommissioning plant safeguards and security.

Based on this projected course, NEI 99-01 should proceed with the planned endorsement. Since being placed on-hold two changes worthy of note have been made in the September 2002 version of Rev. 4.

The first change is an enhancement to the Security EAL for the unusual event class. This EAL has been endorsed by letter from NRR to NEI, dated February 4, 2002, in response to October 6, 2001, Safeguards Advisory addressing a Site-Specific Credible Threat at a Nuclear Power Plant. The second change involves revisions to the "Toxic gas" EALs for the unusual event and alert classes. Due to the nature of these changes they require additional discussion, evaluation, and assessments. In September 2002, NEI submitted a request that NRC endorse NEI-99-01 regardless of issues with EALs for Defueled Stations and Independent Spent Fuel Storage Installations. The review is completed. The document is currently in the concurrence phase, including comment resolutions.

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: NEI has been informed that the EAL changes submitted in the September 2002 package have been reviewed by the staff. The change to the Security EAL is acceptable, however the change to the Toxic Gas EAL is a concern. NEI has been offered the opportunity to revise the Toxic Gas EAL to the previous condition and continue the endorsement process or leave the change as is and further review and comments will be provided. Comment issues have been resolved between the staff and NEI. The document is in the NRC office concurrence process. Concurrence comment resolution has led to a projected schedule completion slip of approximately two weeks.

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

NRR Technical Contacts: T. Blount, DIPM, 415-1501
L. Lois, DSSA, 415-3233

Lead Project Manager: P. Wen, DRIP, 415-2832

DAVIS-BESSE LESSONS LEARNED TASK FORCE
RECOMMENDATIONS REGARDING INSPECTION,
ASSESSMENT, AND PROJECT MANAGEMENT GUIDANCE

TAC No.	Description	
MB7281	Develop Action Plan	Last Update: 03/31/03 (Initial Update) Lead Division: DIPM
MB7726	Evaluation of Inspection and Assessment Guidance	Supporting Division: DLPM Supporting Office: Regions

Milestone	Date (T=Target) (C=Complete)	Lead	Support
Part 1 - Evaluation of Inspection Guidance Related To Problem Identification and Resolution			
The NRC should revise its inspection guidance to provide assessments of: (1) the safety implications of long-standing, unresolved problems; (2) corrective actions phased in over several years or refueling outages; and (3) deferred modifications. [LLTF 3.2.5.(2) High] The NRC should revise the overall PI&R inspection approach such that issues similar to those experienced at DBNPS are reviewed and assessed. The NRC should enhance the guidance for these inspections to prescribe the format of information that is screened when determining which specific problems will be reviewed. [LLTF3.3.2.(2) Low] The NRC should provide enhanced Inspection Manual Chapter guidance to pursue issues and problems identified during plant status reviews [LLTF3.3.2.(3) Low] The NRC should revise its inspection guidance to provide for the longer-term follow-up of issues that have not progressed to a finding. [LLTF3.3.2.(4) Low]			
1. Make changes to IP 71152 to require annual follow-up of three to six issues.	01/02 (C)	DIPM	
2. PI&R focus group assess lessons learned recommendations.	03/03 (C)	DIPM	Regions
3. Develop draft procedure changes based on PI&R group recommendations and provide to regions for review.	04/03 (T)	DIPM	Regions
4. Provide training on procedure changes.	11/03 (T)	DIPM	

Milestone	Date (T=Target) (C=Complete)	Lead	Support
5. Issue procedure changes.	12/03 (T)	DIPM	
PART 2 - Evaluation of IMC 0350 Guidance			
The NRC should develop guidance to address the impacts of IMC 0350 implementation on the regional organizational alignment and resource allocation. [LLTF3.3.5.(4) High]			
1. Assess past and present IMC 0350 data and associated inspection approaches.	04/03 (T)	DIPM	Regions
2. Develop enhanced structure to the inspection approach used for IMC 0350 plants.	08/03 (T)	DIPM	Regions
3. Develop draft revisions to IMC and issue for regional comment.	09/03 (T)	DIPM	
4. Issue procedure revisions.	12/03 (T)	DIPM	
5. Include estimated resources for IMC 0350 plants into budget cycles.	12/03 (T)	DIPM	
Part 3 - Evaluation of Project Management Guidance			
The NRC should establish guidance to ensure that decisions to allow deviations from agency guidelines and recommendations issued in generic communications are adequately documented. [LLTF 3.3.7.(2) High]			
1. The DLPM Handbook will be updated with a new section that addresses documenting staff decisions.	02/03 (C)	DLPM	
2. A training package emphasizing compliance with the requirements of MD 3.53 will be developed and distributed to all Offices and regions.	04/03 (T)	DLPM	

Description: The Davis Besse Lessons Learned Task Force (LLTF) identified several issues concerning the NRC's oversight, inspection, and project management guidance. The LLTF recommended that changes be made to the NRC's inspection program to ensure that sufficient inspections are conducted of long-standing unresolved problems, that guidance be developed to assess the impacts of Inspection Manual Chapter 0350 on regional resource allocations, and that guidance be developed to ensure that decisions to allow deviations from agency guidelines in generic communications are adequately documented.

Historical Background: The Davis Besse LLTF conducted an independent evaluation of the NRC's regulatory processes related to assuring reactor vessel head integrity in order to identify and recommend areas of improvement applicable to the NRC and the industry. A report summarizing their findings and recommendations was published on September 30, 2002. The report contains several consolidated lists of recommendations. The LLTF report was reviewed by a Review Team (RT), consisting of several senior management personnel appointed by the EDO. The RT issued a report on November 26, 2002, endorsing all but two of the LLTF recommendations, and placing them into four overarching groups. On January 3, 2003, the EDO issued a memo to the Director, NRR, and the Director, RES, tasking them with a plan for accomplishing the recommendations. This action plan addresses the Group 3 recommendations of the Davis-Besse Lessons Learned Task Force regarding inspection, assessment, and project management guidance. As directed by the EDO's memo, this action plan includes the 3 high priority recommendations in the "Evaluation of Inspection, Assessment, and Project Management Guidance" grouping. In addition, three low priority recommendations are included since they are closely related to the high priority recommendations and will be accomplished in conjunction with the work necessary to resolve the high priority items. The LLTF recommendations are also listed in the attached Table 1.

Proposed Actions: Parts 1, 2, and 3 of this action plan are unrelated and will be worked as three independent efforts. The recommendations associated with the inspection program will be reviewed by the Problem Identification and Resolution (PI&R) focus group which is made up of headquarters and regional representatives. The focus group will assess whether changes to the current PI&R inspection approach are warranted. Procedure changes will then be made as appropriate, and inspector training will be conducted.

The recommendation associated with IMC 0350 will be assessed by evaluating the previous inspection approaches used and associated resource expenditures for plants that entered the IMC 0350 process. The staff will then attempt to better define a more enhanced inspection framework for a plant that enters IMC 0350. Once this additional inspection guidance is completed, a better estimate of resources will be made, and resources for IMC 0350 will be included in budget projections.

Project management guidance regarding documentation when accepting deviations from generic communications recommendations will be incorporated into the DLPM handbook and into training materials to be distributed to all Offices and Regions.

Originating Documents:

Memorandum from Travers, W.D. to Collins, S. and Thadani, A. C., dated January 3, 2003, "Actions Resulting From The Davis-Besse Lessons Learned Task Force Report Recommendations." (ML023640431)

Memorandum from Paperiello, C.J. to Travers, W.D., dated November 26, 2002, "Senior Management Review of the Lessons-Learned Report of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head." (ML023260433)

Memorandum from Howell, A.T. to Kane, W.F., dated September 30, 2002, "Degradation of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head Lessons-Learned Report." (ML022740211)

Regulatory Assessment: It is not anticipated that this action plan will result in any additional regulatory requirements on licensees. The plan focuses on what enhancements should be made to existing inspection and project management guidance to ensure better scope, efficiency, and documentation of such activities.

Current Status: This is the initial update for this Action Plan, which addresses the Group 3 recommendations of the Davis-Besse Lessons Learned Task Force Review Team regarding inspection, assessment, and project management guidance.

Contacts:

NRR Lead for this action plan:	Jeffrey Jacobson, DIPM, 415-2977
Overall Lead for DB LLTF response:	Brendan Moroney, DLPM, 415-3974

References:

Inspection Manual 0350, “Oversight of Operating Reactor Facilities in an Extended Shutdown as a Result of Significant Performance Problems.”

