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DIRECTOR'S STATUS REPORT

on

GENERIC ACTIVITIES

Action Plans

Generic Communication and
Compliance Activities

OCTOBER 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
(FINAL UPDATE)

Open TAC Nos.: MA0792, MA1926, MA1927, MA2326, MA2328, MA3673, MA4203, MA4464, MA4465, MA4467, MA4468, MA5012, MA5140, MA7356, MA9111, MB0271, MB3949, MB7191, MB8969, MB8970, MB9014, MB9698, MB9765, MB9766, MB9767, and MB9899.

Last Update: 10/14/03
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)
2. Current Internal Reviews of BWRVIP Documents	
○ Internal Core Spray Piping and Sparger Replacement Design Criteria (BWRVIP-16)	01/15/04 (T)
○ Roll/Expansion of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)	03/13/98 (CD)
○ Internal Core Spray Piping and Sparger Repair Design Criteria (BWRVIP-19)	01/15/04 (T)
○ Technical Basis for Part Circumferential Weld Overlay Repair of Vessel Internal Core Spray Piping (BWRVIP-34)	12/20/03 (T)
○ Underwater Weld Repair of Nickel Components (BWRVIP-44)	11/15/03 (T)
○ Top Guide / Core Plate Repair Design Criteria (BWRVIP-50)	01/15/04 (T)
○ Jet Pump Repair Design Criteria (BWRVIP-51)	01/15/04 (T)
○ Shroud Support and Vessel Repair Design Criteria (BWRVIP-52)	01/15/04 (T)
○ Standby Liquid Control Line Repair Design Criteria (BWRVIP-53)	01/15/04 (T)
○ Lower Plenum Repair Design Criteria (BWRVIP-55)	01/15/04 (T)
○ LPCI Coupling Repair Design Criteria (BWRVIP-56)	01/15/04 (T)
○ Instrument Penetrations Repair Design Criteria (BWRVIP-57)	01/15/04 (T)
3. Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)	12/15/03 (T)
○ Shroud Vertical Weld Inspection and Evaluation Guidelines (BWRVIP-63)	11/20/03 (T)

MILESTONES	DATE (T/C)[1]
○ BWR Core Shroud Inspection & Flaw Evaluation Guidelines (BWRVIP-76)	11/20/03 (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 (CA)
4. Evaluation of Guidelines for Selection and Use of Material for Repairs to BWR Internals (BWRVIP-84)	10/20/03 (T)
5. Evaluation of BWR Supplemental Surveillance Program Capsules D, G, and H (BWRVIP-87)	01/10/04 (T)
6. Guide for Format and Content of BWRVIP Repair Design Submittals (BWRVIP-95)	10/15/03 (T)
7. Sampling and Analysis Guidelines for Determining the Helium Content of Reactor Internals (BWRVIP-96)	12/30/03 (T)
8. Guidelines for Performing Weld Repairs to Irradiated BWR Internals (BWRVIP-97)	03/15/04 (T)
9. Crack Growth Rates in Irradiated BWR Stainless Steel Internal Components (BWRVIP-99)	05/15/04 (T)
10. Updated Assessment of the Fracture Toughness of Irradiated SS for BWR Core Shrouds (BWRVIP-100)	12/30/03 (T)
11. Evaluation and Recommendations to Address Shroud Support Cracking in BWRS (BWRVIP-104)	12/30/03 (T)
12. Testing and Evaluation of BWR Supplemental Surveillance Program Capsules E, F, and I (BWRVIP-111)	03/15/04 (T)
13. BWR Vessel and Internals Project River Bend 183 Degree Surveillance Capsule Report (BWRVIP-113)	05/15/04 (T)
14. BWR Vessel and Internals Project RAMA Fluence Methodology Theory Manual (BWRVIP-114)	01/15/04 (T)
15. RAMA Fluence Methodology Benchmark Manual-Evaluation of RG 1.190 Benchmark Problems (BWRVIP-115)	01/15/04 (T)
16. Integrated Surveillance Program Implementation for License Renewal (BWRVIP-116)	05/09/04 (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: 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. In addition, the BWRVIP committee is presenting BWRVIP-17 to the Code as a permanent repair.

A meeting to discuss BWRVIP technical issues is scheduled to be held on November 4, 2003. Issues to be discussed include: various BWRVIP reports under review, NDE uncertainty, noble metal chemical addition status, and cracking in 316L stainless steel.

A management meeting to be held between the BWRVIP and BWROG Executives and the NRC management will be held on November 5, 2003. Topics of discussion will include: priority BWRVIP issues, noble metal chemical addition, steam dryer issues, NRC budget/resources, etc.

The Boiling Water Reactor Internals Action Plan will no longer be included in the DQSR program, as we consider the action plan complete. In 1994, core shroud cracking was the most significant of reported internals cracking. Therefore, GL 94-03 was issued to request BWR licensees to inspect their core shrouds by the next outage and to justify continued safe operation until inspections could be completed. In an effort to lower the number of industry and staff resources that were needed, it was important for the staff to interact 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. At that time, an action plan was developed to track industry and staff actions. To date, the BWRVIP has made significant efforts to support plant-specific submittals and the only remaining reviews include those of topical reports that are considered routine in nature. Therefore, NRR has determined that the BWR Internals Action Plan is complete.

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: 09/30/03
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)	Staff completes review and issues safety evaluation on pilot plant application (NEI 97-06, TG: 2, 3, 4, 7)	TBD Note 12	DE E. Murphy	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.15 (MB0631)	Hold steam generator workshop with stakeholders (page 2 of Ref. 1; page 2 of Ref. 2)	02/27/01 (C) ML010820457	DE R. Rothman	
1.16 (MA4265)	Staff completes review of generic package and issues model SE for TSTF in *FR* for public comments (NEI 97-06)	TBD (T) Note 12	DRIP K. Kavanagh	DE E. Murphy
1.17 (MA4265)	Publish Notice of Availability of TSTF in *FR* (NEI 97-06)	TBD (T) Note 12	DRIP K. Kavanagh	
1.18 (MA4265)	Staff briefs the Commission on regulatory framework (NEI 97-06, and WITS Item 199400048)	05/29/03 (C)	K. Karwoski	
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)	05/16/03 (C) ML023540491	DE L. Lund	
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. Klingler	
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) Issue NRR input for incorporation into OEDO initiative ○ 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 (C)	RES W. Krotiuk	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) ML030230822	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) ML030230822	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) ML030230822	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) ML030230822	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) ML030230822	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 (C) ML032080002 (Non-public)	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. (Pgs. 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. (Pgs. 12-13) (See Notes 3 and 5)	09/30/05 (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. (Pgs. 46-47, 12-15) (See Notes 3 and 5) Specific tasks include:			
	a) Perform system level analyses to assess the impact of plant sequence variations (e.g., pump seal leakage and SG tube leakage).	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.	TBD (T) See Note 14	RES C. Boyd	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.	TBD (T) See Note 13	RES C. Boyd	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.	TBD (T) See Note 13	RES C. Boyd	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.	TBD (T) See Note 18	RES J. Page	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.	TBD (T) See Note 18	RES J. Page	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. Page	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.	05/28/04 (T) Note 17	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 C. Boyd H. Woods
3.5 (MB7216)	Develop improved methods for assessing the risk associated with SG tubes under accident conditions. (Pgs. 47, 16-20) (See Note 5) Specific tasks include: a) Development of an integrated framework for assessing the risk for the high-temperature/high-pressure accident scenarios of interest.	 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) Issue report describing improved methods and appropriate treatment of uncertainty for identifying severe accident scenarios that lead to challenges of the reactor coolant pressure boundary.	06/28/03 (C) ML031810770 See Note 16	RES H Woods	DSSA S. Long
	c) Identify scenarios and develop logic framework for improved PRA models of the scenarios identified above, including the impact of operator actions.	02/27/04 (T) See Note 16	RES H. Woods	DSSA S. Long
	d) Using the 3.5(b) methods and (c) model logic, calculate the frequency of containment bypass events at an example plant, make indicated method improvements, and document the improved methods and results.	02/27/04 (T) See Note 16	RES H. Woods	DSSA S. Long
	e) Extend the 3.5(d) improved methods to include consideration of core damage sequences initiated by secondary depressurization events (such as MSLB design basis accident scenarios) that induce tube rupture.	TBD See Note 16	RES H. Woods	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.6	To address an ACRS report conclusion that improvements can be made over the current use of a constant probability of detection (POD) for flaws in SG tubes, RES has recently completed an eddy current round robin inspection exercise on a SG mock-up as part of NRC's research to independently evaluate and quantify the inservice inspection reliability for SG tubes. This research has produced results that relate the POD to crack size, voltage, and other flaw severity parameters for stress corrosion cracks at different tube locations using industry qualified teams and procedures. Complete analysis of research results and prepare topical report to document the results. (Pgs. 47, 33)	12/31/01 (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. (Pgs. 48, 28-29) (See Note 5)	04/26/03 (C) ML031150674	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) ML020070081	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) TBD (T) Note 11 TBD (T) Note 11 TBD (T) Note 11 TBD (T) Note 11	DSSA M. Hart	
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. b) Using the extensive experience on stress corrosion cracking in operating SGs, and results from laboratory testing under prototypic conditions, develop models for predicting the cracking behavior of SG tubing in the operating environment. c) Based on the knowledge accumulated on stress corrosion cracking behavior and the properties of eddy current testing, attempt to explain the observed relationship between changes in eddy current signal voltage response and crack growth.	12/31/05 (T) 12/31/06 (T) 12/31/05 (T)	RES J. Muscara RES J. Muscara RES J. Muscara	DE E. Murphy DE E. Murphy DE E. Murphy
3.11	In order to resolve GSI 163, it is necessary to complete the work associated with tasks 3.1 through 3.5 and 3.7 through 3.9. Upon completion of those tasks, develop detailed milestones associated with preparing a GSI resolution document and obtaining the necessary approvals for closing the GSI, including ACRS acceptance of the resolution. (See Note 9)	12/31/05 (T)	DLPM DE E. Murphy	DSSA S. Long
3.12	Develop outline and a detailed schedule for completing DG 1073, “Plant Specific Risk-Informed Decision Making: Induced SG Tube Rupture (See Note 9)	12/31/05 (T)	DE E. Murphy	DSSA S. Long

Notes:

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

2. NRC has entered into an agreement in April 2003 with Paul Scherrer Institute (PSI) of Switzerland, to participate in the ARTIST program. Testing is to commence in 2004 and is scheduled to be complete in 2007. Some preliminary experimental data from the initial phase of testing will be available in 2004.

