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ADAMS ACCESSION NUMBER: ML020150515

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ADAMS DOCUMENT TITLE: Public Version of January 2002 Director's Quarterly Status Report

DOCUMENT NAME: DIST.WPD

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INTRODUCTION

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

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

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

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

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

ATTACHMENT 1

NRR ACTION PLANS

TABLE OF CONTENTS

DE

BOILING WATER REACTOR INTERNALS	1
STEAM GENERATORS	5
OKONITE CABLE LOCA TEST FAILURES	23

DIPM

EMERGENCY ACTION LEVEL GUIDANCE DEVELOPMENT	26
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DSSA

ECCS SUCTION BLOCKAGE	29
CONTROL ROOM HABITABILITY	38

BOILING WATER REACTOR INTERNALS

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

Last Update: 01/03/02
 Lead NRR Division: DE
 Supporting Division: DSSA
 GSI: Not Available

MILESTONES	DATE (T/C) ¹
PART I: REVIEW OF GENERIC INSPECTION AND EVALUATION CRITERIA	
1. Issue summary NUREG-1544	03/96 (C)
o Update NUREG-1544	3Q/02 (T)
2. Review BWRVIP Re-inspection and Evaluation Criteria	
o Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03)	07/15/99 (CA)
o BWRVIP-03, Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components	07/15/99 (CA)
o BWR Vessel Shell Weld Inspection Recommendations (BWRVIP-05)	07/28/98 (CA)
o BWR Axial Shell Weld Inspection Recommendations	03/07/00 (CA)
o Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07)	04/27/98 (CA)
3. Review of generic repair technology, criteria, and guidance	TBD
4. Review generic mitigation guidelines and criteria	TBD
5. Review of generic NDE technologies developed for examinations of BWR internal components and attachments	TBD
6. Other Internals reviews (safety assessments, evaluations, mitigation measures, inspections, and repairs)	
o Safety Assessment of BWR Reactor Internals (BWRVIP-06)	09/15/98 (CA)
o Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08 & BWRVIP-46)	03/27/98 (CA)
o Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14)	12/03/99 (CA)
o Internal Core Spray Piping and Sparger Replacement Design Criteria (BWRVIP-16)	08/10/00 (CA)
o Roll/Expansion of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)	03/13/98 (CD)
o BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)	12/02/99 (CA)
o BWRVIP-18, Appendix C, BWR Core Spray Internals Demonstration of Compliance With Technical Information Requirements of License Renewal Rule (10 CFR 54.21)	09/06/00 (CA)
o Internal Core Spray Piping and Sparger Repair Design Criteria (BWRVIP-19)	08/10/00 (CA)
o Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25)	12/19/99 (CA)
o Top Guide Inspection and Flaw Evaluation Guideline (BWRVIP-26)	09/29/99 (CA)
o Standby Liquid Control System / Core Plate ΔP Inspection and Flaw Evaluation Guidelines (BWRVIP-27)	04/27/99 (CA)
o Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)	04/10/00 (CA)
o Technical Basis for Part Circumferential Weld Overlay Repair of Vessel Internal Core Spray Piping (BWRVIP-34)	05/31/02 (T)

MILESTONES	DATE (T/C) ¹
○ Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)	.. 07/24/00 (CA)
○ BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)	.. 07/24/00 (CA)
○ BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42)	.. 05/26/00 (CA)
○ Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)	.. 05/26/00 (CA)
○ BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47)	.. 03/27/98 (CA)
○ Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (BWRVIP-48)	.. 10/13/99 (CA)
○ Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49)	.. 09/29/99 (CA)
○ Top Guide / Core Plate Repair Design Criteria (BWRVIP-50)	... 01/29/01 (CI)
○ Jet Pump Repair Design Criteria (BWRVIP-51)	... 10/28/00 (CI)
○ Shroud Support and Vessel Repair Design Criteria (BWRVIP-52)	... 11/02/00 (CI)
○ Standby Liquid Control Line Repair Design Criteria (BWRVIP-53)	... 10/26/00 (CI)
○ Lower Plenum Repair Design Criteria (BWRVIP-55)	... 09/28/01 (CI)
○ LPCI Coupling Repair Design Criteria (BWRVIP-56)	... 03/01/02 (T)
○ Instrument Penetrations Repair Design Criteria (BWRVIP-57)	... 03/01/02 (T)
○ CRD Internal Access Weld Repair (BWRVIP-58)	... 10/17/01 (CI)
○ Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals (BWRVIP-59)	... 07/31/01 (CI)
○ BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants (BWRVIP-60)	.. 07/08/99 (CA)
○ Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)	... 01/30/01 (CI)
○ Shroud Vertical Weld Inspection and Evaluation Guidelines (BWRVIP-63)	... 04/18/00 (CI)
○ BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)	.. 07/27/01 (CA)
○ Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)	... 09/15/00 (CI)
○ BWR Core Shroud Inspection & Flaw Evaluation Guidelines (BWRVIP-76)	... 12/31/02 (T)
○ BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material (BWRVIP-78)	... 03/01/02 (T)
○ Evaluation of Crack Growth in BWR Shroud Vertical Welds (BWRVIP-80)	... 12/31/02 (T)

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

Description: Many components inside boiling water reactor (BWR) vessels (i.e., internals) are made of materials such as stainless steel and various alloys that are susceptible to corrosion and cracking. This degradation can be accelerated by stresses from temperature and pressure changes, chemical interactions, irradiation, and other corrosive environments. This action plan is intended to encompass the evaluation and resolution of issues associated with intergranular stress corrosion cracking (IGSCC) in BWR internals. This includes plant specific reviews and the assessment of the generic criteria that have been proposed by the BWR Owners Group and the BWRVIP technical subcommittees to address IGSCC in core shrouds and other BWR internals.

Historical Background: Significant cracking of the core shroud was first observed at Brunswick, Unit 1 nuclear power plant in September 1993. The NRC notified licensees of Brunswick's discovery of

significant circumferential cracking of the core shroud welds. In 1994, core shroud cracking continued to be the most significant of reported internals cracking. In July 1994, the NRC issued Generic Letter (GL) 94-03 which requires licensees to inspect their shrouds and provide an analysis justifying continued operation until inspections can be completed.