Table 1
LLTF Report Recommendations Included in This Action Plan

RECOMMENDATION NUMBER	RECOMMENDATION	PRIORITY
3.2.5.(2)	The NRC should revise its inspection guidance to provide assessments of: (1) the safety implications of long-standing, unresolved problems; (2) corrective actions phased in over several years or refueling outages; and (3) deferred modifications.	High
3.3.2.(2)	The NRC should revise the overall PI&R inspection approach such that issues similar to those experienced at DBNPS are reviewed and assessed. The NRC should enhance the guidance for these inspections to prescribe the format of information that is screened when determining which specific problems will be reviewed.	Low
3.3.2.(3)	The NRC should provide enhanced Inspection Manual Chapter guidance to pursue issues and problems identified during plant status reviews. [3.3.2.(3)]	Low
3.3.2.(4)	The NRC should revise its inspection guidance to provide for the longer-term follow-up of issues that have not progressed to a finding.	Low
3.3.5.(4)	The NRC should develop guidance to address the impacts of IMC 0350 implementation on the regional organizational alignment and resource allocation.	High
3.3.7.(2)	The NRC should establish guidance to ensure that decisions to allow deviations from agency guidelines and recommendations issued in generic communications are adequately documented.	High

DAVIS-BESSE LESSONS LEARNED TASK FORCE RECOMMENDATIONS REGARDING STRESS CORROSION CRACKING

TAC No.	Description
MB2916	Non plant-specific activities for Bulletin 2001-01
MB3567	VHP Action Plan (Coordination and Administration)
MB3954	Development of CRDM NUREGs (Bulletin 2001-01)
MB4495	Lead PM Activities for Bulletin 2002-01
MB4603	Non plant-specific activities for Bulletin 2002-01
MB5465	Lead PM Activities for Bulletin 2002-02
MB6218	Inspection TI for Bulletin 2002-02
MB6220	Review of NEI/MRP Crack Growth Rate Report (MRP-55)
MB6221	Development of Alternate (to ASME Code) RPV Head and VHP Inspection Requirements
MB6222	Review of NEI/MRP RPV Head and VHP Inspection Plan (MRP-75)
MB6584	RIS: Status of Degradation of RPV Head Penetrations and BACC Programs
MB7182	Orders for Interim Inspection Guidelines

Last Update: 03/31/03 (Initial Update)
Lead Division: DLPM
Supporting Divisions: DE, DSSA, DIPM, & DRIP
Supporting Offices: RES & Regions

Milestone	Date (T=Target) (C=Complete)	Lead	Support
Part I - Reactor Pressure Vessel Head Inspection Requirements			
1. Collect and summarize information available worldwide on Alloy 600, Alloy 690 and other nickel based alloy nozzle cracking for use in evaluation of revised inspection requirements. [LLTF 3.1.1(1)-High]	03/04 (T)	RES/DET	DE
2. Critically evaluate existing SCC models with respect to their continuing use in the susceptibility index. [LLTF 3.1.4(1)-Medium]	05/03 (T)	RES/DET	DE
3. Complete initial evaluation of individual plant inspections in response to Bulletins and Orders.	05/04 (T) (Staff will continue to review future inspection results).	DE	DLPM Regions

Milestone	Date (T=Target) (C=Complete)	Lead	Support
4. Monitor and provide input to industry efforts to develop revised RPV Head inspection requirements (ASME Code Section XI). [LLTF 3.3.4(8)-High LLTF 3.3.7(6)-Low]	Note (1)	DE	RES/DET DSSA Regions Industry
5. Participate in meetings and establish communications with appropriate stakeholders (e.g., MRP, ASME). [LLTF 3.3.4(8)-High]	Ongoing	DE	RES/DET DLPM DRIP DSSA industry
6. Make decision to endorse revised ASME Code requirements, when issued, or implement alternative requirements. [LLTF 3.3.4(8)-High]	Note (1)	DE	RES/DET
7. If alternative, determine appropriate regulatory tool and establish schedule for implementation.	Note (1)	DE	DRIP DIPM DSSA RES/DET industry public
Part II - Boric Acid Corrosion Control			
1. Collect and summarize information available worldwide on boric acid corrosion of pressure boundary materials for use in evaluation of revised inspection requirements. [LLTF 3.1.1(1)-High]	10/04 (T)	RES/DET	DE
2. Evaluate individual plant responses to Bulletin 2002-01 regarding Boric Acid Inspection Programs (60-day responses and necessary follow-up) and summarize plant responses on BACC programs in an appropriate public document.	04/03 (T)	DE	DLPM
3. Participate in meetings and establish communications with appropriate stakeholders (e.g.,MRP, ASME).	Ongoing	DE	RES/DET DLPM DRIP DSSA industry
4. Evaluate need to take additional regulatory actions and determine appropriate regulatory tool(s).	04/03 (T)	DE	DLPM DRIP DIPM DSSA Regions

Milestone	Date (T=Target) (C=Complete)	Lead	Support
5. Develop milestones for additional regulatory actions, as necessary.	05/03 (T)	DE	DLPM DSSA
6. Review and evaluate the adequacy of revised ASME Code Requirements for Pressure Testing/Leakage Evaluation being developed by the ASME Code, Section XI, Task Group on Boric Acid Corrosion.	01/05 (T)	DE	RES/DET
Part III - Inspection Programs			
1. Develop inspection guidance or revise existing guidance to ensure that VHP nozzles and the RPV head area are periodically reviewed by the NRC during licensee ISI activities. [LLTF 3.3.4(3)-High]	03/04 (T)	DIPM	DE Regions
2. Develop inspection guidance that provides for timely, periodic inspection of PWR plant BACC programs. [LLTF3.3.2(1)-High]	03/04 (T)	DIPM	DE Regions
3. Develop inspection guidance for assessing the adequacy of PWR plant BACC programs (implementation effectiveness, ability to identify leakage, adequacy of evaluation of leaks). [LLTF 3.2.2(1)-High]	03/04 (T)	DIPM	DE RES/DET Regions

Notes: (1) Milestone dates will be set when a target date for issuing revised Code requirements is established. However, staff may initiate action to establish alternative inspection requirements, if appropriate, prior to completion of industry activities.

Description: The reactor vessel head (RVH) degradation found at Davis-Besse, along with other documented incidences of circumferential cracking of vessel head penetration (VHP) nozzles, have prompted the staff to question the adequacy of current RVH and VHP inspection programs that rely on visual examinations as the primary inspection method. Also, the failure to adequately address indications of boric acid leakage at Davis-Besse raised questions as to the efficacy of industry boric acid corrosion control (BACC) programs. Finally, review of the Davis-Besse event identified deficiencies in the NRC inspection programs.

Historical Background: In March 2002, while conducting inspections in response to Bulletin 2001-01, the Davis-Besse Nuclear Power Station identified three CRDM nozzles with indications of axial cracking, which were through-wall, and resulted in reactor coolant pressure boundary leakage. During the nozzle repair activities, the licensee removed boric acid deposits from the RVH, and conducted a visual examination of the area, which identified a 7 inch by 4-to-5 inch cavity on the downhill side of nozzle 3, down to the stainless steel cladding. The extent of the damage indicated that it occurred over an extended period and that the licensee’s programs to inspect the RPV head and to identify and correct boric acid leakage were ineffective.

One of the NRC follow-up actions to the Davis-Besse event was formation of a Lessons Learned Task Force (LLTF). The LLTF conducted an independent evaluation of the NRC's regulatory processes related to assuring reactor vessel head integrity in order to identify and recommend areas of improvement applicable to the NRC and the industry. A report summarizing their findings and recommendations was published on September 30, 2002. The report contains several consolidated lists of recommendations. The LLTF report was reviewed by a Review Team (RT), consisting of several senior management personnel appointed by the Executive Director for Operations (EDO). The RT issued a report on November 26, 2002, endorsing all but two of the LLTF recommendations, and placing them into four overarching groups. On January 3, 2003, the EDO issued a memo to the Director, NRR, and the Director, RES, tasking them with developing a plan for accomplishing the recommendations. This action plan addresses the recommendations in the "Assessment of Stress Corrosion Cracking" grouping of the RT report. The LLTF recommendations are listed in the attached Table 1, and have been identified under the appropriate milestone(s).

Proposed Actions: The staff is interacting with all PWR licensees, the American Society of Mechanical Engineers (ASME), the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP), and other external stakeholders in addressing the issues discussed above. This action plan includes milestones aimed at guiding the NRC and industry to effectively manage RVH degradation and BACC. Throughout the implementation of this action plan, the NRC will establish the necessary communications mechanisms to ensure that the NRC, the industry, and all stakeholders are informed and sharing the same information. This will be accomplished through public meetings, technical working groups, ACRS briefings, and web site postings, as appropriate.