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. .The scope of the work is being re-evaluated.

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. Based on staff comments, the Catawba submittal was revised on

July 30, 2003. By letter dated September 9, 2003, the industry submitted a revised TSTF package to be consistent with the July 30, 2003 submittal for Catawba. The staff recently issued an RAI to Catawba on 9/17/03 and expects a response by 10/15/03 from the licensee. The last remaining technical issue is how to address the safety factors to burst in the new framework and what the appropriate safety factors are.

Additionally, the staff issued a Commission Paper (SECY-03-0080) on May 16, 2003, and briefed the Commission on May 29, 2003. In the paper/briefing, the staff discussed its basis for concluding that there is reasonable assurance of tube integrity, the progress made in revising the regulatory framework, and the remaining issues to be addressed in the Catawba submittal. The staff also discussed the process by which the Catawba and TSTF reviews would be conducted. The staff expects to complete its review of the lead plant submittal three months after receipt of the RAI responses, and issue the generic safety evaluation six months after receipt of the final generic submittal (which the staff expects to receive near the completion of the Catawba review).

13. This work will be incorporated into the new RES plan to address the severe accident induced steam generator tube integrity issue. The new plan includes and updates the base case and a set of sensitivity studies that need to be completed before the analysis is completed. Once the new RES plan is formalized, scheduled target dates will be assigned to unfinished tasks. This modification to the plan resulted from the lessons learned and the findings of the work completed so far.

14. This work is partially complete. RES is developing an updated plan to address the severe accident induced steam generator tube integrity issue. This new plan will incorporate the existing SG Action Plan milestones and add new tasks to address issues that may not be specifically outlined in the SG Action Plan. Once the new RES plan is complete, scheduled target dates will be assigned to unfinished tasks. This modification to the plan resulted from the lessons learned and findings of the work completed so far.

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.

16. Lessons learned from the work completed so far necessitated modifications to the milestones and target completion dates that are being formalized in the RES operating plan. Scheduled completion date for item 3.5.e will be provided when available.

17. The results from this item feed into the program on primary system component response to severe accident condition action plan and are not needed until the new date. Hence, the item is rescheduled for budget flexibility.

18. The scope and schedule for these milestones are being re-evaluated to be consistent with the next update of the RES operating plan.

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. This review process was subsequently revised as described in SECY-03-0080 (see Note 12).

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.
- September 9, 2002	Issuance of a letter to NEI transmitting staff comments on the draft generic license change package (ML022520413)
- 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.
- May 16, 2003	Issuance of SECY-03-0080, “Steam Generator Tube Integrity (SGTI) - Plans for Revising the Associated Regulatory Framework.”
- May 29, 2003	Staff briefing of the Commission on the status of SG Regulatory Framework Modifications. An industry briefing preceded the staff briefing.
- September 4, 2003	Public meeting between NRC, Duke Power, and NEI on lead plant submittal.

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EMERGENCY ACTION LEVEL GUIDANCE DEVELOPMENT
(FINAL UPDATE)

TAC No.: MA3695 Revision to NESP-007 Last Update: 09/30/03
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)	05/15/03 (C)
17. Obtain Office (NRR/OGC/NMSS/RES) concurrence and or waiver.	05/15/03 (C)
18. Obtain Committee (CRGR/ACRS) concurrence and or waiver.	06/13/03 (C)
19. ADM publish final Regulatory Guide in FR.	07/28/03 (C)

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 has completed the concurrence phase, including comment resolutions. The regulatory guide revision endorsing the latest NEI guidance has been provided to the Office of Research for issuance and distribution.

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 revised regulatory guide has completed the NRC office concurrence process. Concurrence comment resolution had led to a projected schedule completion slip. The projected date of issuance of the RG is the end of July 2003.

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.
20. Memorandum from B. Boger to J. Larkins, May 14, 2003.
21. Memorandum from R. Borchardt to C. Ader, May 15, 2003.
22. Email from C. Ader to R. Borchardt, June 3, 2003.
23. Memorandum from J. Larkins to W. Travers, June 13, 2003.

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

TAC No.	Description	
MB7281	Develop Action Plan	Last Update: 09/30/03 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]			
2. 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 (C) ML031390010	DIPM	Regions
4. Provide training on procedure changes.	09/03 (C)	DIPM	

Milestone	Date (T=Target) (C=Complete)	Lead	Support
5. Issue procedure changes.	09/03 (C)	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.	06/03 (C) MI031890873	DIPM	Regions
2. Develop enhanced structure to the inspection approach used for IMC 0350 plants.	08/03 (C)	DIPM	Regions
3. Develop draft revisions to IMC and issue for regional comment.	10/03 (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 (C) ML030300067	DLPM	
3. Follow up with Offices and Regions to determine effectiveness of training.	12/03 (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: Part 1 and Part 2 milestone activities have been initiated and those scheduled for this quarter were completed. The Part 3 milestones were completed as scheduled, and a new milestone to follow up on the effectiveness of training was added. The Action Plan lead participated in a meeting with industry representatives in June.

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

SIGNIFICANCE DETERMINATION PROCESS (SDP) IMPROVEMENT (INITIAL UPDATE)

TAC Nos. MA9164, MB0046, & MB2203

Last Update: Initial Update
Lead Division: DIPM
Supporting Division: DSSA

Mission: To improve the effectiveness and efficiency of the Significance Determination Process (SDP), consistent with the vision. The Plan delineates assigned responsibilities and completion dates for the tasks to achieve the stated objectives.

Coordinator: Peter Koltay, IIPB/DIPM/NRR

Task		Completion Date	Lead	Status
1. Improve Focus on Early Resolution of Specific Technical Questions and Internal Staff Disagreements				
Objective 1.1	Implement a weekly management status report on SDP issues in process. **[SDP 3.9.3(1)]**	04/01/02 (C)	IIPB	SDP Activities Tracking List implemented 2/1/02 to address Objectives 1.1-1.2.
Objective 1.2	Incorporate features to provide for early identification of SDP issues that are likely to become untimely due to technical, policy, or process issues. **[OIG - 6]**	06/30/04 (T)	IIPB	The active issues matrix focused management attention on timeliness issues. The timeliness metric improved from 57% in FY 02 to 73% in FY 03. The goal was 75%. The FY 04 goal is 80%.
Objective 1.3	Develop and track/trend SDP timeliness metrics within ROP Self-Assessment Process, including the cycle-time calculation for major process steps.	06/28/02 (C)	IIPB	IMC 0307, Reactor Oversight Process Self-Assessment Program, incorporates the relevant timeliness metrics.
Objective 1.4	Implement a requirement to conduct a self-assessment for SDP results that are not timely.	06/28/02 (C)	IIPB	Lessons learned evaluations are initiated on an as-needed basis.
Objective 1.5	Rectify the difference between the NRR Operating Plan and IMC 0307 for SDP timeliness. [SDP 3.9.3(3)]	10/1/03 (C)	IIPB	Timeliness goals in the NRR Operating Plan are referenced in IMC 0609.

Task		Completion Date	Lead	Status
Objective 1.6	Incorporate SDP timeliness metrics Into the Regional Operating Plans. **[SDP 3.9.3(1)]**	12/30/03 (T)	IIPB	Ongoing.
Objective 1.7	Change IMC 0307 "ROP Self-Assessment Program" to improve evaluation of inspection effectiveness in timely identification of performance deficiencies inspection**.** **[OIG - 5]**	12/30/03 (T)	IIPB	This metric will be incorporated into the Active Issues Matrix.
2. Improve SDP Process				
Objective 2.1	Revise Attachment 1 of IMC 0609 to clarify the roles and responsibilities of the SERP, to include an escalation process for resolution of issues for which the SERP cannot reach a consensus position, and to include process timeliness goals.(1)		IIPB	IMC 0609 Att. 1 was revised April 30, 2002, to incorporate this enhancement to the SERP process.
a.	Clearly define the accounting process of the 90 day time period including: Starting time End time	08/01/02 (C)		Guidance is provided in IMC 0609 Att. 1 and tracked under Objective 1.1 of the Plan.
b.	Communicate the Agency's timeliness goals to licensees (e.g., Choice Letters, Regulatory Conferences, Reg. Information Conference, etc.). **[SDP 3.9.3(2)]**	12/01/03 (T)		IIPB will submit a change to IMC 0609.01 (Choice Letter) to communicate timeliness goals to licensees.
c.	Improve the SERP process: Clearly identify SERP participants and define their respective roles and responsibilities in IMC0609.01.	06/28/02 (C)		IMC 0609 Att. 1 was revised April 30, 2002, to identify SERP participants and their roles and responsibilities.

Task	Completion Date	Lead	Status
d. Outline the escalation process for issues where the SERP fails to reach consensus in IMC0609.01.	06/28/02 (C)		IMC 0609 Att. 1 was revised April 30, 2002, to outline the escalation process when SERP fails to reach Concensus.
e. Improve the Regulatory Conference process and associated activities: Designation of NRC participants Post conference caucus Post conference re-SERP Post conference SDP and re-SERP.	6/28/02 (C)		IMC 0609 Att. 1 was revised April 30, 2002, to improve the effectiveness of the Regulatory Conference and post-conference caucus.
Objective 2.2 Engage the regions to confirm their understanding and implementation of the expectations regarding use of the SDP, including guidance on the level and type of licensee engagement that is appropriate during the conduct of:(2) **[SDP 3.2.3(2)]**		IIPB Support: SPSB	Routine bi-weekly teleconferences are held with the Regions. NRR emphasizes expectations noted in IMC 0609 and the August 9, 2002, memorandum from S. Collins to the Regional Administrators on "Reactor Oversight Expectations for Inspector Use of the Significant Determination Process". This memo provides specific instructions on the level and type of licensee engagement for each phase of the SDP.
a. SDP Phase 2 risk analyses.	08/01/02 (C)		
b. SDP Phase 3 risk analyses.	08/01/02 (C)		
c. Communicate expectations for inspector use of the phase 2 notebooks during interim period in which enhanced pre-solved tables are being developed.	08/01/03 (C)		A revised "Expectations" memorandum provides clear instructions regarding the use of the phase 2 risk notebooks during the development of the pre-solved SDP tables. ML031270689

Task		Completion Date	Lead	Status
Objective 2.3	Issue guidance on the use of the site specific risk-informed inspection notebooks within the overall context of the SDP.[2,3]		IIPB Support: SPSB	Based on experience gained from the initial notebook benchmarking efforts and ROP implementation, additional notebook usage guidelines were developed and presented to SRAs for discussion and comments. The final version of the guidelines were incorporated into "Expectations Memorandum" and IMC 0609.
a.	Use of the revision 0 risk notebooks (pre-benchmarking).	05/31/02 (C)		
b.	Use of the benchmarked risk notebooks, revision 1.	05/31/02 (C)		
c.	Guidance when additional analysis beyond the capability of the risk notebooks needs to be conducted.	05/31/02 (C)		
Objective 2.4	Evaluate revising the SDP to require that the preliminary characterization of potentially risk significant issues be "potentially greater than green," rather than a specific color.[2] **[SDP 3.9.3(4)]**		IIPB	This issue was presented to the DRP/DRS Division Directors during the August 20-21, 2002, counterpart meeting and the proposed change to 0609 was issued for review and comment. The proposed revision to the 0609 guidance was also discussed during the January 2003 ROP public meeting.
a.	Collect and evaluate regional input.	01/31/03 (C)		
b.	Make final determination on changing the process to preliminary greater than green, or stay with the existing process or preliminary specific color.	04/30/03 (C)		The revision to IMC 0609 Att. 1 was issued on March 21, 2003, to allow for the use of "greater than green" preliminary SDP characterization.