A special industry review group (Boiling Water Reactor Vessels and Internals Project - BWRVIP) was formed to focus on resolution of reactor vessel and internals degradation. This group was instrumental in facilitating licensee responses to NRC's GL 94-03. The NRC evaluated the review group's reports, submitted in 1994 and early 1995, and all plant specific responses.

All of the plants evaluated were able to demonstrate continued safe operation until inspection or repair on the basis of: 1) no 360° through-wall cracking observed to date, 2) low frequency of pipe breaks, and 3) short period of operation (2-6 months) before all of the highly susceptible plants complete repairs of or inspections to their core shrouds.

In late 1994, extensive cracking was discovered in the top guide and core plate rings of a foreign reactor. The design is similar to General Electric (GE) reactors in the U.S., however, there have been no observations of such cracking in U.S. plants. GE concluded that it was reasonable to expect that the ring cracking could occur in GE BWRs with operating time greater than 13 years. In the special industry review group's report, that was issued in January 1995, ring cracking was evaluated. The NRC concluded that the BWRVIP's assessment was acceptable and that top guide ring and core plate ring cracking is not a short term safety issue.

Proposed Actions: The staff has been interacting with the BWRVIP and individual licensees. In an effort to lower the number of industry and staff resources that will be needed in the future, it is important for the staff to continue interacting with the industry on a generic basis in order to encourage them to continue their proactive efforts to resolve IGSCC of BWR internals as a voluntary industry initiative. The BWRVIP has submitted over 50 generic documents, supporting plant-specific submittals, for staff review. The staff is ensuring that the generic reviews are incorporating recent operating experience on all BWR internals.

Originating Document: Generic Letter 94-03, issued July 25, 1994, which requested BWR licensees to inspect their core shrouds by the next outage and to justify continued safe operation until inspections can be completed.

Regulatory Assessment: In July 1994, the NRC issued Generic Letter 94-03 which required licensees to inspect their shrouds and provide an analysis justifying continued operation until inspections could be performed. The staff has concluded in all cases that licensees have provided sufficient evidence to support continued operation of their BWR units to the refueling outages in which shroud inspections or repairs have been scheduled. In addition, in October 1995, industry's special review group submitted a safety assessment of postulated cracking in all BWR reactor internals and attachments to assure continuing safe operation.

Current Status: Almost all BWRs completed inspections or repairs of core shrouds during refueling outages in the fall of 1995. Various repair methods have been used to provide alternate load carrying capability, including preemptive repairs, installation of a series of clamps and use of a series of tie-rod assemblies. The NRC has reviewed and approved all shroud modification proposals that have been submitted by BWR licensees. Review by NRC continues on individual plant reinspection results and plant-specific assessments.

The BWRVIP has submitted Appendices to the Inspection and Flaw Evaluation Guidelines. These appendices address the use of BWRVIP generic inspection guidelines for compliance with requirements of the license renewal rule (10 CFR Part 54). The staff is reviewing these appendices in conjunction

with its review of the BWRVIP guidelines, and has issued the first several of thirteen license renewal SEs on BWR internals, with the remaining expected to be completed by February 2002. The schedule change for BWRVIP-76 is due to the staff waiting for the BWRVIP to supplement its original submittal in accordance with the open items in the staff's initial SE.

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

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

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

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

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

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

Action Plan dated April 1995.