The Part I milestones deal with development of improved inspection requirements for the RPV head and VHP nozzles. Interim inspection guidelines (TI-150) have been issued for use by NRC inspectors and are being updated as needed based on inspection results. The first effort in development of new regulatory requirements is for the staff to establish the technical basis for new inspection requirements through ongoing and planned research programs. This will include collecting and evaluating information on VHP nozzle inspection results and evaluating current methodologies for determining leakage probability, nondestructive testing, crack susceptibility, crack growth propagation, and failure margins. In parallel with these activities, the staff will monitor and assess the adequacy of revisions to the ASME Boiler and Pressure Vessel Code, which will be based on the inspection program developed by the EPRI MRP. If the revised ASME Code requirements are acceptable, based on the staff's technical evaluations, the NRC will initiate action to endorse them in a revision to 10 CFR 50.55a. If the revised ASME Code requirements cannot be made acceptable to the NRC, then alternate requirements would have to be developed and implemented by the revision to 10 CFR 50.55a. The staff may initiate action to establish alternative inspection requirements, if appropriate, prior to completion of industry activities.

The Part II milestones evaluate whether industry BACC programs are meeting NRC expectations and whether additional inspection guidance should be issued. First, the staff will establish a technical basis for BACC program requirements through ongoing and planned research programs. This will include evaluation of boric acid corrosion events in past reports and in responses to Bulletin 2002-01, and studies of corrosion rates of reactor pressure boundary materials in boric acid solutions. The staff is also monitoring development of revised ASME Code requirements by the Section XI Task Group on Boric Acid Corrosion. If the staff determines that additional interim guidelines are needed prior to issuance of the revised Code requirements, they will be issued by an appropriate regulatory tool. When the ASME Code requirements are revised, the NRC will initiate action to endorse them, if acceptable. If the revised ASME code requirements cannot be made acceptable to the NRC, then alternate requirements would have to be developed and implemented by an appropriate regulatory tool.

The Part III milestones address the LLTF findings that the NRC inspection guidelines did not provide effective oversight of licensee RPV head inspection and BACC programs. Revised guidelines for these activities will be developed. Throughout the process of establishing new requirements, existing NRC inspection procedures would be evaluated to verify whether they adequately address the revised requirements, and would be updated as needed.

Originating Documents:

Memorandum from Travers, W.D. to Collins, S. and Thadani, A. C., dated January 3, 2003, "Actions Resulting From The Davis-Besse Lessons Learned Task Force Report Recommendations." (ADAMS Accession No. ML023640431)

Memorandum from Paperiello, C.J. to Travers, W.D., dated November 26, 2002, "Senior Management Review of the Lessons-Learned Report of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head." (ADAMS Accession No. ML023260433)

Memorandum from Howell, A.T. to Kane, W.F., dated September 30, 2002, "Degradation of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head Lessons-Learned Report." (ADAMS Accession No. ML022740211)

Regulatory Assessment: The current method for managing PWSCC in the VHP nozzles of U.S. PWRs is dependent on the implementation of inspection methods intended to provide early detection of degradation of the reactor coolant pressure boundary. Title 10, Section 50.55a(g)(4) of the *Code of Federal Regulations* requires, in part, that ASME Code Class 1, 2, and 3 components must meet the inservice inspection requirements of Section XI of the ASME Boiler and Pressure Vessel Code throughout the service life of a boiling or pressurized water reactor. Pursuant to Inspection Category B-P of Table IWB-2500-1 to Section XI of the ASME Boiler and Pressure Vessel Code, licensees are required to perform VT-2 visual examinations of their vessel head penetration nozzles and reactor vessel heads once every refueling outage for the system leak tests, and once an inspection interval for the hydrostatic pressure test.

Based on the experience with the VHP nozzle cracking phenomenon, the VT-2 visual examination methods required by the ASME Code for inspections of VHP nozzles do not provide reasonable assurance that leakage from a through-wall flaw in a nozzle will be detected. The VT-2 visual examination methods specified by the ASME Code are not directed at detecting the very small amounts of boric acid deposits, e.g., on the order of a few grams, that have been associated with VHP nozzle leaks in operating plants. In addition, the location of thermal insulating materials and physical obstructions may prevent the VT-2 visual examination methods from identifying minute amounts of boric acid deposits on the outer surface of the vessel head. Specifically, Paragraph IWA-5242 of Section XI of the ASME Boiler and Pressure Vessel Code does not require licensees to remove thermal insulation materials when performing ASME VT-2 visual examinations of reactor vessel heads. Cleanliness of reactor vessel heads during the examinations, which is critical for visual examination methods to be capable of distinguishing between boric acid residues that result from VHP nozzle leaks and those residues that result from leaks in other reactor coolant system components, is not addressed by the ASME Code.

Based on knowledge obtained from evaluation of the Davis-Besse event, and information provided from PWR licensees in response to Bulletins 2001-01, 2002-01 and 2002-02, the NRC issued an Order to all PWR plants establishing enhanced inspection requirements on an interim basis, which will provide adequate assurance of safe plant operation until permanent requirements are established and promulgated.

Current Status: This is the initial update for this Action Plan, which addresses the Group 1 recommendations of the Davis-Besse Lessons Learned Task Force Review Team regarding Stress Corrosion Cracking.

Contacts:

NRR Lead PM:	Brendan Moroney, DLPM, 415-3974
NRR Technical Contacts:	Allen Hiser, EMCB, 415-1034
	Edmund Sullivan, EMCB, 415-2796
RES Technical Contact:	William Cullen, DET/MEB, 415-6754
NRR/DIPM Lead Contact:	Jeffrey Jacobson, IIPB, 415-2977
NRR/DRIP Lead Contact:	Terrence Reis, RORP, 415-3281

References:

Orders establishing interim inspection requirements for reactor pressure vessel heads at pressurized water reactors, February 11, 2003.

NRC Bulletin 2002-02, “Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs,” August 9, 2002.

NRC Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” March 18, 2002.

Information Notice 2002-11, “Recent Experience With Degradation of Reactor Pressure Vessel Head,” March 12, 2002.

NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” August 3, 2001.

Information Notice 2001-05, “Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3,” April 30, 2001.

Generic Letter 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,” April 1, 1997.

Information Notice 96-11, “Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations,” February 14, 1996.

NUREG/CR-6245, ”Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking,” October 1994.

Letter from Russell, W. T., (USNRC) to Rasin, W., (Nuclear Management and Resources Council), dated November 19, 1993, “Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking.”

Information Notice 90-10, “Primary Water Stress Corrosion Cracking of INCONEL 600,” February 23, 1990.

Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” March 17, 1988.

Table 1
LLTF Report Recommendations Included in SCC Action Plan

High Priority

NUMBER	RECOMMENDATION
3.1.1(1)	The NRC should assemble foreign and domestic information concerning Alloy 600 (and other nickel based alloys) nozzle cracking and boric acid corrosion from technical studies, previous related generic communications, industry guidance, and operational events. Following an analysis of nickel based alloy nozzle susceptibility to stress corrosion cracking (SCC), including other susceptible components, and boric acid corrosion of carbon steel, the NRC should propose a course of action and an implementation schedule to address the results.
3.2.2(1)	The NRC should inspect the adequacy of PWR plant boric acid corrosion control programs, including their implementation effectiveness, to determine their acceptability for the identification of boric acid leakage, and their acceptability to ensure that adequate evaluations are performed for identified boric acid leaks.
3.3.2(1)	The NRC should develop inspection guidance for the periodic inspection of PWR plant boric acid corrosion control programs.
3.3.4(3)	The NRC should strengthen its inspection guidance or revise existing guidance, such as IP 71111.08, to ensure that VHP nozzles and the RPV head area are periodically reviewed by the NRC during licensee ISI activities. Such NRC inspections could be accomplished by direct observation, remote video observation, or by the review of videotapes. General guidance pertaining to boric acid corrosion observations should be included in IP 7111.08
3.3.4(8)	The NRC should encourage ASME Code requirement changes for bare metal inspections of nickel based alloy nozzles for which the code does not require the removal of insulation for inspections. The NRC should also encourage ASME Code requirement changes for the conduct of non-visual NDE inspections of VHP nozzles. Alternatively, the NRC should revise 10 CFR 50.55a to address these areas.