Task		Completion Date	Lead	Status
Objective 2.5	Assemble a focus group of internal stakeholders to identify key SDP-related issues going forward and provide recommendations for their resolution, consistent with the ROP principles and objectives.(3)		IIPB Support: SPSB, Regions	The SDP Task Group was formed consisting of regional and headquarters staff. A charter was developed and the SDPTG completed a comprehensive review of the SDP and provided recommendations to enhance the overall effectiveness of the process. The recommendations have been accepted by NRR and incorporated into this Plan, as noted.
a.	Identify focus group members.	05/01/02 (C)		
b.	Develop charter.	06/28/02 (C)		
c.	Present recommendations.	12/20/02 (C)		
Objective 2.6	Develop a plan for long range improvements to the SDP.(3) **[OIG-1]**	Complete		The SDP Improvement Initiative Task Action Plan is NRR's tool for tracking SDP improvement activities.
a.	Issue the proposed SDP basis document, including the current performance expectations for the Phase 2 notebooks. The risk notebook "construction rules" should also be included or referenced in the proposed SDP Basis Document. **[SDP 3.2.3(1)] [SDP 3.6.3(4)]**	03/31/04 (T)	IIPB, SPSB	The At-Power SDP Basis Document is expected to be issued by December 31, 2003, and should include the phase 2 risk notebook construction rules.b.
b.	Re-evaluate the performance expectations of the SDP tools after completion of the risk notebook benchmarking and modify program guidance, as appropriate, to reflect any revisions to the expectations. **[SDP 3.2.3(3)]**	04/30/04 (T)	IIPB	Evaluation of SDP effectiveness and performance expectations is conducted as part of the routine annual assessment process outlined in IMC 0307.
3. Improve SDP Tools				
Objective 3.1	Revise IMC 0609 App. A to improve the guidance for conducting a phase 2 analysis to:(3)		SPSB Support: IIPB	

Task	Completion Date	Lead	Status
a. Develop tools and simplify the process of accounting for external initiators in phase 2 of the SDP.	6/30/04 (T)		IMC 0609, Appendix A issued March 8, 2002, included guidance for screening external initiators. However, specific guidance on how to calculate risk contribution is not provided. SPSB designated a senior risk analyst to form a task group responsible for the development of methodology for the evaluation of risk contribution from external events.
- Form Task Group to identify methodology for the assessment of external event contributions.	12/31/03 (T)	DSSA	
- Develop and test methodology.	02/28/04 (T)		
- Incorporate methodology into IMC 0609 for implementation.	10/31/04 (T)		
b. Clarify the guidance on the treatment of concurrent issues.	04/01/02 (C)		Guidance incorporated in March 18, 2002, revision to IMC 0609 App. A Section III.
c. Develop pre-solved risk tables developed from existing benchmarked phase 2 risk notebooks. **[SDP 3.1.3(2)] [SDP 3.6.3(1)] [OIG-1]**	10/30/05 (T)	IIPB	This activity is not budgeted for completion in FY ‘04.
d. Evaluate training needs and issue revised guidance for the use of the pre-solved risk tables.	10/30/05 (T)	IIPB	Supplemental training needs will be evaluated prior to issuance of the pre-solved risk tables.

Task		Completion Date	Lead	Status
Objective 3.2	Develop a plan to benchmark and revise all of the site specific risk-informed inspection notebooks (Revision 1). Develop and implement a quality assurance (QA) plan for the development of revision 1 to the site specific risk-informed inspection notebooks.(2, 3) **[SDP 3.1.3(2)]**		SPSB Support: IIPB	
a.	Schedule and complete benchmarking plan (site visits)	10/01/03 (C)		All benchmarking trips completed. Final Revision 1 notebooks are available to internal stakeholders on the DSSA/SPSB and SRA web pages.
	- Standardize risk-informed inspection notebooks.	12/31/04 (T)		Notebooks that were benchmarked during the early stages of the initiative will be revised to incorporate lessons learned from the benchmarking process. This may require approximately 10 additional site visits.
	- Complete the basis document.	3/30/04 (T)		
b.	Develop and implement QA plan for development of the site-specific risk notebooks.	03/01/02 (C)		QA Plan developed and provided to BNL for development of the phase 2 risk notebooks.
c.	Implement a process to compare the results of the QA'd SPAR models and benchmarked phase 2 risk notebooks. **[SDP 3.6.3(3)]**	09/30/03 (C)		All benchmarking has been completed. Outcomes continue to be verified.
d.	Develop risk notebook maintenance schedules to review and update the phase 2 tools to address licensee PRA changes and/or plant modifications. [SDP 3.6.3(2)]	4/30/05 (T)		This will be accomplished on a case by case basis.

Task	Completion Date	Lead	Status
Objective 3.3 Develop or improve existing SDP tools as applicable in the following areas: **[OIG-3]**			
a. Fire protection	05/31/04 (T)	SPSB	Comments will be evaluated at each level of stakeholder involvement. Training will be developed in conjunction with the review process. The development of inspection guidance for manual actions and associated circuits is in parallel with the SDP which will be used to assess findings in those areas.
- Issue draft to all stakeholders for comment	10/20/03 (T)		
- Public workshop	10/30/03 (T)		
- Table top benchmarking	12/31/03 (T)		
- User training	02/04/04 (T)		
b. Maintenance rule	12/31/03 (T)	SPSB	Internal stakeholder comments are being incorporated.
c. Containment	12/31/03 (T)	SPSB	Issued to industry and NRC regions for review and comments May 03. Public meeting participation in July. Issue final document 12/03.
d. Steam generator tube integrity	02/28/04 (T)	SPSB	Public meeting with NEI participation was held on 09/24/03. Some industry comments have been resolved. A new draft SDP will be issued to stakeholder comments at the October ROP meeting. Industry workshop proposed for December 2003 or January 2004.

Task		Completion Date	Lead	Status
e.	Shutdown	12/30/03 (T)	SPSB	SDP presented to NEI and SRAs July 2002 and October 2002. Workshop held January 2003. Enhanced Appendix G to be issued November 2003. Some training has been completed, and additional training of SRA is ongoing on an as needed basis.
f.	Spent Fuel	12/31/04 (T)	IIPB	Under development.
Objective 3.4	Improve the physical protection SDP, if necessary, accounting for any safeguards policy changes.	12/31/04 (T)	NSIR Support: IIPB	This item is dependent on the availability of resources from The Office of Nuclear Security and Incident Response (NSIR).
Objective 3.5	Develop a database of all completed phase 3 analyses.(3)	10/01/02 (C)	SPSB	Database of submitted phase 3 analyses was created and is accessible via the SPSB web page. Continuing to add information.
Objective 3.6	Consider development of analysis criteria and standards for conducting detailed phase 3 analysis.(3) **[SDP 3.5.3(2)] [OIG-4]**	6/28/02 (C)	SPSB, Support: RES, Regions	11/26/02, RES developed procedures incorporating high level ASP guidance. The documents were provided to the SRAs for review to determine applicability to the phase 3 SDP.
a.	Identify participating RES and NRR personnel and establish responsibilities and a completion schedule.	8/30/03 (C)		NRR/SPSB and RES initiated the Risk Assessment Standardization Project (RASP) to develop standard methodologies and procedures for conducting phase 3 analyses.
b.	Develop criteria and to allow the staff to recognize situations where “the state of knowledge” correlation, which is described in RG 1.174, might warrant a Phase 3 analysis. **[SDP 3.7.3(1)]**	12/30/04 (T)		The Risk Assessment Standardization Project will evaluate the possibility for developing advanced risk criteria for recognizing when modeling parameter

Task	Completion Date	Lead	Status
			uncertainties warrant a more in-depth analysis to properly characterize the significance of an inspection finding.
c. Develop guidance to allow the staff to determine whether the results of a licensee's risk analysis of a finding is of sufficient quality to use as an input to the staff's final significance determination. **[SDP 3.11.2.3(1)] [OIG-4]**	12/30/04 (T)		Risk Assessment Standardization Project will evaluate the feasibility to provide guidance on an acceptable approach to determine the quality of PRA results that may be used to support ROP decision-making process.
Objective 3.7 Evaluate accelerating the SPAR Model Development Program (i.e., Revision 3i SPAR models, low power/shutdown models, LERF models, and external events analysis capability).(2)		RES	In the response to SRM CMEXM-01-0001, RES committed to complete SPAR Model QA in line with the risk notebook benchmarking by the end of FY 2003.
a. Develop Rev. 3i SPAR models.	9/30/02 (C)		Complete.
b. Complete onsite QA verification (benchmarking) of Rev. 3i SPAR models.	10/31/03 (C)		Complete.
c. Develop Low Power/Shutdown model.	12/31/05 (T)		RES is developing generic templates for each class of licensed reactor plants. Four models have been completed.
d. Develop LERF model	12/31/06 (T)		Draft event trees have been developed.
4. Improve Staff Training in The Use of SDP Tools			
Objective 4.1 Develop and conduct training on the use of the site specific risk-informed inspection notebooks. Develop initial and periodic refresher training on the SDP.(3)		IIPB Support: SPSB	

Task		Completion Date	Lead	Status
a.	Develop training materials for IMC 0609A revision.	4/15/02 (C)		Complete.
b.	Complete IMC 0609A training at inspector counterpart meetings: **[OIG-3]**	10/01/02 (C)		Complete.
	Region I			
	Region II			
	Region III			
	Region IV			
c.	Encourage regions to conduct annual SDP refresher training during routine inspector seminars. **[SDP 3.5.3(1)]**	6/30/03 (C)		Refresher training will be provided by regional and headquarters SRAs on an annual basis.
d.	Develop systematic assessment of training needs in the area of risk, with a particular focus on identifying and advancing the knowledge, skills, and abilities (KSAs) for implementing the SDP. **[SDP 3.5.3(3)]**	1/31/04 (T)		NRR's Risk Informed Environment Initiative and IMC 1245 Working Groups are engaged in evaluations of the necessary skills and training needs as they relate to understanding and using risk in regulatory activities. Based on their evaluations, the groups will make recommendations to enhance the training program for inspectors and risk analysts and propose improvements to staff processes, practices, and infrastructure.
Objective 4.2	Increase staffing and/or staff development in the areas of shutdown risk, seismic, fire protection, and containment risk analysis. **[OIG-3]**	6/30/02 (C)	IIPB Support: SPSB	NRR has staffed additional SRA positions within SPSB. The newly hired staff is currently completing required training for SRA certification.