STEAM GENERATORS

<p><u>TAC Nos.</u> M88885 M99432 MA4265 MA5037 MA5260 MA7147 MA9881 MB0258 MB0553 MB0576 MB0631 MB0633 MB0737</p>	<p><u>Description</u> Steam Generator (SG) Integrity Rulemaking GL: SG Tube Integrity NEI 97-06 SG Action Plan DPO on SG Issues GSI-163 Regulatory Issue Summary - IP2 SG Tube Failure SG Action Plan Administration SG Inspection Program Licensee SG Inspection Results Summary Reports & SG Tube Integrity Amendment Review Guidance SG Workshop OL No. 803 Revisions per SG Action Plan IIPB SG Action Plan Activities</p>	<p>Last Update: 12/31/01 Lead Division: DLPM Supporting Divisions: DE, DIPM, DSSA Supporting Office: RES</p>
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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)	<p>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:</p> <p>a) review and revise the baseline inspection program.</p> <p>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</p> <p>b.2) develop and issue draft revision of risk-informed SDP using information identified in b.1 above</p> <p>c) review and revise the training program for inspectors</p> <p>c.1) Provide IP training material to Regions</p> <p>c.2) Formal training to inspectors</p> <p>(Attachment 3 to Ref. 1; TG: 5a, 5b, 5c, 5d, 5f, 6c)</p>	<p>04/30/01 (C) ML011210293</p> <p>09/21/01 (C) ML012680252</p> <p>02/28/02 (T)</p> <p>10/11/01 (C) ML012970361</p> <p>02/01/02 (T)</p>	<p>DE C. Khan</p> <p>DSSA S. Long</p> <p>DIPM P. Koltay</p> <p>DIPM E. Kleeh</p>	<p>DIPM DSSA S. Long</p> <p>DE C. Khan DIPM P. Koltay</p> <p>DSSA S. Long DE C. Khan</p> <p>DE C. Khan</p>
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	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.14 (MA4265)	Staff completes review and prepares draft safety evaluation of NEI 97-06 including addressing issues raised in OIG report and IP2 lessons learned report (NEI 97-06, TG: 2, 3, 4, 7)	08/31/02 (T)	DE E. Murphy	
1.15 (MB0631)	Hold steam generator workshop with stakeholders (page 2 of Ref. 1; page 2 of Ref. 2)	02/27/01 (C) ML010820457	DE R. Rothman	
1.16 (MA4265)	Staff briefs CRGR on NEI 97-06 (NEI 97-06)	10/31/02 (T)	DE E. Murphy	
1.17 (MA4265)	Publish SE on NEI 97-06 in FR for public comment (NEI 97-06)	10/31/02 (T)	DLPM M. Banerjee	
1.18 (MA4265)	ACRS review of NEI 97-06 (NEI 97-06)	10/31/02 (T)	DE E. Murphy	
1.19 (Later)	Issue generic communication related to steam generator operating experience and status of steam generator issues	10/31/01 (C)	DE Z. Fu	
1.20 (MA4265)	Staff briefs Commission on endorsing NEI 97-06 (NEI 97-06, and WITS Item 199400048)	12/31/02 (T)	DE L. Lund	
1.21 (MA4265)	Staff issues endorsement package on NEI 97-06 in a safety evaluation and includes the approval of the generic technical specification change in a Regulatory Issue Summary	01/31/03 (T)	DE E. Murphy	
2.1	Evaluate the need for a new communication protocol with the U.S. Secret Service that would cover emergency situations at all NRC licensed facilities (Attachment 3 of Ref. 1)	12/05/00 (C) ML010460485 ML010820457	IRO F. Congel	
2.2 (MB0258)	Establish NRC web site for Steam Generator Action Plan	01/16/01 (C) ML010820457	DLPM R. Ennis	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
2.3 (MB0258)	Review and revise, as appropriate, the policy for project manager involvement with the morning call between the resident inspectors and the region. (Attachments 3 and 4 of Ref. 1)	03/23/01 (C) ML011020026	DLPM R. Ennis	
2.4 (MB0737)	Review program requirements for routine communications between the resident inspectors and local officials based on public interest. Based on weighing current resident inspector responsibilities (e.g., inspection requirements, following up on plant events) against this review, revise program requirements if needed. (Attachment 3 of Ref. 1)	04/03/01 (C) ML010890426	DIPM T. D'Angelo	
2.5 (MB0737)	Develop, revise, and implement, as appropriate, a process for the timely dissemination of technical information to inspectors for inclusion in the inspection program (TG: 5g)	04/03/01 (C) ML010890426	DIPM G. Klinger	
2.6 (MB0258)	<p>Incorporate experience gained from the IP2 event and the SDP process into planned initiatives on risk communication and outreach to the public (TG: 9)</p> <ol style="list-style-type: none"> 1. Issue NRR input for incorporation into OEDO initiative 2. Address SRM dated 12/26/01 	<p>02/28/02 (T)</p> <p>TBD</p>	<p>PMAS M. Kotzalas</p> <p>TBD</p>	
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	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
2.8 (MB0633)	<p>Review and revise, as appropriate, the amendment review process, including concurrence responsibilities, supervisory oversight, and second-round requests for additional information.</p> <p>a. Issue OI LIC-101</p> <p>b. Issue procedure for NRR and RES interactions (Attachment 3 of Ref. 1; TG: 6b, 6d, 6e; page 6 of Ref. 1)</p>	<p>8/31/01 (C)</p> <p>02/28/02 (T)</p>	<p>DLPM M. Banerjee DLPM M. Fields</p>	
3.1	<p>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)</p> <p>Specific tasks include:</p> <p>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.</p>	<p>12/31/02 (T)</p>	<p>RES J. Uhle</p>	<p>DSSA W. Jensen</p>

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

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.1 (continued)	<p>i) Conduct tests of degraded tubes under pressure and with axial and bending loads to validate the analytical results from above tasks.</p> <p>j) Conduct analyses similar to above with refined load estimates if necessary.</p> <p>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).</p>	<p>06/30/03 (T)</p> <p>06/30/04 (T)</p> <p>02/28/05 (T)</p>	<p>RES J. Muscara</p> <p>RES J. Muscara</p> <p>DSSA S. Long</p>	<p>DE E. Murphy</p> <p>DE E. Murphy</p> <p>DE E. Murphy RES J. Muscara E. Thornbury</p>
3.2	<p>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)</p> <p>Specific tasks include:</p> <p>a) Complete tests of jet impingement under MSLB conditions.</p> <p>b) Conduct long duration tests of jet impingement under severe accident conditions.</p> <p>c) Document results from tasks 3.2a and 3.2b.</p>	<p>12/31/01 (C)</p> <p>12/31/01 (C)</p> <p>12/31/01 (C)</p>	<p>RES J. Muscara</p> <p>RES J. Muscara</p> <p>RES J. Muscara</p>	<p>DE E. Murphy</p> <p>DE E. Murphy</p> <p>DE E. Murphy</p>
3.3	<p>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)</p>	<p>09/30/04 (T)</p> <p>See Note 2</p>	<p>RES R. Lee</p>	<p>DSSA S. Long</p>

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4	<p>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)</p> <p>Specific tasks include:</p> <p>a) Perform system level analyses to assess the impact of plant sequence variations (e.g., pump seal leakage and SG tube leakage).</p> <p>b) Re-evaluate existing system level code assumptions and simplifications.</p> <p>c) Examine 1/7 scale data to assess tube to tube temperature variations and estimate variations for plant scale.</p> <p>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.</p> <p>e) Examine SG tube severe accident T-H conditions using computational fluid dynamics (CFD) methods. This includes the following:</p> <p>e.1) Benchmark CFD methods against 1/7 scale test data.</p>	<p>09/28/01 (C) ML012720004</p> <p>03/31/02 (T)</p> <p>08/31/02 (T)</p> <p>12/31/02 (T)</p> <p>08/31/01 (C) ML012750061</p>	<p>RES C. Tinkler</p> <p>RES C. Tinkler</p> <p>RES C. Tinkler</p> <p>RES C. Tinkler</p> <p>RES C. Boyd</p>	<p>DSSA W. Jensen S. Long</p>