Medium Priority

NUMBER	RECOMMENDATION
3.1.4(1)	The NRC should determine if it is appropriate to continue using the existing SCC models as predictors of VHP nozzle PWSCC susceptibility given the apparent large uncertainties associated with the models. The NRC should determine whether additional analysis and testing are needed to reduce uncertainties in these models relative to their continued application in regulatory decision making.

Low Priority

NUMBER	RECOMMENDATION
3.3.7(6)	Determine whether ISI summary reports should be submitted to the NRC, and revise the ASME submission requirement and staff guidance regarding disposition of the reports, as appropriate.

ECCS SUCTION BLOCKAGE

TAC Nos. MA6454, MA2452, MA4014, MA0704, M95473 MA6204, MA0698, MB4047, MB6411, MB3103, MB8052, MB7776, and MB4864

Last Update: 03/26/03
Lead NRR Division: DSSA
Supporting Divisions: DE, DRIP, and DET (RES)
GSI: 191

MILESTONES	DATE (T/C)
PART I: **BWR ECCS SUCTION STRAINER CLOGGING ISSUE**	
1. NRCB 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors"	10/01 (C)
PART II: **NPSH EVALUATIONS**	
1. GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps" ○ Complete review of licensee responses ○ Complete revision of Draft RG 1.1/RG 1.82 (DG-1107)	 03/00 (C) 09/03 (T)
PART III: **CONTAINMENT COATINGS**	
1. GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"	07/00 (C)
2. NRC-sponsored research program on the potential for coatings to fail during an accident	03/01 (C)
PART IV: **GSI 191, "ASSESSMENT OF DEBRIS ACCUMULATION ON PRESSURIZED WATER REACTOR (PWR) SUMP PERFORMANCE"**	
1. NRC-sponsored research program on the potential for loss of ECCS NPSH during a LOCA due to clogging by debris ○ Preliminary (qualitative) risk assessment (NRR) ○ Complete collection of plant data to support research program ○ Integrate industry activities into this Action Plan ○ Complete research program on PWR sump blockage ○ Evaluate need for regulatory action based on research program results (NRR)	 03/99 (C) 06/99 (C) 04/00 (C) 09/01 (C) 03/02 (C)

MILESTONES	DATE (T/C)
2. Resolve ECCS suction clogging issue for PWRs (Regulation/Guidance Development and Issuance Stages of GSI process in MD 6.4 (Stages 4 and 5))	
○ Update ECCS Suction Clogging Action Plan to include resolution of the issue for PWRs	01/02 (C)
○ Brief NRR ET to obtain approval to prepare a generic letter (GL)	02/02 (C)
○ Public meeting with NEI, WOG, B&WOG, CEOG	03/02 (C)
○ ACRS Briefing on proposed draft GL	02/03 (C)
○ CRGR Briefing on proposed Bulletin addressing compliance/degraded condition	04/03 (T)
○ Information Paper to Commission, Issue Bulletin	05/03 (T)
○ CRGR Briefing on proposed draft GL	05/03 (T)
○ Proposed draft GL issued for Public Comment	06/03 (T)
○ Public meeting with NEI, WOG, B&WOG during Public Comment period	08/03(T)
○ Public Comment period ends	
○ Resolution of Public Comments and revisions to proposed GL made, as necessary	08/03 (T) 09/03 (T)
○ CRGR Briefing on proposed final GL	
○ Information Paper sent to Commission, issue GL	10/03 (T)
○ NEI publish PWR Industry Evaluation Guidelines	11/03 (T)
○ NRC starts Reviews of GL Responses and Selective Audits	09/03 (T) 03/04 (T)

Description: This action plan was originally prepared to comprehensively address the adequacy of ECCS suction design, and to ensure adequate ECCS pump net positive suction head (NPSH) during a loss-of-coolant accident (LOCA). Specifically, the concern is whether debris could clog ECCS suction strainers or sump screens during an accident and prevent the ECCS from performing its safety function. The plan is risk informed.

This plan has four parts. First, for boiling-water reactors (BWRs), this issue has been addressed by licensee responses to NRCB 96-03. At the time this action plan was developed, the staff was confirming the adequacy of the licensee solutions implemented in response to the bulletin; therefore, the staff's confirmatory effort included in this action plan for completeness. The staff's activities related to NRCB 96-03 are complete. Second, the adequacy of licensee (both PWR and BWR) net positive suction head (NPSH) calculations was evaluated through NRR review of licensee responses to GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," dated October 7, 1997. The staff's activities related to GL 97-04 are complete. The third part of the plan consists of two efforts by the staff. The first effort assessed the adequacy of the implementation and maintenance of current licensee coating programs through NRR review of licensee responses to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," dated July 14, 1998. The second effort is a research program to assess the potential for coatings to become debris, including the timing of any failures that might occur, and the cause and the characteristics of the debris. These two efforts combined provided NRR the necessary technical bases on which to assess the potential threat to the ECCS by coating debris and the adequacy of coating licensing bases (both PWR and BWR). The staff's activities related to GL 98-04 and the coatings research program are complete. The results of these two programs also feed into the fourth part of the action plan: an evaluation of the potential for clogging of PWR ECCS recirculation sumps during a LOCA. RES completed its assessment of the potential for debris clogging of PWR ECCS sumps during a LOCA. The study supports the resolution of GSI -191, "Assessment of Debris Accumulation on PWR Sump Performance." RES performed a parametric evaluation to demonstrate whether sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation was ill suited for determining whether sump blockage will impede or prevent long-term recirculation at a specific plant. By memorandum dated September 28, 2001, RES transferred the lead for GSI-191 to NRR .

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 GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," dated December 3, 1985. This generic letter documented the NRC's resolution of USI A-43. The staff concluded at that time that no new requirements would be imposed on licensees; however, the staff did recommend that Regulatory Guide 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," be used as guidance for the conduct of 10 CFR 50.59 reviews dealing with change out and/or modification of thermal insulation installed on primary coolant system piping and components. NUREG-0897, Revision 1, "Containment Emergency Sump Performance" (October 1985), contained technical findings related to USI A-43, and was the principal reference for developing the revised regulatory guide.

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

Similarly, on January 16 and April 14, 1993, two events involving the clogging of ECCS strainers occurred at the Perry Nuclear Power Plant, a domestic BWR. In the first Perry event, the suction strainers for the residual heat removal 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 considered.

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

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

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

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

Proposed Actions: This action plan was initially divided into four parallel efforts. Three of these efforts are complete. The action plan has been updated to provide additional NRR actions necessary to respond to RES findings related to GSI-191. The first effort was for the staff to complete its review of the resolution of NRCB 96-03. Most licensees installed their new strainers under 10 CFR 50.59, concluding that installing the new strainer modification did not constitute an unreviewed safety question. Since the staff did not receive detailed responses from these licensees describing their resolutions, the staff audited four plants to determine if any significant issues exist. No significant safety issues were identified. The issue was closed based on the audit findings and the findings of the staff's review of coatings related issues (discussed below). The staff summarized the review results in a memorandum from R. Elliott to G. Holahan, "Completion of Staff Reviews of NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-water Reactors," and NRC Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode'" dated October 18, 2001.

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

The third effort involved the evaluation of coatings as a potential debris source. Concerns raised in this area were due to events where qualified coatings have failed during normal operation at a number of sites. The failure of qualified coatings during normal operation led to two specific staff concerns. The first concern is whether the qualification of coatings is adequate to ensure that coatings do not pose a potential threat to the ECCS. Accordingly, the staff, led by RES, evaluated the potential for coatings to become debris during an accident and consequently, become a threat to the ECCS performing its safety function. This research program is complete and the findings are discussed below under "Current Status." The second concern relates to the adequacy of licensee programs to apply and maintain coatings consistent with their licensing bases. This concern was addressed by NRR staff through review of license responses to GL 98-04. The staff has completed its review of licensee responses to GL 98-04 to determine if licensee coating programs (application and maintenance of protective coatings in containment) are adequate to meet their current licensing bases. The staff review of the responses to GL 98-04 is complete and identified no significant issues. This issue is applicable to BWRs and PWRs.

The fourth effort involves an evaluation of PWR sumps based on new information learned during the development of the staff's resolution for NRCB 96-03. RES conducted a program to evaluate PWR sump designs and their susceptibility to blockage by debris. This evaluation included a risk assessment. Risk insights support the conclusions drawn relative to the need for licensees to address the potential for ECCS suction clogging. RES's PWR sump study is complete. RES parametrically evaluated whether sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs. As noted above, this action plan has been updated to include NRR actions necessary to address RES's findings.

The research program needed plant data to bound the problem to be evaluated. The Nuclear Energy Institute (NEI) conducted a survey of PWR licensees and provided the information needed by RES. The staff is coordinating 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 larger strainers by BWR licensees.