Task		Completion Date	Lead	Status
5. Improve Clarity of Risk-Informed ROP Decision Guidance				
Objective 5.1	Develop improved criteria on the cost-benefit decision of ceasing to refine risk analyses when the benefit is not justifiable. **[OIG-3]**	08/01/03 (T)	IIPB Support: SPSB	The staff will continue efforts to develop cost-benefit decision-making criteria for continuing phase 3 analysis when the benefit may not be justifiable.
Objective 5.2	Develop guidance that defines the attributes of a minimally acceptable risk-informed decision for use within the ROP. **[OIG-3]**	03/26/03 (C)	IIPB Support: SPSB	The attributes for reaching the minimally acceptable risk-informed decision are described in IMC 0609 Att. 1, Exhibit 4.
Objective 5.3	Revise the ROP guidance to explicitly indicate that traditional engineering analysis considerations (e.g., reduction of safety margin, or significant loss of defense-in-depth) should be used to determine an appropriate color to associate with findings where the uncertainty in the risk evaluation arising from the characterization of the impact of the inspection finding is large enough that the color is indeterminate on the basis of the risk analysis. This guidance should promote consistency and be used only where the uncertainty is significant (i.e., when alternate assumptions yield results which vary over more than two orders of magnitude). **[SDP 3.7.3(2)]**	12/31/05 (T)	IIPB Support: SPSB Regions	IIPB is in the process of identifying findings where this could be applicable and developing guidance for evaluating issues when there is a significant reduction of safety margin or loss of defense-in-depth.

Task		Completion Date	Lead	Status
6.	**Clarify Expectations for ASP and SDP Process Coordination**			
Objective 6.1	Issue guidance to delineate the role of the Office of Research in the SDP, in order to minimize the potential for unexpected or unreasonable differences in the results of the SDP and ASP processes. Explore efficiencies and quality enhancements that would result in better coordination and/or integration of these two programs. **[SDP 3.11.1.3(1)]**	06/30/04 (T)	IIPB Support: RES	Currently, based on a user need memo, RES reviews all greater than green issues and provides a quarterly assessment of the specific implementation of the process. The Risk Assessment Standardization Project will explore the development of common methodology for evaluating risk under both the SDP and ASP. See also Objective 3.6.
a.	NRR and RES should identify avenues to enhance the staff's knowledge of the ASP program, including adding a module to the P-111 course regarding the ASP program. **[SDP 3.11.1.3(2)]**	12/31/04 (T)	SPSB	This issue is under Review by the IMC 1245 Working Group.

(1) Staff Requirements Memorandum M010720A of August 2, 2001, which resulted from the Commission briefing on the results of initial implementation of the reactor oversight process held on Friday, July 20, 2001.

(2) Staff Requirements Memorandum of February 5, 2002, resulting from COMEXM-01-0001, D.C. Cook Potential Red Finding, and the Implementation of the Significance Determination Process Within the Reactor Oversight Program

(3) Response to Differing Professional View NRR-02-DPV-02, dated February 18, 2002, concerning the continued performance of significance determination process phase 2 analysis

(4) Memorandum dated December 20, 2001, from Ellis Merschoff, Regional Administrator, Region IV, and Frank Congel, Director, Office of Enforcement, to Samuel Collins, Director, Office of Nuclear Reactor Regulation, on the treatment of programmatic issues by the SDP.

Description: In conjunction with IMC 2515, "The Policy For the Light-Water Operating Reactor Inspection Program", IMC 0609, "The Significance Determination Process (SDP)", was developed to assist the staff in using risk insights, where appropriate, to help NRC inspectors and staff determine the safety significance of inspection findings. The appendices to IMC 0609 support safety cornerstones

associated with the strategic performance areas as defined in IMC 2515. The SDP determinations for inspection findings and the Performance Indicator (PI) information are combined for use in assessing licensee performance in accordance with guidance provided in IMC 0305, "Operating Reactor Assessment Program."

The SDP is an essential component in the ROP that serves to improve the objectivity of the ROP so that subjective decisions and judgment are not central process features. The SDP is an objective, risk-informed, and scrutable process that ensures that NRC resources are focused on those aspects of plant performance having the greatest impact on safe plant operation and that NRC actions have a clear tie to licensee performance.

Historical Background: In SECY-99-007, "Recommendations for Reactor Oversight Process Improvements," dated January 8, 1999, the staff provided its recommendations to the Commission for improving the reactor regulatory oversight processes, including proposed changes to the NRC's inspection, assessment, and enforcement processes. The staff's efforts to develop the proposed changes was guided by three objectives: 1) improve the objectivity of the [reactor] oversight process so that subjective decisions were not central process features; (2) improve the scrutability of these processes so that NRC actions have a clear tie to licensee performance; and (3) risk-inform the process so that NRC and licensee resources are focused on those aspects of performance having the greatest impact on safe plant operations. With respect to the assessment process, the staff sought to develop a process that would allow the integration of various information sources relevant to licensee safety performance. In SECY-99-007, the staff concluded that adequate assurance of licensee performance would be achieved through the use of risk-informed performance indicators (PIs) and inspection findings. The staff also highlighted the need to develop a method for characterizing the risk of inspection findings and indicated that a "level of risk significance, based on a risk scale, will be determined and documented for the findings."

In SECY-99-007A, "Recommendations For Reactor Oversight Process Improvements" (follow-up to SECY-9-007), Attachment 2, dated March 22, 1999, the staff introduced the Significance Determination Process (SDP) as the method for characterizing the risk of inspection findings. The SDP was designed to assess only those inspection findings associated with at-power operations in the Reactor Safety Strategic Performance Area cornerstones of Initiating Event (IE), Mitigating Systems (MS) and Barrier Integrity (BI); however, concepts for characterizing the risk significance of inspection findings in the emergency preparedness, radiation safety, and safeguards areas were under development. The SDP provided a means to screen out inspection findings that have minimal or no risk significance and trigger a more detailed analysis of potentially risk-significant findings.

To support the start of the initial implementation of the revised Reactor Oversight Process (ROP) in April 2000, the staff issued Inspection Manual Chapter (IMC) 0609, "Significance Determination Process." Appendix A to IMC 0609 provided guidance for the staff to estimate the unintended increase in risk during at-power plant conditions caused by deficient licensee performance. The guidance was intended to provide a simplified probabilistic framework for use by the staff in identifying potentially risk significant findings in the reactor safety area--either the IE, MS, or BI cornerstones.

When the ROP was initially implemented in April 2000, the staff's efforts to develop the Phase 2 notebooks for each nuclear plant were still in progress. As a result, the draft notebooks that were made available for staff use at initial ROP implementation were considered to be incomplete. By late 2000, the staff had made sufficient progress in the site visits associated with the development of Phase 2 SDP notebooks, that it began to issue the "Revision 0" notebooks to the sites. After issuance of the first

Rev. 0 notebooks, the staff identified problems with the accuracy of the notebooks and concluded that benchmarking was needed to confirm the adequacy of the notebooks. Using NRC risk analysts and contractor resources, the staff began its efforts to benchmark the notebooks in April 2001. As of November 12, 2002, the staff had issued 24 Revision 1, Phase 2 notebooks.

In a memorandum dated November 8, 2001, Troy Pruett, Senior Reactor Analyst, Region IV, submitted a differing professional view (DPV) to the Director of the Division of Reactor Safety in Region IV. The DPV expressed concerns about the performance of the SDP Phase 2 analyses. An Ad Hoc Panel, appointed by the Regional Administrator by memorandum dated November 16, 2001, was formed to review the DPV and make appropriate recommendations. The DPV Panel documented its findings in a report to the Region IV Administrator dated January 10, 2002. This report was forwarded to the Director, NRR, for program office consideration and appropriate action. In a memorandum dated February 18, 2002, the Director, NRR informed Mr. Pruett of the results of the review of his DPV. Mr. Pruett expressed several concerns with the results of the DPV review and, in a memorandum to the EDO dated March 15, 2002, recommended an independent review of the concerns in his DPV. Through a memorandum dated April 9, 2002, the EDO convened an Ad Hoc panel to review Mr. Pruett's DPO.

The DPO Panel completed its review and issued conclusions and recommendations in a report dated June 28, 2002. The DPO Panel generally agreed with the overall analysis performed by the DPV panel and its response to Mr. Pruett's recommendations. The DPO Panel found that "NRC management and staff are in the process of addressing many of the Ad Hoc DPV Panel's observations and recommendations in the SDP Improvement Initiative." However, the DPO Panel also recommended that the NRC conduct an independent review of the SDP assessment tools.

Between May and October 2001, the OIG conducted an audit of the SDP. The objectives of the audit, as indicated in the OIG's report (OIG-02-A-15) dated August 21, 2002, were to determine whether (1) the SDP is achieving desired results, (2) NRC staff clearly understand the process, and (3) NRC staff are using [the] SDP in accordance with agency guidance. In its report, OIG concluded that "while the SDP is meeting is objectives and agency staff are using SDP in accordance with guidance, additional refinements are needed." The report provided a number of recommendations, including that the NRC develop an action plan to correct Phase 2 analysis weaknesses or eliminate this portion of the SDP.

<u>Proposed Actions</u>: In a memorandum to the Director, NRR dated August 6, 2002, the EDO directed that a plan be developed to address both the DPO Ad Hoc Panel and OIG recommendations. The EDO's memorandum indicated that this "plan shall address the DPO Panel recommendation for an overall objective review of the SDP." The plan developed by the Director, NRR included the formation of the SDP Task Group to conduct an independent review of the SDP.