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	e.2) Perform full scale plant calculations (hot leg and SG) for a 4 loop Westinghouse design. Evaluate scale effects.	03/31/02 (T)	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).	07/31/02 (T)	RES C. Boyd	DSSA W. Jensen S. Long
	f) Examine the uncertainty in the T-H conditions associated with core melt progression.	01/31/03 (T)	RES C. Tinkler	DSSA W. Jensen S. Long
	g) Perform experiments to develop data on inlet plenum mixing impacts due to SG tube leakage and hot leg/inlet plenum configuration.	03/31/03 (T)	RES C. Tinkler	DSSA W. Jensen S. Long
	h) Perform a systematic examination of the alternate vulnerable locations in the RCS that are subject to failure due to severe accident conditions. This includes the following:			
	h.1) Evaluate the creep failure of primary system passive components such as pressurizer surge line and the hot leg taking into account the material properties of the base metal, welds, and heat affected zones in the presence of residual and applied stresses, in addition to the pressure stress, and the presence of flaws.	11/30/03 (T)	RES J. Muscara	DE E. Murphy DSSA S. Long
	h.2) Evaluate the failure of active components such as PORVs, safety valves, and bolted seals based on operability and "weakest link" considerations for these components.	11/30/03 (T)	RES J. Muscara	DE E. Murphy DSSA S. Long
h.3) Conduct large scale tests if needed.	11/30/05 (T)	RES J. Muscara	DE E. Murphy DSSA S. Long	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	<p>i) Develop data and analyses for predicting leak rates for degraded tubes in restricted areas under design basis and severe accident conditions.</p> <p>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).</p> <p>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).</p>	<p>12/31/03 (T)</p> <p>06/30/04 (T)</p> <p>02/28/05 (T)</p>	<p>RES J. Muscara</p> <p>DSSA S. Long</p> <p>DSSA S. Long</p>	<p>DSSA S. Long DE E. Murphy</p> <p>DE E. Murphy RES J. Muscara</p> <p>DE E. Murphy RES J. Muscara C. Tinkler E. Thornbury</p>
3.5	<p>Develop improved methods for assessing the risk associated with SG tubes under accident conditions. (Pgs. 47, 16-20) (See Note 5)</p> <p>Specific tasks include:</p> <p>a) Development of an integrated framework for assessing the risk for the high-temperature/high-pressure accident scenarios of interest.</p>	03/29/02 (T)	RES E. Thornbury	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.7	Assess the need for better leakage correlations as a function of voltage for 7/8" SG tubes. (Pgs. 48, 28-29) (See Note 5)	04/30/03 (T)	DE E. Murphy	RES J. Muscara
3.8	Develop a program to monitor the prediction of flaw growth for systematic deviations from expectations. (Pg. 48) (See Note 5)	1/3/02 (C)	DE J. Tsao	
3.9	<p>Develop a more technically defensible position on the treatment of radionuclide release to be used in the safety analyses of design basis events. (Pgs. 48, 38-44) (See Note 5)</p> <p>Specific tasks include:</p> <p>a) Assess Adams and Atwood and Adams and Sattison spiking data with respect to the ACRS comments.</p> <p>b) Based upon the assessment performed in task 3.9a, develop a response to the ACRS comments.</p> <p>c) Publish in the <i>Federal Register</i> for public comment, the response to ACRS' comments.</p> <p>d) Complete review of public comments.</p> <p>e) Based upon task 3.9d, determine if additional work needs to be performed.</p>	<p>08/09/01 (C)</p> <p>02/28/02 (T)</p> <p>04/30/02 (T)</p> <p>10/31/02 (T)</p> <p>08/15/02 (T)</p>	DSSA J. Hayes	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.10	<p>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)</p> <p>Specific tasks include:</p> <p>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.</p> <p>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.</p> <p>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.</p>	<p>12/31/05 (T)</p> <p>12/31/06 (T)</p> <p>12/31/05 (T)</p>	<p>RES J. Muscara</p> <p>RES J. Muscara</p> <p>RES J. Muscara</p>	<p>DE E. Murphy</p> <p>DE E. Murphy</p> <p>DE E. Murphy</p>

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.11	In order to resolve GSI 163, it is necessary to complete the work associated with tasks 3.1 through 3.5 and 3.7 through 3.9. Upon completion of those tasks, develop detailed milestones associated with preparing a GSI resolution document and obtaining the necessary approvals for closing the GSI, including ACRS acceptance of the resolution. (See note 9)	12/31/05 (T)	DLPM J. Zimmerman	DE E. Murphy DSSA S. Long
3.12	Develop outline and a detailed schedule for completing DG 1073, "Plant Specific Risk-Informed Decision Making: Induced SG Tube Rupture (See note 9)	12/31/05 (T)	DE E. Murphy	DSSA S. Long

Notes:

1. For SG Action Plan milestones associated with the SG DPO (i.e., Item Nos. 3.1 - 3.11), the page numbers referenced in the milestone description indicate the source of the milestone as described in ACRS Report NUREG-1740, "Voltage-Based Alternative Repair Criteria." The ACRS report was included as an enclosure to a memorandum from D. Powers to W. Travers dated February 1, 2001 (Accession No. ML010780125).
2. With respect to milestone Item No. 3.3, the ARTIST program plan is being finalized for implementation. A firm testing schedule is not currently available but testing is expected to commence in 2002.
3. The work described in this milestone is related, in part, to previously planned work associated with an NRR User Need request dated February 8, 2000 (Accession No. ML003682135), and the associated RES response to the request dated September 7, 2000 (Accession No. ML003714399). In addition, portions of this work were undertaken on an anticipatory basis by RES.
4. The work described in this milestone is related, in part, to previously planned work associated with GSI 188, "Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass."
5. The work described in this milestone is related, in part, to previously planned work associated with GSI 163, "Multiple Steam Generator Tube Leakage."
6. The thermal-hydraulic analyses (items 3.1a through 3.1c) will provide input into the tube integrity analyses (items 3.1d through 3.1j) on an on-going basis. The end dates for these two areas coincide because of the close integration between these two RES efforts. Also, the end dates reflect the target date for the final report documenting the RES findings.