The staff believes that there is sufficient new information and concerns raised relative to the potential for debris clogging in PWRs that this action plan has been updated to address PWR sump blockage concerns. As noted above, RES's parametric evaluation demonstrated that sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill suited for making a determination that sump blockage will impede or prevent long-term recirculation at a specific plant. Therefore, it is not clear how significant a threat to PWR ECCS operation exists. The staff considers continued operation of PWRs during the implementation of this action plan to be acceptable because the probability of the initiating event (i.e., large break LOCA) is extremely low. More probable (although still low probability) LOCAs (small, intermediate) will generate smaller quantities of debris, require less ECCS flow, take more time to use up the water inventory in the refueling water storage tank (RWST), and in some cases may not even require the use of recirculation from the ECCS sump because the flow through the break would be small enough that the operator will have sufficient time to safely shut the plant down. In addition, all PWRs have received approval by the staff for leak-before-break (LBB) credit on their largest RCS primary coolant piping. While LBB is not acceptable for demonstrating compliance with 10 CFR 50.46, it does demonstrate that LBB-qualified piping is of sufficient toughness that it will most likely leak (even under safe shutdown earthquake conditions) rather than rupture. This, in turn, would allow operators adequate opportunity to shut the plant down safely (although debris generation and transport for an LBB size through-wall flaw will still need to be considered). Additionally, the staff notes that there are sources of margin in PWR designs which may not be credited in the licensing basis for each plant. For instance, NPSH analyses for most PWRs do not credit containment overpressure (which would likely be present during a LOCA). Any containment pressure greater than assumed in the NPSH analysis provides additional margin for ECCS operability during an accident. Another example of margin would be that it has been shown, in many cases, that ECCS pumps would be able to continue operating for some period of time under cavitation conditions. Some licensees have vendor data demonstrating this. Design margins such as these examples may prevent complete loss of ECCS recirculation flow or increase the time available for operator action (e.g., refilling the RWST) prior to loss of flow. And finally, the staff believes that continued operation of PWRs is also acceptable because of PWR design features which may minimize potential blockage of the ECCS sumps during a LOCA. The RES study on sump blockage attempted to capture many of the PWR design features parametrically, however, it is not possible for a generic study of this nature to capture all the variations in plant-specific features that could affect the potential for ECCS sump blockage (e.g., piping layouts, insulation location within containment, etc.). Therefore, evaluation on a plant-specific basis is necessary to determine the potential for ECCS sump clogging in each plant.

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

As part of the GSI-191 study, RES's contractor, Los Alamos National Laboratory (LANL), performed a generic risk assessment to determine how much core damage frequency (CDF) is changed by the findings of the parametric analysis. Utilizing initiating event frequencies that consider LBB credit consistent with NUREG/CR-5750, LANL calculated an overall CDF of 3.3E-06 when debris clogging as a failure mechanism is not considered, and an overall CDF of 1.5E-04 when debris clogging is considered. However, these CDFs were calculated without giving any credit for operator action, and without consideration to whether the ECCS or containment spray pumps would be able to continue operating after the headloss across the sump screen exceeds the calculated licensing basis NPSH margin. The change in CDF is also dominated by the small and very small break LOCAs which are events where there are significant operator actions that can be taken to prevent core damage. The risk benefit of certain interim compensatory measures is demonstrated by the NRC-sponsored technical report LA-UR-02-7562, "The Impact of Recovery from Debris-Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," dated February 2003. On this basis, the schedule for issuing a generic communication to address the PWR sump clogging issue outlined above is considered to be appropriate.

These conclusions clearly support this action plan as outlined herein.

Current Status: The review of NRCB 96-03 responses is complete.

NRR review of GL 97-04 responses is complete.

The review of Generic Letter (GL) 98-04 responses is complete. No significant issues were identified in the review. In addition, RES completed its coating research program and incorporated the results of this program into the PWR sump study. Available evidence from limited industry tests of the transport of coating debris indicates that coating debris (chips) may not transport very well under conditions approximating those of containment sump flow. In fact, only very small amounts of debris actually reached the screens in these tests.

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

RES's PWR sump study is complete. To date, the industry has monitored the NRC's activities in this area rather than conduct any testing or research.

RES presented the results of the GSI-191 parametric evaluation to the ACRS on July 12 and September 5, 2001. Also, a public meeting between the NRC, the Nuclear Energy Institute, and the PWR Owners' Groups was held on July 26 and 27, 2001, to discuss the parametric evaluation with interested stakeholders. RES published the Los Alamos National Laboratory report entitled, "GSI-191: Parametric Evaluation for Pressurized Water Reactor Recirculation Sump Performance," as NUREG/CR-6762 in August 2002. The staff continues to hold regular public meetings with the PWR owners groups and NEI on the progress toward resolving GSI-191.

The PWR Industry has commenced a two-step program to assess the current conditions and evaluate sump recirculation performance. The first guidance document, NEI 02-01, "Condition Assessment Guidelines: Debris Sources inside Containment," was published in September 2002. In September 2003, NEI plans to publish the second guidance document, which will recommend methodologies for evaluating a PWR's susceptibility to sump clogging based upon the information collected in accordance with NEI 02-01. The NRC staff is monitoring the development of NEI's sump evaluation guidance program. Consistent with the risk significance of the PWR sump-clogging concern, the staff is preparing a bulletin that will request information on compliance within 60 days and information on interim compensatory measures if noncompliant. The staff is also preparing a Generic Letter that will request that licensees evaluate the ECCS recirculation performance and take appropriate corrective actions depending on the results of the evaluation.

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 Reviewers:	Ralph Architzel, SPLB, 415-2804 John Lehning, SPLB, 415-3285
NRR Technical Contacts:	Rich Lobel, SPLB, 415-2865 Nicholas Saltos, SPSB, 415-1072
RES Technical Contact:	B. P. Jain, ERAB, 415-6778

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.

NUREG/CR-6762, "GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2002.

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

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

NEI 02-01, “Condition Assessment Guidelines: Debris Sources inside Containment,” Revision 1 published in September 2002.

Technical Letter Report LA-UR-02-7562, “The Impact of Recovery from Debris-Induced Loss of ECCS Recirculation on PWR Core Damage Frequency,” dated February 2003.

CONTROL ROOM HABITABILITY

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

Last Update: 04/04/03
Lead NRR Division: DSSA
Supporting Division: TBD

MILESTONES		DATE (T/C)
1.	Staff review of NEI 99-03 and redline and strikeout version provided to NEI Control Room Habitability task force	04/17/01 (C)
2.	Staff prepare Generic Letter and develop draft Regulatory Guides on Control Room Habitability at Nuclear Power Reactors (DG-1114), Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (DG-1115), Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light Water Nuclear Power Reactors (DG-1113), and Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (DG-1111)	07/01/01 (C)
3.	Office review of draft Regulatory Guides DG-1111 and DG-1113	12/31/01 (C)
4.	Office review of draft Regulatory Guides DG-1114 and DG-1115 and draft Generic Letter	03/01/02 (C)
5.	Brief CRGR on draft Regulatory Guides DG-1111 and DG-1113	12/31/01 (C)
6.	Brief CRGR on draft Regulatory Guides DG-1114 and DG-1115 and draft Generic Letter	draft GL: 04/29/02 (C) DG-1114, DG-1115: 03/11/02 (C)
7.	Issue draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115 and draft Generic Letter for public comment	draft GL: 05/09/02 (C) DG-1111: 12/31/01 (C) DG-1113: 01/31/02 (C) DG-1114: 03/28/02 (C) DG-1115: 03/28/02 (C)
8.	Public meeting on draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115 and draft Generic Letter	RI: 07/11/02 (C) RII: 07/16/02 (C) RIII: 08/06/02 (C) RIV: 07/18/02 (C)
9.	Resolve public comments on draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115	DG-1111, DG-1113: 12/10/02 (C) DG-1114, DG-1115: 01/15/03 (C)
10.	Office review and concurrence of final Regulatory Guides and Generic Letter	DG-1111, DG-1113: 01/31/03 (C) DG1114, DG-1115, and GL 2003-XX: 03/24/03 (C)
11.	Brief ACRS on final Regulatory Guides and Generic Letter	04/10/03 (T)

MILESTONES	DATE (T/C)
12. Brief CRGR on final Regulatory Guides and Generic Letter	04/08/03 (T)
13. Commission Information Paper on Generic Letter	05/03 (T)
14. Issue final Regulatory Guides and Generic Letter	06/03 (T)

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

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

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

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

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

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

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

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

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

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

<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: Four draft regulatory guides, numbered DG-1111, DG-1113, DG-1114 and DG-1115, have been issued for public comment. Proposed Generic Letter 2002-XX, “Control Room Envelope Habitability,” (ADAMS accession number ML021430317) was published on May 9, 2002, at 67 FR 31385. The staff has completed review and disposition of comments received during the public comment period and has completed making necessary revisions to the draft guides and generic letter. Regulatory Guides 1.xxx, formerly DG-1111, DG-1113, and DG-1114, and Generic Letter 2003-xx have NRR and OGC approval and have been sent to the ACRS and CRGR for their review. Regulatory Guide 1.xxx, formerly DG-1115, has NRR approval and has been sent to OGC for approval and the ACRS and CRGR for their review. The staff will prepare a Commission Information Paper for the generic letter and a temporary instruction for inspection.