Consistent with the Charter, the Task Group's review focused on the SDP for the Reactor Safety Strategic Performance Area and, in particular, issues pertaining to the SDP for the Initiating Events (IE), Mitigating Systems (MS) and Barrier Integrity (BI) Cornerstones. As a result, the Task Group did not perform a detailed review of the SDP for the Radiation Safety Performance Area or Safeguards Performance Area. In addition, because the Emergency Preparedness (EP) Cornerstone SDP was not the focus of the DPO Panel Response or OIG Audit Report, and because the relevant EP SDP issues are the focus of other NRC review activities, the Task Group did not emphasize this area in its review.

The SDP Improvement Task Action Plan (The Plan) was developed to guide staff efforts aimed at implementing the recommendations developed by the SDPTG and lessons learned since initial implementation of the ROP. The Plan delineates responsible organizations, establishes aggressive completion dates, and provides status updates for each of the specified Plan action items.

Originating Documents: Memorandum from S. Collins to V. McCree dated September 18, 2002, “Significance Determination Process Task Group.” (ADAMS Accession No. ML022620580)

Office of Inspector General Audit Report, OIG-02-A-15, “Review of NRC’s Significance Determination Process,” dated August 21, 2002. (ADAMS Accession No. ML022470372)

Memorandum from Johnson, J.W. to Travers, W.D. dated June 28, 2002, “Differing Professional Opinion (DPO) Concerning the Significance Determination Process.” (ADAMS Accession No. ML021830090)

Regulatory Assessment: No adjustment to the current regulatory framework is warranted at this time. The current regulatory framework provides reasonable assurance that operating commercial light-water reactor facilities are safe.

Resource Requirements: The use of contractors will cost approximately $1.9 million. NRR implementation of the improvement plan will require approximately 9.75 FTE. This includes 7.75 FTE from SPSB as the lead and 2 FTE as the support in the completion of the objectives.

Contact:

Peter Koltay, DIPM/IIPB/RIS, 415-0213

References:

SECY 99-007	Recommendations for Reactor Oversight Process Improvements.
SECY 99-007A	Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007).
IMC 0609	The Significance Determination Process.
IMC 2515	Light-Water Reactor Inspection Program -Operations Phase.

Status Summary: N/A

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)
MB7182	Orders for Interim Inspection Guidelines
MB9522	Review of Bulletin 2002-01 Responses
MB8915	Generic Activities for Lower Head Inspection
MB9891	Develop Bulletin 2003-02

Last Update: 09/30/03
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]	07/03 (C) ML032461221 ML032161224	RES/DET	DE

Milestone	Date (T=Target) (C=Complete)	Lead	Support
3. Complete initial evaluation of individual plant inspections in response to Bulletins and Orders. b. Continue to review future inspection results until permanent guidelines are issued.	05/04 (T) Ongoing	DE DE	DLPM Regions DLPM Regions
4. Incorporate Order EA-03-009 requirements into 10 CFR 50.55a 1. Develop rulemaking plan and obtain Commission approval. 2. Publish proposed rule 3. Evaluate/incorporate public comments and publish final rule.	TBD TBD TBD	DE	DRIP DSSA DLPM
5. 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]	09/04 (T) Note (1)	DE	RES/DET DSSA Regions Industry
6. 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
7. Make decision to endorse revised ASME Code requirements, when issued, or implement alternative requirements. [LLTF 3.3.4(8)-High]	09/04 (T) Note (1)	DE	RES/DET
8. If alternative, determine appropriate regulatory tool and establish schedule for implementation.	09/04 (T) Note (1)	DE	DRIP DIPM DSSA RES/DET industry public

Milestone		Date (T=Target) (C=Complete)	Lead	Support
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.	a. Evaluate individual plant responses to Bulletin 2002-01 regarding Boric Acid Inspection Programs (60-day responses and necessary follow-up) b. Issue public document to summarize evaluation of plant responses.	06/03 (C) ML031760568 07/03 (C) ML032100653	DE DE	DLPM DLPM DRIP
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).	06/03 (C)	DE	DLPM DRIP DIPM DSSA Regions
5.	Issue Bulletin 2003-02 on Reactor Vessel Lower Head inspection	08/03 (C) ML032320153	DE	DLPM
6.	Develop milestones for additional regulatory actions, as necessary.	07/03 (C)	DE	DLPM DSSA DRIP
7.	Incorporate revised requirements for inspection of lower head and other RCPB components into 10 CFR 50.55a 1. Develop rulemaking plan and obtain Commission approval. 2. Publish proposed rule 3. Evaluate/incorporate public comments and publish final rule.	 TBD TBD TBD	DE	DRIP DSSA DLPM

Milestone		Date (T=Target) (C=Complete)	Lead	Support
8.	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) Note (1)	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 are based on projected issuance of industry proposals. 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: Part I activities included continued monitoring of outage inspection results, follow-up with plants discovering defects, and evaluation of requests for relaxation from Order EA-03-009. In Part II activities, the review and evaluation of licensee responses to Bulletin 2002-01 regarding BACC were completed. A summary of the evaluation was published in RIS 2003-13. Based on this review and the discovery of leakage on undervessel penetrations at South Texas Project, Bulletin 2003-02 was issued. Also, a decision was made to prepare a request for approval of rulemaking to incorporate revised inspection requirements for RPV lower heads and other RCPB components into 10 CFR 50.55a.

Contacts:

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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:
NRC Bulletin 2003-02, "Leakage From Reactor Pressure Vessel Lower Head Penetrations And Reactor Coolant Pressure Boundary Integrity," August 21, 2003.

Order EA-03-009 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, MB9470, MB4864, MB9931, MC0307, and MC0725/6

Last Update: 10/03/03
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 RG 1.1/RG 1.82 (DG-1107)	 03/00 (C) 10/03 (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))	
○ 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 2003-01	04/03 (C)
○ Information Paper to Commission, Issue Bulletin 2003-01	06/03 (C)
○ CRGR Briefing on proposed draft GL	01/04 (T)
○ Proposed draft GL issued for Public Comment	02/04 (T)
○ Public meeting with Stakeholders during Public Comment period	03/04 (T)
○ Public Comment period ends	04/04 (T)
○ Resolution of Public Comments and revisions to proposed GL made, as necessary	05/04 (T)
○ CRGR Briefing on proposed final GL	05/04 (T)
○ Information Paper sent to Commission, issue GL	08/04 (T)
○ NEI publish PWR Industry Evaluation Guidelines	10/03 (T)
○ NRC starts Reviews of GL Responses and Selective Audits	11/04 (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 was 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, 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 to support 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 and resolution 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 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 used 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 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. 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. 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.

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 demonstrated that excessive buildup of debris on ECCS pump strainers from thermal insulation, corrosion products, and other particulates 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 more efficient and 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 (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.

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 generic communications to address the PWR sump clogging issue outlined above is considered to be appropriate.

These conclusions clearly support this action plan as outlined herein.

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

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

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 sump performance taskforce 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 October 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 issued Bulletin 2003-01 on June 9, 2003, requesting information on compliance within 60 days or information on interim compensatory measures to reduce risk until an evaluation to determine compliance is completed. 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 (NRCB 2003-01, PWR Sumps) Bob Pulsifer, PD I-2, 415-3016 (Containment Coatings, GL 98-04, GE Topical Report)
NRR Lead Technical Reviewer:	Ralph Architzel, SPLB, 415-2804
NRR Technical Contacts:	Rich Lobel, SPLB, 415-2865 Nicholas Saltos, SPSB, 415-1072
RES Technical Contact:	T. Y. Chang, ERAB, 415-6450

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.

NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors” dated June 9, 2003.

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.

NUREG/CR-6808, “Knowledge Base for the Effect of Debris on Pressurized Water Reactor ECCS Sump Performance” dated February 2003.

Letter from Mario V. Bonaca to Nils Diaz, “Draft Final Revision 3 to Regulatory Guide 1.82, “Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident”,” dated September 30, 2003.

GENERIC SAFETY ISSUE (GSI) 189 - SUSCEPTIBILITY OF ICE CONDENSER AND MARK III CONTAINMENTS TO EARLY FAILURE FROM HYDROGEN COMBUSTION DURING A SEVERE ACCIDENT (INITIAL UPDATE)

TAC No. MB7245

Last Update: Initial Update
Lead NRR Division: DSSA
Supporting Division: DLPM
Supporting Office: RES

MILESTONES	DATE (T/C)
1. Transfer GSI from RES to NRR. Issue Resolution Process letter from J. Zwolinski, NRR, to F. Eltawila, RES.	12/2002 (C)
2. Issue Task Action Plan - First draft for issuing Order. Final draft ready for issuing an Order. New draft for Rulemaking.	03/14/03 (C) 04/30/03 (C) 06/30/03 (C)
3. Engage the affected stakeholders: BWROG Management Meeting, ICUG, and NEI.	02/19/03 (C)
3. Review RES and contractor Cost and Benefit Analyses, technical assessment, and supporting/reference material. Conduct additional analyses if required.	02/28/03 (C)
4. Determine best solution and course of action (order, rule making, generic letter, severe accident management guidelines, etc.) Order initially selected.	02/12/03 (C)
5. Prepare regulation and guidance development memoranda and provide results and recommendations to NRR management.	03/05/03 (C) 03/05/03 (C)
6. Brief DLPM Management.	03/06/03 (C)
7. Brief LT and obtain approval for Order.	03/13/03 (C)
8. Distribute Draft Order and draft SECY Letter.	03/26/03 (C)
9. Provide Draft Order to OGC.	03/28/03 (C)
10. Brief ET.	03/19/03 (C)
11. Brief NRR/D.	03/19/03 (C)
12. Draft Secy Letter to EDO.	03/27/03 (C)
13. Finalize CRGR Package.	03/26/03 (C)
Course of action changed per OGC and ET - Will conduct a Public Meeting and pursue Rulemaking	
14. Meet with Rulemaking Committee.	05/05/03 (C)
15. Schedule Public Meeting.	05/14/03 (C)
16. Issue Press Release regarding Public Meeting.	05/29/03 (C)

MILESTONES	DATE (T/C)
17. Public Meeting.	06/18/03 (C)
18. Conduct Post Public Meeting Debrief and determine course of action.	06/18/03 (C)
19. Meet with OPA to develop Communications Plan and Website.	06/24/03 (C)
20. Complete Communications Plan Draft and route for approval.	07/10/03 (C)
21. Complete Website (DEFERRED until after ACRS meeting and decision to proceed).	TBD
22. Meet with Rulemaking Committee to determine if Rulemaking applicable.	November 2003
23. Complete Stage 4, Regulation and Guidance Development, of Management Directive 6.4 and enter Stage 5, Regulation and Guidance Issuance.	After ACRS meeting of 11/06/03
24. Second Public Meeting to address issues regarding design criteria of backup power supply and cost/benefit analysis refinements. (Combine with ACRS meeting).	11/06/03
25. Schedule meeting with ACRS. (M Weston)	November 2003
26. Develop Rulemaking Plan (Action by DRIP, Policy and Rulemaking Program Section).	TBD

<u>Description</u>: To resolve GSI-189, NRR is recommending the addition of a backup power supply for the combustible gas igniters for licensees with Ice Condenser or Mark III containments. The generic issue was proposed in response to SECY 00-198, “Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control).”