7. Item Nos. 1.1 through 2.8 in the above table were developed from Attachment 1 of a memorandum from J. Zwolinski, J. Strosnider, B. Boger and G. Holahan to B. Sheron and R. Borchardt dated March 23, 2001 (Accession No. ML010820457). That memorandum provided a revision to the Steam Generator Action Plan that was originally issued via a memorandum from B. Sheron and J. Johnson to S. Collins dated November 16, 2000 (Accession No. ML003770259).
8. Item Nos. 3.1 through 3.11 in the above table were developed from Attachment 1 of a memorandum from S. Collins and A. Thadani to W. Travers dated May 11, 2001 (Accession No. ML011300073). That memorandum provided a revision to the Steam Generator Action Plan as requested by a memorandum from W. Travers to S. Collins and A. Thadani dated March 5, 2001 (Accession No. ML010670217).
9. The completion date assumes need for large scale test.
10. The ADAMS accession no. listed under "Date" is the closure document.

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

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

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

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

technical requirements manual section. SECY-00-0078 outlines the staff's proposed review process associated with the revised steam generator tube integrity regulatory framework described in NEI 97-06.

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

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

Current Status:

- November 1, 2000 Issuance of "Indian Point 2 Steam Generator Tube Failure Lessons-Learned Report" via memorandum from W. Travers to the Commission (Accession No. ML003765272).
- November 3, 2000 Issuance of "Staff Review of OIG Report on the NRC's Response to the Steam Generator Tube Failure at Indian Point 2 and Related Issues" via memorandum from W. Travers to the Commission (Accession No. ML003753067).
- November 16, 2000 Issuance of "Steam Generator Action Plan" via memorandum from B. Sheron/J. Johnson to S. Collins (Accession No. ML003770259).
- February 1, 2001 ACRS Ad Hoc Subcommittee report related to SG DPO issued (NUREG-1740).
- May 11, 2001 Issuance of a memorandum providing a revision to the SG Action Plan to address the issues related to the DPO on SG tube integrity issues (Accession No. ML011300073).
- August 2, 2001 Issuance of a letter to NEI transmitting a draft NRC paper on NEI 97-06 SG generic change package (Accession No. ML012200349).
- August 29, 2001 Public meeting between NRC and NEI to discuss revisions to the proposed regulatory framework in NEI 97-06 (meeting summary: Accession No. ML012690666).
- September 18, 2001 Issuance of a memorandum with staff comments on SG inspection intervals proposed by the industry in NEI 97-06 (Accession No. ML012610664).
- September 21, 2001 Issuance of memorandum documenting completion of Item Nos 1.11.b.1 (Accession No. ML012680252)
- 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.
- September 28, 2001 Issuance of memorandum documenting completion of Item Nos 3.4.a (Accession No. ML012750061).
- October 4, 2001 Staff briefing of ACRS full-committee on SG action plan status.

- October 18, 2001 ACRS letter to the Chairman documenting their comment on staff action plan to address the SG DPO (ML012960166).
- November 28, 2001 Public meeting between NRC and NEI management to discuss NEI 97-06 and TMI tube severance issues.
- November 29, 2001 Staff briefing of ACRS Subcommittee on Materials and Metallurgy on NEI 97-06.
- December 3, 2001 Staff briefing of the Commission on the status of SG action plan.
- December 06, 2001 Staff briefing of ACRS on NEI 97-06.

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OKONITE CABLE LOCA TEST FAILURES

TAC Nos. MA8193, MA9199, MA9200, & MA9201

Last Update: 01/08/02

Lead Division: DE

MILESTONES	DATE (T/C)
1. Meet with Okonite to discuss LOCA test #5 cable failure results	02/08/00 (C)
2. Meet with nuclear industry to discuss LOCA test #5 cable failure results	02/16/00 (C)
3. Issue letter to Okonite with BNL test report	05/17/00 (C)
4. Issue letter to NEI with BNL test report	05/18/00 (C)
5. Meet with NEI and Okonite to discuss impact on operating reactors and responses being considered by NRC and industry	06/22/00 (C)
6. Based on the 10/12 meeting with industry and Okonite to discuss the results of the NEI survey, staff will determine if any of the following regulatory actions are warranted:	
a. If a small number of plants are affected, they will be addressed individually.	05/30/02
b. If industry sufficiently addresses the issues and several plants are affected, the staff will publish a Regulatory Issue Summary in accordance with SECY-99-143.	05/30/02
c. If the industry initiative is inadequate, the staff will issue a generic letter to licensees to obtain information on affected safety-related equipment and plants.	05/30/02

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

Historical Background: In related past research, Sandia National Laboratories, under contract to the NRC, performed tests on the same Okonite cable, along with several other cables. The results of this testing are described in NUREG/CR-5772, "Aging, Condition Monitoring, and Loss-of-Coolant Accident (LOCA) Tests of Class 1E Electrical Cables," Volumes 1, 2, and 3. In that program, one of the cable

types that failed during the accident tests was the Okonite/Okalon single-conductor cable. A similar failure mechanism was found, namely splitting and opening of the jacket. On the basis of these findings, the NRC issued Information Notice 92-81, "Potential Deficiency of Electrical Cables With Bonded Hypalon Jackets," to alert licensees to a potential deficiency in the environmental qualification of electrical cables with bonded jackets. RES was doing additional testing on this and other cable types as part of GSI-168.

Proposed Actions: The action plan is divided into three parallel efforts. Once we get feedback from Okonite and the industry we will determine if any regulatory action is warranted. There are three potential courses of action we may pursue once we have responses from the vendor and the industry:

- (1) If only a small number of safety-related equipment items are affected, or only a small number of plants are affected, the staff may address these cases individually.
- (2) If the industry initiative sufficiently addresses the issue and several plants are affected, the staff will publish a Regulatory Issue Summary to document the resolution of the issue in accordance with SECY-99-143, "Revisions to Generic Communication Program."
- (3) If the industry initiative is inadequate, the staff may issue a generic letter to nuclear power plant licensees to obtain information on the affected safety-related equipment and plants.