The staff’s proposed changes to technical specifications for control room emergency ventilation systems have been presented to the Technical Specifications Task Force, and industry has said that they will prepare a package to address the staff’s proposal. On December 30, 2002, NEI sent Industry/TSTF Standard Technical Specification Change Traveler TSTF-448, “Control Room Habitability,” to the NRC for consideration.

NEI provided Revision 1 to NEI 99-03, "Control Room Habitability Assessment Guidance," on March 11, 2003. Staff is currently assessing the impact of the revision on the finalization of the Generic Letter and Regulatory Guides.

Power reactor licensees that have performed integrated tracer gas leakage testing of their control room envelopes continue to inform the NRC staff of their findings.

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

References:

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

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

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

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

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

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

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

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

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

DAVIS-BESSE LESSONS LEARNED TASK FORCE RECOMMENDATIONS REGARDING OPERATING EXPERIENCE PROGRAM EFFECTIVENESS

TAC No.	Description	
MB7280	Develop Operating Experience Action Plan	Last Update: 03/31/03 (Initial Update) Lead Division: DRIP
MB7347	Overall Assessment of Agency's Operating Experience Program	Supporting Divisions: DE, DSSA, DIPM, & DLPM Supporting Office: RES & Regions

Milestone	Date (T=Target) (C=Complete)	Lead	Support
Part I - Operating Experience Program: Objective Phase			
1. Form Task Force with Steering Committee and develop Charter.	03/03 (C)	NRR/RES	
b. Identify desirable agency operating experience program objectives and attributes, and 2.a. Provide documented staff proposals of operating experience program objectives and attributes. 2.b. Obtain executive management endorsement.	04/03 (T) 04/03 (T) 05/03 (T)	Task Force	DRIP, DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions
Part II - Operating Experience Program: Assessment Phase			
1. Define functional needs/areas and processes to meet objectives and attributes.	9/03 (T)	Task Force	DRIP, DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions
2. Review and evaluate current processes. [LLTF 3.1.6(1)]	9/03 (T)	Task Force	DRIP, DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions

Milestone	Date (T=Target) (C=Complete)	Lead	Support
3. Identify areas for improvements. [LLTF 3.2.4(1)]	09/03 (T)	Task Force	DRIP, DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions
4. Task Force issues draft report.	09/03 (T)	Task Force	
5. Task Force provides final report to Steering Committee documenting its specific program improvement proposals.	11/03 (T)	Task Force	
6. Steering Committee makes recommendations to office management on improvements to be made. 6.a Responsible organizations achieve consensus on proposals to implement.	12/03 (T) 12/03 (T)	Steering Committee NRR/RES	 Regions
Part III - Operating Experience Program: Implementation Phase			
1. Develop implementation plan based on 6.a in Part II. 1.a Implement specific improvements per implementation plan (1/04-12/04). [LLTF 3.1.6(2)] [LLTF 3.1.6(3)] [LLTF 3.3.4(2)]	01/04 (T) 12/04 (T)	NRR/RES	Regions
2. Establish processes to monitor effectiveness.	06/04 (T)	NRR/RES	Regions
Part IV - Inspection Program Enhancements			
1. Provide training and reinforce expectations to NRC managers and staff members to address the following areas: (1) maintaining a questioning attitude in the conduct of inspection activities; (2) developing inspection insights stemming from the DBNPS event relative to symptoms and indications of RCS leakage; (3) communicating expectations regarding the inspection follow-up of the types of problems that occurred at DBNPS; and (4) maintaining an awareness of surroundings while conducting inspections. Training requirements should be evaluated to include	12/03 (T)	DIPM	DE, DSSA, DET/RES, Regions

Milestone		Date (T=Target) (C=Complete)	Lead	Support
	the appropriate mix of formal training and on-the-job training commensurate with experience. Mechanisms should be established to perpetuate these training requirements. [LLTF 3.3.1(1)]			
2.	Implement actions to maintain NRC expertise by ensuring that NRC inspector training includes: (1) boric acid corrosion effects and control; and (2) PWSCC of nickel based alloy nozzles. [LLTF 3.3.5(1)]	12/03 (T)	DIPM	DE, DSSA, DET/RES, Regions

Description: Initiatives to assess and improve the agency's reactor operating experience program has been initiated and ongoing for some time. Also, the report of the Davis-Besse Lessons Learned Task Force (LLTF), issued on September 30, 2002, contains a number of recommendations on operating experience program improvements. It is important to note that opportunities to improve access and use of operating experience information will continue in parallel with the systematic assessment of the agency's operating experience program described in this action plan.

Historical Background: Up until 1999, the Office of Analysis and Evaluation of Operational Data (AEOD) performed various activities pertinent to systematically collecting and evaluating operating experience, and communicating the lessons learned to the NRC staff and the regulated industry. With the abolishment of AEOD per SECY-98-228, "Proposed Streamlining and Consolidation of AEOD Functions and Responsibilities," October 1, 1998, the roles and responsibilities of AEOD associated with the operating experience program were transferred to the Offices of Nuclear Regulatory Research (RES) and Nuclear Reactor Regulation (NRR). NRR was generally assigned the short-term operating experience reviews and RES long-term operating experience activities.

Since this time, both NRR and RES have recognized the need to make operating experience more efficiently available to users. RES has made substantial advances in making existing databases available through the internal web. These databases include licensee event reports (LERs), INPO's EPIX database, and monthly operating reports. RES uses these data to provide initiating event frequencies, safety system reliabilities, component failure probabilities, and common-cause failure parameter estimates, as well as related insights. The RES internal web page, for which significant further advances are already planned, will allow NRC staff easier and more timely access these estimates, related trends, and insights in a more timely manner. In addition, the RES internal web site will provide a new expanded LER search tool for use by NRC staff. It is planned that in April 2003, the accident sequence precursor (ASP) database will be accessible through the RES internal web site to the NRC staff. In September 2003, this will be followed by an expanded web site that will further integrate presently contained in separate databases and NUREG and NUREG/CR reports. NRR has similarly improved communications of its short term operating experience program outputs through web technology and is currently replatforming its events and assessment database.

However, despite individual program improvements, the effectiveness of the agency wide program has been questioned. Many believed that the current program activities should be more proactive, risk-informed, and integrated. Many also indicated that the insights gained and lessons learned from operating experience reviews should be better communicated to the users. In addition, both NRR and RES recognized that the governing agency policy, i.e., Management Directive 8.5, “Operational Safety Data Review,” December 23, 1997, and various guidance documents clearly needed updates. In late 2001, NRR created the Operating Experience Section (OES) under the Division of Regulatory Improvement Programs (DRIP). In late 2002, OES spearheaded an effort to assess the agency’s overall operating experience program by soliciting support from various organizations responsible for agency’s program activities. As a result, the Operating Experience Working Group has since been formed to better coordinate the multi-office effort for assessing and improving the agency’s overall operating experience program.

One of the NRC follow-up actions to the Davis-Besse event was formation of a LLTF. The LLTF conducted an independent evaluation of the NRC’s regulatory processes pertinent to the event in order to identify and recommend areas of improvement applicable to the NRC and the industry. A report summarizing their findings and recommendations was published on September 30, 2002. The report contains several consolidated lists of recommendations. The LLTF report was reviewed by a Review Team (RT), consisting of several senior management personnel appointed by the EDO. The RT issued a report on November 26, 2002, endorsing all but two of the LLTF recommendations, and placing them into four overarching groups. On January 3, 2003, the EDO issued a memo to the Directors of NRR and RES, tasking them with developing action plans for accomplishing High-Priority items in the four groups. This Action Plan addresses the assessment and improvement of the agency’s operating experience program. It also addresses the recommendations of the Davis-Besse LLTF regarding operating experience program effectiveness. All of the seven High-Priority recommendations in “Assessment of Operating Experience, Integration of Operating Experience into Training, and Review of Program Effectiveness” grouping are included in this Action Plan.