<u>Historical Background</u>: The generic issue was proposed (Memorandum to John Flack, Chief, Regulatory Effectiveness and Human Factors Branch, Division of Systems Analysis and Regulatory Effectiveness, RES, from Mark Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, RES, “Information Concerning Generic Issue on Combustible Gas Control for PWR Ice Condenser and BWR Mark III Containment Designs,” August 15, 2001, ML012330522) in response to SECY-00-198, “Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control).” This SECY paper explored means of making 10 CFR 50.44 risk-informed. As a part of this, the paper recommended that safety enhancements that have the potential to pass the backfit test be assessed for mandatory application through the generic issue program.

Under station blackout (SBO) conditions, the PWR ice condenser and BWR Mark III containments are vulnerable to failures from hydrogen (H2) deflagrations or detonations, failures that would otherwise be prevented if the existing H2 igniter system were energized. The 13 susceptible units are: 4 dual unit PWR ice condenser containment stations - McGuire, Catawba, DC Cook, and Sequoyah; the single unit PWR Watts Bar ice condenser containment plant; and, 4 single unit BWR Mark III containment plants - Grand Gulf, River Bend, Clinton, and Perry.

At the request of RES a technical assessment was conducted by: (1) Brookhaven National Laboratory (BNL) to perform the benefits analysis; (2) Information Systems Laboratories (ISL) to perform the cost analysis; and, (3) Sandia National Laboratories (SNL) to perform targeted plant analysis. RES staff has also worked with cognizant NRR staff throughout the development of this technical assessment.

For these analyses, initiating events, core damage frequencies (CDF), conditional containment failure (CCF) probabilities, and release categories were extracted from existing studies. The severe accident progression scenarios, including conditional containment failure probabilities, were based primarily on NUREG-1150, “Severe Accident Risk: An Assessment of Five US Nuclear Plants.” The conditional probability of early failure (CPEF) of containment was taken from NUREG/CR-6427, “Assessment of the DCH [direct containment heating] Issue for Plants with Ice Condenser Containments.” Some plant specific analysis data was also used from Duke Power PRAs and the Sequoyah (ice condenser) and Grand Gulf (Mark III) plants. The combination of this data was then used to develop a benefit-cost analysis enveloping all plants.

The technical assessment quantified the reduction in the conditional containment failure probability associated with combustible gas (H2) control being available during station blackout (SBO) events, which was then converted to a dollar value based on the expected values for averting public exposure and offsite property damage associated with the availability of combustible gas control. These averted costs (benefits) were then compared to the overall cost for the implementation and maintenance of several alternative safety enhancements to determine if there was a potential cost beneficial back-fit.

The RES analyses were based on consideration of internal events only. However, sufficient information was provided in the RES analyses associated with external events for some of the plants to evaluate the impact external events could have on the analyses. When considering external events, averted costs (benefits) increase substantially, generating a larger net positive benefit to cost value. Though the backup power system would not be required to be designed to withstand the external events that could be precursors of the SBO, it is expected that the small, backup power supply will be located in an area capable of withstanding those external events. Therefore, there would not be a substantial increase in the cost of installation attributed to external events. Even though the RES cost/benefit analysis was based on averted costs associated only with internal events, the NRR technical staff believes, based on its review, that additional and/or plant specific analyses are not required to strengthen the basis for rulemaking as a regulatory option to require that the applicable licensees add backup power to one train of igniters. Also, whether external events are included or not in the cost/benefit analysis, the NRR technical staff believes this backfit is a safety enhancement that provides a substantial increase in the overall protection of the public health and safety with implementation costs (for the portable or pre-staged system) that are justified in view of this increased protection [10 CFR 50.109, “Backfitting”, paragraph (a)(3)].

For PWRs with large dry or sub-atmospheric containments, containment loads associated with hydrogen combustion are non-threatening. However, it was discovered in the study associated with NUREG/CR-6427, “Assessment of the DCH [direct containment heating] Issue for Plants with Ice Condenser Containments,” that, for ice condenser containments, the early containment failure probability is dominated by non-DCH hydrogen combustion events, due to the relatively low containment free volume and low containment strength in these designs. These containments rely on the pressure-suppression capability of their ice beds. Therefore, for a design-basis accident, where the pressure is a result of the release of steam from blowdown of the primary (or secondary) system, an ability to withstand high internal pressures is not needed.

In a beyond-design-basis accident, where the core is severely damaged, significant quantities of hydrogen gas can be released. To deal with large quantities of hydrogen, the ice condenser containments are equipped with AC-powered igniters, which are intended to control hydrogen concentrations in the containment atmosphere by initiating limited “burns” before a large quantity accumulates. In essence, the igniters prevent the hydrogen (or any other combustible gas) from accumulating in large quantities and then suddenly burning (or detonating), posing a threat to containment integrity.

For most accident sequences, the hydrogen igniters can deal with the potential threat from combustible gas buildup. The situation of interest for this generic safety issue only occurs during accident sequences associated with station blackouts, where the igniter system is not available because they are AC-powered. Thus, this does not affect the frequency of severe accidents, but does affect the likelihood of a significant release of radioactive material to the environment should such an accident occur.

The issue also applies to BWR Mark III containments because they also have a relatively low free volume and low strength (comparable to those of the PWR ice condenser designs) and are similarly potentially vulnerable in an accident sequence associated with station blackout. Consequently, the Mark III designs also provide hydrogen igniters. The Mark I and Mark II designs are also pressure-suppression designs, but are operated with the containment “inerted,” i.e., the drywell and the air space above the suppression pool are flooded with nitrogen gas and a nitrogen makeup system maintains oxygen level below a set limit by maintaining a slight positive nitrogen pressure within the primary containment.

RES briefed the ACRS on the GSI-189 technical assessment on June 6, 2002, and November 7, 2002, and briefed the ACRS Thermal Hydraulic Phenomena and the Reliability and PRA Sub-committees on November 5, 2002. In a letter to the Commission dated November 13, 2002, the ACRS stated that they agreed with RES that further regulatory action by NRR was warranted for ice condenser and Mark III containments. RES also considered qualitative benefits, such as defense-in-depth, public confidence, and regulatory coherence, in their recommendation to pursue further action to provide backup power to one train of igniters for both ice condenser and Mark III plants. Additionally, RES pointed out that the cost benefit analysis did not consider potential benefits due to averting some late containment failures.

The ACRS suggested that the form of action be through the use of plant-specific severe accident management guidelines (SAMG). Responding to the ACRS letter, a letter from the EDO stated that the NRR staff would engage the affected stakeholders in developing additional information related to implementing various alternatives, including an option of using the severe accident management guidelines. A Public Meeting was held on June 18, 2003, to discuss and receive comments on GSI-189. At that meeting the licensees stated that they did not think that the use of SAMGs was viable because they are not implemented until late in the accident sequence and the igniters might be needed sooner. Also they felt that operator action to install a portable generator was not practical since it could distract operators from more critical activities associated with mitigating the accident. Therefore, NRR is basing its evaluation on a pre-staged system with procedures incorporated into EOPs. This did not change the conclusion that the backfit should be pursued. The NRR staff recommendation will be presented to the ACRS for review and comment before any proposed action to complete resolution of GSI-189 goes to the Commission.

Proposed Actions: Take further action to pursue requiring that back-up power to one train of igniters be provided for ice condenser and Mark III containments.

Meet with Rulemaking Committee and with ACRS to finalize course of action.

Originating Documents: Memorandum to John Flack, Chief, Regulatory Effectiveness and Human Factors Branch, Division of Systems Analysis and Regulatory Effectiveness, RES, from Mark Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, RES, “Information Concerning Generic Issue on Combustible Gas Control for PWR Ice Condenser and BWR Mark III Containment Designs,” August 15, 2001, ML012330522).

SECY 00-198, “Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control).”

Regulatory Assessment: Defense-in-Depth - As pointed out in the analyses, NRR technical staff recognized that there are significant uncertainties in both the cost and benefit calculations done for RES which can shift the benefit to cost from a net negative number to a net positive number. This is why NRR technical staff agreed with RES and ACRS that applying the defense-in-depth philosophy is applicable and appropriate here. One of the prime reasons for defense-in-depth is to manage uncertainties. The NRR technical staff believes that adding a backup power supply provides that defense-in-depth to compensate for those uncertainties.

Backfit Rule - NRR technical staff believes that adding backup power provides a safety enhancement that yields a substantial increase in the overall protection of the public health and safety and has implementation costs that are justified in view of this increased protection, thereby meeting the requirements of the Backfit Rule.

Rulemaking - Licensees do not think implementation can be done under SAMG since the igniters may be needed sooner in the accident scenario.

Current Status: NRR has recommended that backup power be provided to the combustible gas igniters. NRR technical staff will meet with the ACRS on November 6, 2003, to discuss the technical basis for resolve GSI-189. Also, the technical staff wants to reach agreement with ACRS on the technical basis as a resolution to GSI-189 before selecting a regulatory option to implement the resolution. The public, licensees, and other interested stakeholders will be invited to participate.