Originating Document: Memorandum from Brian Sheron to Samuel Collins dated May 9, 2000, informing Mr. Collins of the action plan to address the LOCA test failures of Okonite single-conductor bonded jacket cables based on the May 2, 2000, memorandum from Ashok Thadani to Samuel Collins.

Regulatory Assessment: The NRR staff is continuing to work with the vendor, industry, and RES to determine if any regulatory action is warranted. Based on industry statements in previous meetings related to the application and limited use of the subject cable, the staff believes that continued operation of nuclear power plants is warranted while it evaluates the potential deficiency of these cables.

The Code of Federal Regulations (10 CFR 50.49) requires that each item of electric equipment important to safety is qualified for its application, and meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

The staff believes that there is sufficient new information and concerns relative to the operability of Okonite single-conductor bonded jacket cable under design basis conditions to warrant the actions outlined in the action plan dated May 9, 2000.

Current Status: The staff conducted meetings with representatives from Okonite and industry on February 8, and 16, 2000, respectively. By letters dated May 17 and 18, 2000, the staff requested Okonite to evaluate the BNL test report to determine if the test failures represent a deviation or a failure to comply with 10 CFR 21 and, NEI to schedule a meeting to discuss possible options for addressing the issue. At the June 22, 2000, meeting, NEI committed to conduct a survey of all nuclear power plants. The results of the NEI survey were presented to the staff in a meeting on October 12, 2000. NRC is waiting for a response from NEI on the February 7, 2001, letter to NEI. By letter dated July 26, 2001, Okonite provided the staff with the test protocol for EQ testing of Okonite Okalon cables. The EQ test at Wyle Laboratories, including the test results, were provided to the staff from Okonite by letter dated December 20, 2001. The staff is currently evaluating the test results and will issue a final RIS or, take appropriate action as required.

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

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

References:

1. Memorandum from Jack Strosnider to Brian Sheron, January 21, 2000.
2. Memorandum from Ashok Thadani to Samuel Collins, May 2, 2000.
3. Memorandum from Brian Sheron to Samuel Collins, May 9, 2000.
4. Letter from Samuel Collins to Okonite, May 17, 2000.
5. Letter from Samuel Collins to NEI, May 18, 2000.
6. Letter Report from BNL on LOCA Test #5, March 26, 2000.
7. Minutes of NRC Meeting on February 8, 2000, with Okonite.
8. Minutes of NRC Public Meeting on February 16, 2000.
9. Minutes of NRC Public Meeting on June 22, 2000.
10. Minutes of NRC public meeting on October 12, 2000.
11. NRC Regulatory Issue Summary 2000-25, December 26, 2000.
12. Letter from Jack Strosnider to NEI, February 7, 2001.
13. Letter from Okonite to Samuel Collins, May 2, 2001.
14. Letter from NEI to Jack Strosnider, July 17, 2001.
15. Letter from Okonite to Samuel Collins, July 26, 2001.
16. Letter from Jack Strosnider to Okonite, August 23, 2001.
17. Letter from Okonite to Samuel Collins, December 20, 2001.

EMERGENCY ACTION LEVEL GUIDANCE DEVELOPMENT

TAC No.: MA3695
M98020

Revision to NESP-007
Shutdown EAL Guidance

Last Update: 12/31/01
Lead NRR Division: DIPM

EAL GUIDANCE FOR COLD SHUTDOWN, REFUELING AND LONG TERM FUEL STORAGE (“SHUTDOWN EAL GUIDANCE” NEI-99-01)

MILESTONES	DATE (T/C)
1. Meet with NEI to resolve staff concerns on NEI’s guidance (proposed in NEI-97-03) for EALs applicable in the shutdown mode of operation	01/28/99 (C)
2. NEI to provide new shutdown EAL guidance (NEI-99-01) for NRC review	04/07/99 (C)
3. NRC provides comments to NEI on NEI-99-01	05/11/99 (C)
4. Meet with NEI to discuss comments	05/13/99 (C)
5. Comments resolved and final draft of NEI-99-01 submitted for endorsement	07/99 (C)
6. Draft guide developed endorsing NEI-99-01 developed in form of a draft guide for CRGR/ACRS review.	03/06/00 (C)
7. Determination made on whether to issue a Generic Letter on plant-specific implementation of shutdown EALs - no GL to be issued	08/30/00 (C)
8. CRGR/ACRS meeting on generic letter - canceled	08/30/00 (C)
9. Draft Guide issued for public comment	03/22/00 (C)
10. Public comments addressed (NEI-99-01 revised as needed)	07/14/00 (C)
11. CRGR/ACRS meeting on final guide NEI 99-01 (meeting waived)	11/01/00 (C)
12. Regulatory Guide issued (On hold due to spent fuel pool study impact)	TBD

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

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

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

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

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

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

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

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

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

References:

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

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

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ECCS SUCTION BLOCKAGE

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

Last Update: 1/01/02
Lead NRR Division: DSSA
Supporting Divisions: DE, DRCH, and DET (RES)
GSI: 191

MILESTONES	DATE (T/C)
PART I: BWR ECCS SUCTION STRAINER CLOGGING ISSUE	
1. NRCB 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors"	10/01 (C)
PART II: NPSH EVALUATIONS	
1. GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps" <ul style="list-style-type: none"> ○ Complete review of licensee responses ○ Complete revision of RG 1.1/RG 1.82 (DG-1107) 	03/00 (C) 9/02 (T)
PART III: CONTAINMENT COATINGS	
1. GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"	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 <ul style="list-style-type: none"> ○ 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 (including final risk assessment) ○ 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 (T)