Proposed Actions: This Action Plan describes the key high-level steps for the agency’s operating experience overall program review, which goes beyond the scope of the Davis-Besse LLTF recommendations. This approach is expected to be more effective than addressing only the LLTF items separately from the overall operating experience program review. The High-Priority LLTF items are specifically designated in the milestones under appropriate Parts or steps to address the requirements prescribed in the January 3, 2003, Tasking Memorandum. The designated LLTF items represent only a subset of multiple activities for the corresponding milestone.

The milestones are grouped into Parts I, II, III, and IV.

Part I is associated with defining the objectives and attributes of the agency’s desirable operating experience program and receiving the endorsement from the agency’s executive management. An interoffice Task Force will be formed to perform the activities in Parts I and II. An interoffice (NRR, RES, and Regions) executive Steering Committee will also be formed to guide the Task Force activities. A Charter describing the goals and responsibilities of the Task Force will be jointly developed by the offices. The purpose of this Task Force is to complete the milestones described in the objective and assessment Phases (Parts I and II of this Action Plan) by December 31, 2003.

Part II describes the milestones associated with the assessment phase of the agency’s overall operating experience program review. These assessment activities will be performed and completed by the Task Force. The scope of the assessment phases will include, but is not necessarily limited to, those operating experience functions identified by SECY-98-228. The output of the assessment activities will be the development of specific proposals for improvement in functional areas to effectively achieve the objectives established in Part I. The Task Force will issue a draft report for review when its preliminary observations, conclusions, and proposals are identified. The Task Force will subsequently provide a final report to the Steering Committee documenting its specific program improvement proposals and the basis for those proposals. The Steering Committee will make recommendations to the offices on improvements to be made an office management will make appropriate assignments. The target date for the Part II milestones is December 31, 2003.

The Part III improvements would include a number of actions that could significantly improve the agency's overall operating experience program effectiveness. These actions will be taken by line organizations in accordance with an implementation plan in response to the recommendations by the Steering Committee. The implementation plan is expected to contain both short-term and long-term improvements. The short-term improvements are expected to be implemented starting in early 2004 and long-term improvements in mid- to late 2004. Actions are expected to require significant interoffice coordination and interaction. If the improvements requires significant changes to the policy, resource, or organizational structure, interactions with the Commission would be necessary. Meetings and communications with both internal and external stakeholders, e.g., INPO, are also expected and encompassed within the scope of the milestones listed in Parts II and III. The target date for completion all the Part III milestones is December 31, 2004.

Part IV lists the two inspection-related High-Priority LLTF items that are focused on enhancing inspection activities.

Originating Documents:

Memorandum from Travers, W.D. to Collins, S. and Thadani, A. C., dated January 3, 2003, "Actions Resulting From The Davis-Besse Lessons Learned Task Force Report Recommendations." (ML023640431)

Memorandum from Paperiello, C.J. to Travers, W.D., dated November 26, 2002, "Senior Management Review of the Lessons-Learned Report of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head." (ML023260433)

Memorandum from Howell, A.T. to Kane, W.F., dated September 30, 2002, "Degradation of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head Lessons-Learned Report." (ML022740211)

Regulatory Assessment: The agency performs a broad range of activities that relate to collection, assessment, feedback, and dissemination of nuclear reactor operating experience. The main purpose of these activities is to generate valuable insights and lessons learned from operating experience and provide feedback to the NRC regulatory programs and the industry. The output of these activities should positively influence both the NRC regulatory programs and the nuclear industry performance. These operating experience program activities provide mechanisms for an independent assessment of the effectiveness of the current NRC regulatory programs and activities and generate long-term, historical, and objective perspectives on individual nuclear power plant and industry performance.

The LLTF recommended that the effectiveness of the current operating experience program be evaluated. As stated earlier, a systematic review of the overall operating experience program has been ongoing and would proceed according to this Action Plan.

Again, the regulatory basis for the agency's current operating experience functions generally stems from the roles and responsibilities defined in SECY-98-228. Any changes in the organizational and/or functional responsibilities defined in this SECY will likely require Commission consultation.

Current Status: This is the initial update for this Action Plan, which addresses the recommendations of the Davis-Besse LLTF regarding operating experience program. The milestones also include management oversight efforts and continuing interaction between the NRC, industry and other stakeholders.

Contacts:

NRR Lead PM:	Ian Jung, RORP, 415-1837
NRR Technical Contact:	Terrence Reis, RORP, 415-3281
DSSA Lead Contact:	Michael Johnson, SPSB, 415-3183

DIPM Lead Contact:	Cynthia Carpenter, IIPB, 415-4006
DRIP Lead Contact:	William Beckner, RORP, 415-3281
DLPM Lead Contact:	Herbert Berkow, LPDII, 415-1485
DE Lead Contact:	Goutam Bagchi, 415-3005
DET/RES Lead Contact:	Nilesh Chokshi, 415-0190
DRAA/RES Lead Contact:	Patrick Baranowsky, OERAB, 415-7493
DSARE/RES Lead Contact:	John Flack, REAHFB, 415-8742
Regional Offices:	Charles Casto, Region II, 404-562-4600

References:

Management Directive 8.5, “Operational Safety Data Review,” December 23, 1997.

SECY-98-228, “Proposed Streamlining and Consolidation of AEOD Functions and Responsibilities,” October 1, 1998.

Table 1
LLTF Report Recommendations (High Priority)

RECOMMENDATION NUMBER	RECOMMENDATION
3.1.6(1)	The NRC should take the following steps to address the effectiveness of its programs involving the review of operating experience: (1) evaluate the agency's capability to retain operating experience information and to perform longer-term operating experience reviews; (2) evaluate thresholds, criteria, and guidance for initiating generic communications; (3) evaluate opportunities for additional effectiveness and efficiency gains stemming from changes in organizational alignments (e.g., a centralized NRC operational experience "clearing house"); (4) evaluate the effectiveness of the Generic Issues Program; and (5) evaluate the effectiveness of the internal dissemination of operating experience to end users.
3.1.6(2)	The NRC should update its operating experience guidance documents.
3.1.6(3)	The NRC should enhance the effectiveness of its processes for the collection, review, assessment, storage, retrieval, and dissemination of foreign operating experience.
3.2.4(1)	The NRC should assess the scope and adequacy of its requirements governing licensee review of operating experience.
3.3.4(2)	The NRC should strengthen its inspection guidance pertaining to the periodic review of operating experience. The level of effort should be changed, as appropriate, to be commensurate with the revised guidance.
3.3.1(1)	The NRC should provide training and reinforce expectations to NRC managers and staff members to address the following areas: (1) maintaining a questioning attitude in the conduct of inspection activities; (2) developing inspection insights stemming from the DBNPS event relative to symptoms and indications of RCS leakage; (3) communicating expectations regarding the inspection follow-up of the types of problems that occurred at DBNPS; and (4) maintaining an awareness of surroundings while conducting inspections. Training requirements should be evaluated to include the appropriate mix of formal training and on-the-job training commensurate with experience. Mechanisms should be established to perpetuate these training requirements.
3.3.5(1)	The NRC should maintain its expertise in the subject areas by ensuring that NRC inspector training includes: (1) boric acid corrosion effects and control; and (2) PWSCC of nickel based alloy nozzles.