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2. NUREG/CR-4551, Vol. 3, Rev. 1, Part 1, “Evaluation of Severe Accident Risks: Surry Unit 1, Main Report,” October 1990.
3. NUREG/CR-4551, Vol. 3, Rev. 1, Part 3, “Evaluation of Severe Accident Risks: Surry Unit 1, External Events,” December 1990.
4. NUREG/CR-4551, Vol. 5, Rev. 1, Part 1, “Evaluation of Severe Accident Risks: Sequoyah, Unit 1, Main Report,” December 1990.
5. NUREG/CR-4551, Vol. 6, Rev. 1, Part 1, “Evaluation of Severe Accident Risks: Grand Gulf, Unit 1, Main Report,” December 1990.
6. NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” December 1990.
7. Letter from V. Mubayi, Brookhaven National Laboratory, to H. VanderMolen, NRC, “NUREG-1150 Consequence Calculations,” July 20, 1994.
8. T. D. Brown *et. al.,* “NUREG-1150 Data Base Assessment Program: A Description of the Computational Risk Integration and Conditional Evaluation Tool (CRIC-ET) Software and the NUREG-1150 Data Base,” letter report, March 1995.
9. NUREG/BR-0184, “Regulatory Analysis Technical Evaluation Handbook,” Final Report, January 1997.
10. 10 CFR 50.44, “Standards for combustible gas control system in light-water-cooled power reactors,” January 1, 2000 (last revised 1987).
11. NUREG/CR-6427, “Assessment of the DCH Issue for Plants with Ice Condenser Containments,” April 2000.

12. NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,” July 2000.
13. Memorandum to Samuel Collins, Director, Office of NRR, from Ashok Thadani, Director, Office of RES, September 29, 2000, regarding Research Information Letter RIL-0005, “Completion of Research to Address Direct Containment Heating Issue for All Pressurized Water Reactors.” (ML003755724).
14. Memorandum to Ashok Thadani, Director, Office of RES, to Samuel Collins, Director, Office of NRR, November 22, 2000, regarding Research Information Letter RIL-0005, “Completion of Research to Address Direct Containment Heating Issue for All Pressurized Water Reactors.” ML003761979).
15. NUREG-1742, “Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program, Main Report,” Draft Report for Public Comment, April 2001.
16. Memorandum to John Flack, Chief, Regulatory Effectiveness and Human Factors Branch, Division of Systems Analysis and Regulatory Effectiveness, RES, from Mark Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, RES, “Information Concerning Generic Issue on Combustible Gas Control for PWR Ice Condenser and BWR Mark III Containment Designs,” August 15, 2001 (ML012330522).
17. Memorandum to M. Snodderly (NRC) from M. Zavisca and M. Khatib-Rahbar (ERI), “Combustible Gas Control Risk Calculations (DRAFT) for Risk-Informed Alternative to Combustible Gas Control Rule for PWR Ice Condenser, BWR Mark I, and BWR Mark III (10 CFR 50.44),” October 22, 2001.
18. Management Directive 6.4 (MD 6.4), “Generic Issues Program,” December 4, 2001.
19. Management Directive 6.3 (MD 6.3), “The Rulemaking Process,” July 31, 2001.
20. Memorandum from John H. Flack, Chief, REAHFB:DSARE:RES to Jack E. Rosenthal, Chief, SMSAB:DSARE:RES and Mark A. Cunningham, Chief, PRAB:DRAA:RES, dated February 6, 2002, regarding “Panel Review of GSI-189, Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident.”
21. Memo from Farouk Eltawila, Director, RES, to Ashok C. Thadani, Director RES, dated February 13, 2002, regarding RES Task Action Plan for Resolving Generic Safety Issue 189: “Post Accident Combustible Gas Control in Pressure Suppression Containments.”
22. Memorandum from William Travers, EDO, to The Commissioners, dated May 13, 2002 (SECY-02-0080), Proposed Rulemaking–Risk Informed 10CFR50.44, “Combustible Gas Control In Containment”, (WITS 20010003).
23. Advisory Committee on Reactor Safeguards Meeting Minutes, 493rd Meeting, June 6, 2002, regarding Technical Assessment Generic Safety Issue (GSI)-189.
24. Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment, Rev. 2, September 24, 2002, by Information Systems Laboratories, Inc., Rockville, MD.
25. Hydrogen Control Calculations for the Sequoyah Plant, draft letter report, Rev. 3, September 30, 2002, by Sandia National Laboratories.
26. Memorandum from Ashok Thadani, RES to William Travers, EDO, dated October 1, 2002, regarding, “Revision to NRC’s Regulatory Analysis Guidelines [NUREG/BR-0058] and RES Office Letter 1 to Conform to OMB’s Information Quality Guidelines.”
27. Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice condenser and Mark III Containment Plants, draft letter report, October 4, 2002, by Brookhaven National Laboratory. ADAMS ML022880554.
28. Advisory Committee on Reactor Safeguards Subcommittee on Thermal-Hydraulic Phenomena and Subcommittee on Reliability and Probabilistic Risk Assessment Meeting Minutes, November 5, 2002, regarding Generic Safety Issue (GSI)-189.
29. Advisory Committee on Reactor Safeguards Meeting Minutes, 497th Meeting, November 7, 2002, regarding Technical Assessment Generic Safety Issue (GSI) -189.

30. Memo from George E. Apostolakis, Chairman Advisory Committee on Reactor Safeguards, to the Commission Chairman Richard A. Meserve, dated November 13, 2002, regarding "Recommendations Proposed by the Office of NRR for Resolving Generic Safety Issue -189, Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident. ML023230513
31. Memo from Ashok C. Thadani, Director RES, to Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, dated December 17, 2002, regarding RES Proposed Recommendation for Resolving Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident." ML023510161
32. Attachment to Memo from Ashok C. Thadani, Director RES, to Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, dated December 17, 2002, "Technical Assessment Summary for GSI-189: Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
33. Memo from John A. Zwolinski, Director, Division of Licensing Project Management, NRR to Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, RES, dated January 21, 2003, regarding, "Resolution Process for Generic Safety Issue (GSI) 189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
34. Memo from Jack Rosenthal, Branch Chief, Safety Margins and Systems Analysis Branch, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research to John Hannon, Branch Chief, Plant Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation dated June 19, 2003, regarding, Final Contractor's Reports: Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
35. Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report, Energy Sciences and Technology Department, Brookhaven National Laboratory, December 23, 2002 (ML031700011).
36. Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment, Revision 2, Information Systems Laboratories, Inc., September 24, 2002 (ML031700015).
37. Hydrogen Control Calculations for the Sequoyah Plant, Final Letter Report, March 2003, Prepared By Sandia National Laboratories, March 2003 (ML031700025).

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

TAC No.	Description	
MB7280	Develop Operating Experience Action Plan	Last Update: 09/30/03 Lead Division: DIPM Supporting Divisions: DE, DSSA, & DLPM
MB7347	Overall Assessment of Agency's Operating Experience Program	Supporting Offices: RES & Regions
MB8220	Operating Experience Task Force Activities	

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) ML030900117	NRR/RES	
b. 2.a. 2.b.	Identify desirable agency operating experience program objectives and attributes, and Provide documented staff proposals of operating experience program objectives and attributes. Obtain executive management endorsement.	04/03 (C) 04/03 (C) ML031200312 ML031490535 05/03 (C) ML031350156	Task Force	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 (C)	Task Force	DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions
2.	Review and evaluate current processes. [LLTF 3.1.6(1)]	9/03 (C)	Task Force	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 (C)	Task Force	DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions
4.	Task Force issues draft report.	09/03 (C) ML032740058	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.	12/03 (T)	Steering Committee	
6.a	Responsible organizations achieve consensus on proposals to implement.	12/03 (T)	NRR/RES	Regions
Part III - Operating Experience Program: Implementation Phase				
1.	Develop implementation plan based on 6.a in Part II.	01/04 (T)	NRR/RES	Regions
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)]	12/04 (T)		
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	12/03 (T)	DIPM	DE, DSSA, DET/RES, Regions

Milestone		Date (T=Target) (C=Complete)	Lead	Support
	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. [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.

<u>Proposed Actions</u>: 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.

<u>Originating Documents</u>:

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)

<u>Regulatory Assessment</u>: 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.

<u>Current Status</u>: All Part 1 (Objective Phase) activities are complete. The Operating Experience Task Force was formed, and completed development of program objectives and attributes, which were endorsed by the Steering Committee. Phase 2 (Assessment Phase) activities are in progress. The Task Force delivered its draft report to the Steering Committee in September.

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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

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

Summary Report (07/01/2003 - 09/30/2003)

TAC NO.	TAC TITLE	AGE	LEAD ORG
MB4864	GL: Potential Clogging of Containment Recirculation Sump Screens by Debris Accumulation at PWRs	18	DSSA
MB6585	RIS: Protecting SG Info & Withholding Sensitive Homeland Security Info from Public Disclosure (Reckley (DLPM/Petrone)	12	DLPM
MB6736	RIS: Potential for Water Hammer During Restart of Residual Heat Removal Pumps (R3/Lee)	11	R3
MB6903	RIS: Scope of Required For-Cause Fitness-For-Duty Testing Required by 10 CFR 26.24(a)(3) (NSIR/Stansky/Petrone)	10	NSIR
MB7262	GL: Steam Generator Tube/Tubesheet Inspection Issues (DE/Lund/Petrone)	9	DE
MB7263	IN 80-13, Sup 1 - General Electric Type SBM Control Switches Defective Cam Followers (DIPM/Hodge)	9	DIPM
MB7595	IN: Part 21 Concerning Whiting Cranes Purchased Prior to 1980 (Foster/DIPM)	8	DIPM
MB7682	RIS: NRC Endorsement of NEI 99-01 and Revision of RG 1.101 (Petrone/DIPM/Gibson)	8	DIPM
MB7683	RIS: Clarification of NRC Guidance on Timeliness of Event Classification (Petrone/DIPM/Gibson)	8	DIPM
MB7782	IN: Unanalyzed Condition of Reactor Coolant Pump Seal Leakoff Line During Fire Or Station Blackout (DIPM/Petrone)	8	DIPM
MB7979	GL: Steam Generator Regulatory Framework (Karwoski/EMCB/DE/Petrone)	7	DE
MB7995	RIS: Wireless Communication (Tardiff/NSIR/Petrone)	7	NSIR
MB7997	RIS: Two Badge Method for Est Effective Dose Equivalent From External Rad Sources (Pedersen/DIPM/IEHB/Petrone)	7	DIPM
MB8093	RIS: Licensee Changes to Safeguards and Security Compensatory Measures Implementation (WScott\NSIR\Petrone)	7	NSIR
MB8112	RIS: Proposed Risk Informed Inspector Guid for Post-Fire Safe-Shutdn Associated Circuit Inspections (Sally/DSSA/Petrone)	7	DSSA
MB8219	RIS: Issuance of MD 8.17, Licensee Complaints Against NRC Employees (DAllsopp/DIPM/Petrone)	6	DIPM
MB8394	IN: Loss of Charging System at Millstone (JDozier/DIPM)	6	DIPM
MB8909	IN: Trisodium Phosphate Inventory Under Accident Conditions at Davis-Besse (Holmberg/R3/Hodge)	5	R3
MB8980	RIS: Notif of the Mod of Fed Policy Re The Use of Potassium Iodide as a Thyroidal Blocking Agent (KBrock/DIPM/Petrone)	5	DIPM
MB9135	RIS: RIS 2002-12, Supl 1 - NRC Threat Advisory and Protective Measures System (Stransky/Petrone)	4	NSIR
MB9518	IN: Failures In Safety-Related Power Cables (Koshy/DE/Dozier)	4	DE
MB9666	IN: Buna-N Material Aging in ASCO Scram Solenoids (Koshy/DE/Dozier)	4	DE
MB9955	IN: Point Beach Auxiliary Feedwater Orifices (DIPM/Dozier)	3	DIPM
MC0357	BL: Rebaselining of Data in the Nuclear Materials Management & Safeguards System (NRR/NMSS/NSIR/Harris/Petrone)	2	NMSS/NRR
MC0604	RIS: RIS 2003-05, Sup 2-Issuance of Orders Imposing Add;l Physical Prot Measures for ISFSI Using Dry Stor (NMSS/Petrone)	1	NMSS/NRR