MILESTONES		DATE (T/C)
2.	Resolve ECCS suction clogging issue for PWRs (Regulation/Guidance Development and Issuance Stages of GSI process in MD 6.4 (Stages 4 and 5))	
	○ Update ECCS Suction Clogging Action Plan to include resolution of the issue for PWRs	1/02 (T)
	○ Brief NRR ET to obtain approval to prepare a generic letter (GL)	2/02 (T)
	○ Public meeting with NEI, WOG, B&WOG, CEOG	3/02 (T)
	○ Proposed Draft GL to CRGR for review	5/02 (T)
	○ CRGR Briefing on proposed draft GL	6/02 (T)
	○ Proposed draft GL issued for Public Comment	7/02 (T)
	○ Public meeting with NEI, WOG, B&WOG, CEOG during Public Comment period	8/02 (T)
	○ Public Comment period ends	9/02 (T)
	○ Resolution of Public Comments and revisions to proposed GL made, as necessary	10/02 (T)
	○ CRGR Briefing on proposed final GL	11/02 (T)
	○ ACRS Briefing on proposed final GL	12/02 (T)
	○ Information Paper sent to Commission, issue GL	12/02 (T)

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

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

determination that sump blockage will impede or prevent long-term recirculation at a specific plant. By memorandum dated September 28, 2001, RES transferred the lead for GSI-191 to NRR consistent with Management Directive 6.4. The parametric evaluation forms the basis for concluding the Technical Assessment phase of the GSI. RES also recommended in the memorandum that plant-specific analyses be performed by licensees to determine if debris will impede ECCS operation during recirculation, and that appropriate corrective action be taken, if the analyses demonstrate that ECCS operation will be impeded. This plan has been updated to include NRR activities to resolve GSI-191.

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

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

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

On September 11, 1995, Limerick Unit 1 was being operated at 100-percent power when control room personnel observed alarms and other indications that one safety relief valve (SRV) was open. Attempts by the reactor operators to close the valve were unsuccessful, and a manual reactor scram was initiated.

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

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

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

Recent events at a number of plants have raised concerns regarding potential for coatings to form debris during an accident which could clog an ECCS suction. Several cases have occurred where qualified coatings have delaminated during normal operating conditions. Typically, the root cause has been attributed to inadequate surface preparation. This led the staff to raise questions regarding the adequacy of licensee coating programs. The staff issued GL 98-04 to obtain necessary information from licensees to evaluate how they implement and maintain their coating programs. In addition, Regulatory Guide (RG) 1.54 has been revised with the objective to update guidance for the selection, qualification, application, and maintenance of protective coatings in nuclear power plants to be consistent with currently employed ASTM Standards. The endorsement of industry consensus standards is responsive to OMB Circular A-119 and the NRC's Strategic Plan. RES also conducted a research program aimed at providing sufficient technical information regarding the failure of coatings to allow the staff to evaluate the potential for clogging of ECCS suctions by coating debris (or for coatings to contribute to ECCS

suction clogging). The program evaluated the failure modes of coatings, the likely causes, the characteristics (e.g., size, shape) of the debris, and the timing of when coatings would likely fail during an accident. This information was used to evaluate the ability of the coating debris to transport to the ECCS suction screens or strainers during an accident and the ultimate effect on head loss. The conclusions from the coatings portion of this action plan were utilized in both RES's assessment of PWR sump clogging and in the staff's confirmatory evaluation of BWR solutions to the strainer clogging issue.

Proposed Actions: This action plan was initially divided into four parallel efforts. Three of these efforts are complete. The action plan has been updated to provide additional NRR actions necessary to respond to RES findings related to GSI-191. The first effort was for the staff to complete its review of the resolution of NRCB 96-03. Most licensees installed their new strainers under 10 CFR 50.59, concluding that installing the new strainer modification did not constitute an unreviewed safety question. Since the staff did not receive detailed responses from these licensees describing their resolutions, the staff audited 4 plants to determine if any significant issues exist. No significant safety issues were identified. The issue has been closed based on the audit findings and the findings of the staff's review of coatings related issues (discussed below). A summary of the review results is provided in a memorandum from R. Elliott to G. Holahan, "Completion of Staff Reviews of NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-water Reactors," and NRC Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode" dated October 18, 2001.

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

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

The fourth effort involves an evaluation of PWR sumps based on new information learned during the development of the staff's resolution for NRCB 96-03. RES conducted a program to evaluate PWR sump designs and their susceptibility to blockage by debris. This evaluation included a risk assessment. Risk insights will be used to support any conclusions drawn relative to the need for licensees to address the potential for ECCS suction clogging. RES's PWR sump study is complete. RES conducted a parametric evaluation was performed to demonstrate whether sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill

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

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

Originating Document: Not Applicable.

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

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

The staff believes that there is sufficient new information and concerns raised relative to the potential for debris clogging in PWRs that this action plan has been updated to address PWR sump blockage concerns. As noted above, the results of RES's parametric evaluation demonstrated that sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill suited for making a determination that sump blockage will impede or prevent long-term recirculation at a specific plant. Therefore, it is not clear how significant a threat to PWR ECCS operation exists. The staff considers continued operation of PWRs during the implementation of this action plan to be acceptable because the probability of the initiating event (i.e., large break LOCA) is extremely low. More probable (although still low probability) LOCAs (small, intermediate) will generate smaller quantities of debris, require less ECCS flow, take more time to use up the water inventory in the refueling water storage tank (RWST), and in some cases may not even require the use of recirculation from the ECCS sump because the flow through the break would be small enough that the operator will have sufficient time to safely shut the plant down. In addition, all PWRs have received approval by the staff for leak-before-break (LBB) credit on their largest RCS primary coolant piping. While LBB is not acceptable for demonstrating compliance with 10 CFR 50.46, it does demonstrate that LBB-qualified piping is of sufficient toughness that it will most likely leak (even under safe shutdown earthquake conditions) rather than rupture. This, in turn, would allow operators adequate opportunity to shut the plant down safely (although debris generation and transport for an LBB size through-wall flaw will still need to be considered). Additionally, the staff notes that there are sources of margin in PWR designs which may not be credited in the licensing basis for each plant. For instance,