ATTACHMENT 2

GENERIC COMMUNICATION AND COMPLIANCE ACTIVITIES

Summary Report

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

TAC NO. AGE	TAC TITLE LEAD ORG

TAC NO.	TAC TITLE
AGE	LEAD ORG

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

Summary Report (

03/31/2003)

TAC NO.	TAC TITLE
AGE	TAC CLOSED
LEAD ORG	

MB1537

MB6813

MB6838

DIPM

3

02/04/2003

02/10/2003

RISK-INFORMED INITIATIVES

ATTACHMEN

RISK-INFORMED INITIATIVES

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
1. Reactor Oversight Process - Enhanced performance indicators (PIs)	- Joint NRC/industry working group continues to meet periodically to develop consistent approach for safety system unavailability reporting - Conducted public workshop for MSPI Pilot Program ()(1/03) - MSPI Pilot data collection phase completed (3/03)	- Developing mitigating systems performance index (MSPI) for unavailability and unreliability of plant systems - Bench marking and data analysis continuing through 6/03.	- Brief ACRS on MSPI pilot (7/03) - Assess feasibility of enhanced (risk-based) PIs for containment using LERF models - Improve current set of PIs
- Industry-level performance indicators in the Industry Trends Program (ITP)	- Issued SECY-02-0058, “Results of the ITP and Status of Ongoing Development” (4/02) - Briefed Commission and ACRS (5/02) - Briefed ACRS on Initiating Events Performance Index (IEPI) and threshold development (11/02)	- Drafting annual SECY on status of ITP (4/03) - Developing Initiating Events Performance Index (IEPI) based on relative contribution of risk significant initiating events - Developing risk-informed thresholds for ex-AEOD PIs - Developing risk-informed thresholds for ROP PIs - Brief ACRS on IEPI and threshold development (5/03)	- Update data and develop risk-informed thresholds for operating experience information, including system reliability

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
1. Reactor Oversight Process (Continued) - Significance determination process (SDP)	- The SDP task group recommendations were evaluated by the affected program offices to formulate plans to complete their response to the recommendations. - DIPM, DSSA, DEDR, and OIG met on March 26, 2003, to discuss implementation of SDP improvement initiatives.	- The SDP task group recommendations have been evaluated and action plans developed to integrate into existing SDP improvement initiatives.	- Implement all elements of the SDP improvement plan - Complete the bench marking process of risk notebooks. - Develop enhanced pre-solved risk tables to simplify phase 2 process.

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

A. CURRENT INITIATIVES			
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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 industry concepts for initiatives 4, 5, and 7. - Safety evaluations written for CE and BWR topical reports on initiative 1 - Writing safety evaluation on CE topical report on initiative 6.	- Define “pilot” efforts to support initiative 4 and 5 - Continue reviews of other initiatives

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4. Fire protection	- NFPA-805 national standard was issued in April 2001. (NFPA-805 is an alternative performance-based risk-informed fire protection standard for nuclear power plants.)	- The CommissionSRM issued 10/03/02 directed the staff to publish the proposed rule in the *Federal Register* for 75 days. Comment period ended January 15, 2003, and comment resolution is underway. - NEI is interacting with the staff regarding its effort to separately develop implementation guidance for NFPA-805. NRC plans to endorse the implementation guidance via Regulatory Guide.	- Publish final rule in Spring 2004 (10 CFR 50.48) - Publish RG endorsing NFPA 805 implementing guidance.

A. CURRENT INITIATIVES			
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4. Fire protection (cont.)	Circuit Analysis Resolution Program (CARP) - Staff has revised the CARP and is obtaining technical assistance from BNL and SNL to develop a risk-informed approach to resolve the circuit analysis issues.	- The staff has issued a Draft NUREG/CR developed by BNL and SNL. This NUREG/CR compiles the history, Regulations, existing staff guidance (GL, IN, etc.), and provides new guidance on risk-informing the fire protection inspection of post-fire safe-shutdown analysis. This NUREG/CR in concert with NEI 00-01 Draft D form the background material for the public workshop that was held on 2/19/03 with stakeholders. General agreement was reached regarding methods to identify: (1) risk-significant circuits, (2) circuits requiring further research, and (3) circuits not of significant risk so that EGM-98-02 may be withdrawn and inspections resumed in this area. Subsequently, draft input to inspection guidance has been developed and shared with the public (ML030780326).	- The staff is working on issuance of a Regulatory Information Summary (RIS) to provide the regulatory footprint for this issue. The staff plans to withdraw EGM-98-02 and resume inspection in this area.

A. CURRENT INITIATIVES			
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5. Safeguards NOTE: This effort is now the responsibility of the Office of Nuclear Security and Incident Response	- Proposed revisions to 10 CFR 73.55 sent to Commission 6/4/01. Proposal requires that licensees' security programs employ risk insights in identifying target sets of equipment necessary to prevent core damage and/or spent fuel sabotage and create a more performance oriented basis for security regulations. - Proposed 73.55 returned by Commission to staff for rework to reflect lessons learned from September 11, 2001, events.	- Subsumed by staff efforts on post-September 11, 2001, Response to Terrorist Activities.	- Subsumed by staff efforts on post-September 11, 2001, Response to Terrorist Activities.
6. 10 CFR 50.69 rulemaking - risk-informing scope of special treatment requirements	- Pilot plants completed IDP review of categorization, with staff observation - Draft rule language made available for public comment on NRC web site. (Notice of Availability published in November 29, 2001, *Federal Register*); revised drafts posted April 5 and August 2, 2002 - Proposed rule package sent to Commission in paper dated September 30, 2002 .	- On March 28, 2003, Commission approved publishing proposed rule for 75 day public comment period - Proposed rule is being prepared for publication	- Complete review of industry guidance documents - Publish proposed and final rules (10 CFR 50.69)

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
7. RIP50/Option 3 (risk-informing technical requirements)	- Developed framework document to guide Option 3 efforts		
- Combustible Gas Control (10 CFR 50.44)	- Published proposed rule changes to 10 CFR 50.44 on August 2, 2002. - The public comment period closed on October 16, 2002. Comments have been evaluated.	- Final rule package is being prepared	- Publish final rule changes to 50.44
- Fracture Toughness Requirements(10 CFR 50.61)	- Draft technical basis for risk-informed revisions to requirements provided by RES to NRR	- Staff is reviewing the RES recommendations and is continuing to develop technical basis for rulemaking	- Publish proposed and final rule changes to 50.61
- Emergency Core Cooling System (ECCS) requirements (10 CFR 50.46)	- Commission SRM on SECY-02-0057 directed rulemaking on: 1. LOCA maximum break size 2. ECCS acceptance criteria 3. LOCA with coincident LOOP - Staff met with BWROG to discuss their "safety case" approach for risk-informing requirements related to LOCA-LOOP	- Staff is developing plans in response to SRM	- Publish proposed and final rule changes to 50.46

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
8. PRA standards	- ASME standard completed on Level 1 and Level 2 LERF PRA (full power) - Staff prepared SECY paper informing Commission of intent to write Reg Guide addressing use of PRA standards (including ASME PRA standard) and industry peer review process for regulatory applications - Reviewed industry guidance on peer reviews - Issued DG-1122 for public comment	- Continuing work with ANS on external events, low power and shutdown, and internal fires - Revising DG-1122 based on review of public comments - Provide ASME with comments for future revision of standard	- Issue final regulatory guide
9. Creating a risk-informed environment	- Three (3) NRR all employee division meetings held to brief staff on results of current environment assessment -Task order for FY 2003 contract assistance put in place	- Plan for pilot activities developed; implementation underway. - Pilot edition of electronic newsletter on risk-informed activities developed and issued.	- Complete risk-informed environment pilot projects - Develop office-wide implementation plan based on results of pilot activities

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
10. Licensing issues associated with non-LWRs	- NRR issued SECY-02-0180, "Legal and Financial Policy Issues Associated with Licensing New Nuclear Power Plants," October 7, 2002. - RES issued SECY-03-0047, “Policy Issues Related to Licensing Non-LWR Reactor Designs,” March 28, 2003.	- The SRM on SECY-02-0180 endorsed staff positions, so no additional action is required at this time.	- RES/NRR staff will continue to formulate policy issues associated with licensing non-LWRs and engage the Commission as appropriate.
11. Advanced Reactor Regulatory Framework	- Staff met internally to discuss options for an advanced reactor risk-informed regulatory framework. Focus on how framework for new reactors is integrated with ongoing risk-informed initiatives. - NEI submitted a white paper on May 7, 2002, (Accession #: ML021350406)	- RES staff will review NEI white paper as part of their efforts to develop an advanced reactor regulatory framework - NRR/DRIP staff will ensure that efforts for item 13, Improving Coherence Among Risk Informed Activities, are coordinated and integrated to the extent possible with advanced reactor framework development.	
12. Construction Inspection Program reactivation	- Use of risk insights in the Construction Inspection Program is being proposed by NEI.	- Ongoing meetings with NEI	

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
13. Improving Coherence Among Risk Informed Activities	- Staff plans and activities discussed at ANS conference (PSA '02) in Detroit, Michigan - Staff developed detailed coherence plan - Public meetings held on 12/5/02 and 3/12/03	- Sharing draft Process for a Risk-Informed Coherence Effort with stakeholders - Reviewing staff programs and processes	- Hold additional public meetings to gather stakeholder input - Keep Commission informed of progress
14. Risk-Informed Regulation Implementation Plan (RIRIP)	- Last published July 12, 2002 (SECY-02-0131)	- Update provided by EDO to Commission on March 23, 2003	- Publish semiannual updates

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