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

Summary Report (07/01/2003 - 09/30/2003)

TAC NO.	TAC TITLE	AGE	TAC CLOSED	LEAD ORG
MB6186	IN: IN 89-69, Supplement 1 - Loss of Thermal Margin Caused by Channel Box Bow (Jimenez (SRXB/DSSA)/Dozier)	13	09/08/2003	DSSA
MB7681	RIS: Clarification of NRC Guidance for Modifying Protective Actions (Petrone/DIPM/Gibson)	5	07/09/2003	DIPM
MB7980	IN: Identification of Degradation Mechanism in Higher Row U-Bends of A Steam Generator (MMurphy/EMCB/DE/Dozier)	6	09/15/2003	DE
MB8406	IN: Chemical Attack of San Onofre, Unit 3, Linestarters (Hodge/DIPM)	3	07/07/2003	DIPM
MB8747	IN: South Texas Project - Lower Reactor Pressure Vessel BMI Boron Issue (MMitchell/DE/Foster)	3	08/28/2003	DIPM
MB8759	IN: Mark 1 Containment Issue (JTrapp/RI/Dozier)	2	07/09/2003	R1
MB8786	IN: Potential Vulnerability to Worm Infection in Plant Computer Network (SLee/DIPM)	4	09/15/2003	DIPM
MB9134	IN: Potential Flooding Through Unsealed Concrete Floor Cracks (WJones(R4)/Petrone)	1	07/09/2003	R4
MB9522	RIS: NRC Review of Responses to Bulletin 2002-01 (TSullivan/SCoffin(DE)/Petrone)	2	08/13/2003	DE
MB9691	IN: IN 2002-26, Sup 1 - Failure of Steam Dryer Cover Plate After a Recent Power Uprate (Foster(DIPM)/Lyon)	2	08/13/2003	DIPM
MB9891	BL: Leakage fr RPV Lower Head Penetrations & Rx Coolant Pressure Boundary Integrity (Reckley/Monarque/DLPM/Petrone)	2	09/04/2003	DLPM
MB9898	IN: Recent Motor-Operated Valve Maintenance Experience (DIPM/Hodge/DE/Scarbrough)	2	09/23/2003	DIPM
MB9942	IN: Degradation of Criticality Monitoring System at BWX Technologies (NMSS/DIPM/Petrone)	1	08/25/2003	NMSS/NR
MB9946	RIS: Preparation and Scheduling of Operator License Exams (Barnhill/DIPM/Petrone)	2	09/15/2003	DIPM

ATTACHMENT 3

RISK-INFORMED INITIATIVES

RISK-INFORMED INITIATIVES

A. CURRENT INITIATIVES

INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
1. Reactor Oversight Process Status of the risk-informed reactor oversight process is now reported separately as an action plan under DQSR Attachment 1, page 37			
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) - IST - Issued Reg Guide 1.192 that endorses ASME O&M code cases including risk-informed code cases (obviates need to revise RG 1.175 and SRP 3.9.7). - Graded QA (RG 1.176 and GQA inspection guidance) - TS (RG 1.177 and SRP section 16.1)	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) 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			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
2. Risk-informed Licensing Actions (cont.)	- ISI (RG 1.178 and SRP section 3.9.8) Issued hundreds of risk-informed amendments over last few years		
3. Risk-informed technical specifications - Goal is to reflect safety significance of TS requirements **- <u>8 initiatives</u>:** 1. Modified end states 2. Missed surveillance 3. Flexible mode restraints 4. Risk-informed AOTs with a backstop	- Working with NEI and NSSS owners groups on concepts. Initiative 1: Safety evaluations (SE) written for CE and BWR topical reports on initiative 1. Initiatives 2 & 3: Approved and made available for licensee adoption by the CLIIP process. Initiative 4: NEI provided Risk Management Guide (RMG) (process guidance doc), CE pilot TSTF-424, and STP pilot.	- Working with NEI and NSSS owners groups on submittals. Initiative 1: Staff reviewed and provided feedback to industry on proposed TSTF-422, CE TS changes. Staff reviewing TSTF-423, BWR TS changes. Initiatives 2 & 3: Reviewing and approving LARs. Initiative 4: Initial feedback provided on RMG, CE pilot TSTF-424, and STP pilot.	- Working with NEI and NSSS owners groups on implementation. Initiative 1: NEI to respond to staff comments on TSTF-422. Staff to provide feedback on TSTF-423. Initiative 4: NEI to respond to staff comments on RMG, CE pilot TSTF-424, and STP pilot.

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
5. Optimize surveillance frequencies 6. Modify LCO 3.0.3 entry time 7. Inoperable non-TS support systems effect on TS system operability 8. Risk-inform the scope of the TS rule	Initiative 5: SR freq may be determined to be material to the 10 CFR 50.36 requirement for an SR (TS Program may not be an option). Initiative 6: RAIs discussed. Initiative 7: Staff provided comments on TSTF-372 Rev 3 on snubber inoperability, and on TSTF-427 on barrier inoperability. Initiative 8: Conceptual stage on relocating non-risk significant TS systems from TS; possible rulemaking.	Initiative 5: NEI developing white paper on SR freq 10 CFR 50.36 requirements and methodology. Initiative 6: Staff writing SE on CE topical report. Initiative 7: NEI to: Submit TSTF-372, Rev. 4, on snubbers; respond to staff comments on TSTF-427 on barriers. Initiative 8: NEI to develop white paper on guidance and methodology.	Initiative 5: NEI to submit white paper on SR freq 10 CFR 50.36 requirements and methodology. Initiative 6: Staff to complete SE on CE topical report. Initiative 7: Resolve TSTF-372 and TSTF-427 issues. Initiative 8: NEI to submit white paper on guidance and methodology.
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 Commission SRM 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 preparation of a final rulemaking package 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			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
4. Fire protection (cont.)	Circuit Analysis Resolution Program (CARP) - Staff has issued a draft RIS to explain the new Risk-Informed Inspection guidance. Public comments were received and the staff is currently reviewing them for the final issue of the RIS. - In early September, NRR held a working group meeting with the Regions to discuss resuming Associated Circuit Inspections. The Regional inspectors provided invaluable feedback and suggestions on the issue.	- The staff will issue a Draft NUREG for public comment that 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. The availability of this NUREG is expected to be noticed shortly in the *Federal Register*. - The staff is preparing a SECY to address enforcement discretion options regarding associated circuit findings.	- The staff is working on setting up a public meeting to discuss the risk-informed associated circuit inspection guidance. The workshop is expected to be held this winter. - Upon completion of meeting, the staff will put a plan in effect to withdraw EGM-98-02 and resume inspection in this area.
5. Safeguards NOTE: This effort is now the responsibility of the Office of Nuclear Security and Incident Response	- Proposed revisions to 10 CFR 73.55 sent to Commission 6/4/01. Proposal requires that licensees' security programs employ risk insights in identifying target sets of equipment necessary to prevent core damage and/or spent fuel sabotage and create a more performance oriented basis for security regulations. - Proposed 73.55 returned by Commission to staff for rework to reflect lessons learned from September 11, 2001, events.	- Subsumed by staff efforts on post-September 11, 2001, Response to Terrorist Activities.	- Subsumed by staff efforts on post-September 11, 2001, Response to Terrorist Activities.

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
6. 10 CFR 50.69 rulemaking - risk-informing scope of special treatment requirements	- Pilot plants completed IDP review of categorization, with staff observation - Draft rule language made available for public comment on NRC web site. (Notice of Availability published in November 29, 2001, *Federal Register*); revised drafts posted April 5 and August 2, 2002 - Proposed rule package sent to Commission in paper dated September 30, 2002 - On March 28, 2003, Commission approved publishing proposed rule for 75 day comment period. - Proposed rule was published in the *Federal Register* on May 16, 2003.	Comment period extended for 30 days and closed on August 30, 2003. Staff is now evaluating the public comments.	- Complete review of industry guidance documents (NEI 00-04) and finalize implementation of Reg Guide (DG-1122) - Review proposed rule comments and revise final rule accordingly. - Conduct WOG pilot program focused on 50.69 submittal and reflect lessons learned in staff review guidance and potentially final rulemaking package. - Publish 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 completed - SECY-03-0127 (July 24, 2003) provided final rule to Commission. - In SRM of August 28, 2003, Commission approved issuance of final rule. - Final rule published September 16, 2003 (68 FR 54123), effective on October 16, 2003.	Preparing 50.44 Rulemaking Regulatory History. Preparing Regulatory Guide for publication. Preparing revised SRP Section 6.2.5.	None.
- 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 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**
7. RIP50/Option 3 (risk-informing technical requirements) (cont.)	- Staff met with BWROG to discuss their “safety case” approach for risk-informing requirements related to LOCA-LOOP		
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

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
9. Creating a risk-informed environment	- Three (3) NRR all employee division meetings held to brief staff on results of current environment assessment - Technical seminars conducted monthly. - Newsletter on risk-informed activities implemented. - Contractor report on pilot activities drafted and submitted for review.	- Contractor report on pilot activities under review	- Develop office-wide implementation plan based on results of pilot 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.

A. CURRENT INITIATIVES			
INITIATIVE	**RECENT ACTIVITIES**	**CURRENT ACTIVITIES**	**FUTURE ACTIVITIES**
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	
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	- None.	On hold pending availability of resources.
14. Risk-Informed Regulation Implementation Plan (RIRIP)	- Previous update published March 21, 2003 (SECY-03-044)	- Latest update published October 27, 2003 (SECY-03-0181)	- 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.