NPSH analyses for most PWRs do not credit containment overpressure (which would likely be present during a LOCA). Any containment pressure greater than assumed in the NPSH analysis provides additional margin for ECCS operability during an accident. Another example of margin would be that it has been shown, in many cases, that ECCS pumps would be able to continue operating for some period of time under cavitation conditions. Some licensees have vendor data demonstrating this. Design margins such as these examples may prevent complete loss of ECCS recirculation flow or increase the time available for operator action (e.g., refilling the RWST) prior to loss of flow. And finally, the staff believes that continued operation of PWRs is also acceptable because of PWR design features which may minimize potential blockage of the ECCS sumps during a LOCA. The RES study on sump blockage attempted to capture many of the PWR design features parametrically, however, it is not possible for a generic study of this nature to capture all the variations in plant-specific features that could affect the potential for ECCS sump blockage (e.g., piping layouts, insulation location within containment, etc.). Therefore, evaluation on a plant-specific basis is necessary to determine the potential for ECCS sump clogging in each plant.

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

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

These conclusions clearly support this action plan as outlined herein.

Current Status: The review of NRCB 96-03 responses is complete. NRR review of GL 97-04 responses is complete.

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

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

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

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

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RES Technical Contacts: Michael Marshall, ERAB, 415-5895

References:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

CONTROL ROOM HABITABILITY

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

Last Update: 12/31/01
 Lead NRR Division: DSSA
 Supporting Division: TBD

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

Description: General Design Criterion (GDC-19), "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, establishes criteria for a control room. It requires that a control room be provided which allows operators to take actions under normal conditions to

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

In its review of license amendment submittals over the past several years, the staff has identified numerous problems associated with the assessment of control room habitability. These problems have included the overall integrity of the control room envelope and the manner in which licensees have demonstrated the ability of their control room designs to meet GDC-19. Licensees have failed to: (1) assess the impact of proposed changes to plant design, operation, and performance on control room habitability, (2) identify the limiting accident, (3) appropriately credit the performance of control room isolation and emergency ventilation systems in a manner consistent with system design and operation, and (4) substantiate assumptions regarding control room unfiltered inleakage. In response to this latter concern, several utilities performed testing of their control room unfiltered inleakage using methods from ASTM E741-93, "Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." The tests performed represent about 25 percent of the operating plants' control rooms. In all of the tests performed to date, the measured unfiltered inleakage exceeded the design basis analysis assumptions; in several cases by over an order of magnitude. Also, in all of the cases to date, the licensees have been able to ultimately demonstrate compliance to GDC-19 through corrective action and retesting or by re-analysis. The 100 percent failure rate of such a large fraction of the operating plant control rooms creates a large uncertainty in the ability of the remaining untested facilities to meet control room habitability requirements.

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

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

Historical Background: In March 1998, the staff briefed the Office of Nuclear Reactor Regulation Executive Team (ET) on its concerns related to the infiltration testing results and other aspects of control room habitability. The ET directed the staff to work with the Nuclear Energy Institute (NEI) to resolve the issues. Pursuant to this direction, the staff co-hosted, with NEI and the Nuclear Heating Ventilation and Air Conditioning Users Group (NHUG), a workshop on control room habitability in July 1998. Following this workshop, NEI agreed to form a task force to address control room habitability. In August 1999, NEI submitted for staff review and comment a draft of a proposed NEI document intended to address this issue. This document, NEI 99-03, entitled, "Control Room Habitability Assessment Guidance," did not adequately address the staff's concerns. In response to the staff concerns, NEI agreed in December 1999 to restructure NEI 99-03. During the period January 2000 through June 2000, the NEI task force

met with the NRC staff in public meetings on nearly a monthly basis to resolve outstanding issues and to discuss the appropriate content of NEI 99-03. The latest NEI 99-03 revision was sent to the staff on October 13, 2000. The staff reviewed the October 13, 2000, revision and determined that, while there was much agreement on positions taken in the document, areas remained where the staff and industry were in disagreement. The staff has now determined and NEI agrees that the staff should reflect its position in formal regulatory guidance, and the issues should be resolved through the public comment process. NEI issued in June 2001 the final version of NEI 99-03, "Control Room Habitability Assessment Guidance," which is substantially the same as the October 13, 2000, draft reviewed by the NRC staff.

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

During staff interaction with the NEI issue task force, many issues were discussed and it is necessary that proper attention be applied to these issues. The staff feels that additional regulatory guidance is necessary in order that these control room habitability issues are addressed in a complete and thorough manner. In addition, it is necessary that the regulatory information associated in this area be updated to reflect current knowledge. In meetings with the NEI Task Force on Control Room Habitability, changes to design basis accident radiological analysis assumptions were discussed. The staff and industry believe it is necessary to update the analysis guidance contained in numerous current regulatory guides and consolidate it into one regulatory guide on design basis accident radiological analyses using the plant's original design and licensing source term, which in most cases is taken from TID-14844. For those licensees that implement an alternative source term as allowed by 10 CFR 50.67, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," currently provides guidance for performing control room radiological analyses. The staff also believes that creating regulatory guidance on meteorology for control room habitability assessment is necessary and appropriate. These regulatory guides would be vehicles to present to the industry and public more realistic assumptions based on current knowledge that are acceptable to the staff. In addition, it has been almost 20 years since the staff updated its information on control room habitability. Various staff and industry studies have been conducted in those 20 years. These studies have uncovered issues which were addressed to only a limited extent in the previous guidance on control room habitability. A regulatory guide on control room habitability would assist licensees to determine the present state of their control room envelope integrity. Along with the control room habitability regulatory guide, an additional regulatory guide on control room envelope integrity testing would provide guidance to the industry on how plants may determine control room envelope integrity and continually demonstrate that integrity. Such regulatory guidance would utilize the information gleaned from testing 25 percent of the control room envelopes.

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

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

Originating Document: None.

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

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

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

Current Status: DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" was issued for public comment on December 31, 2001 (ADAMS accession number ML013130132). The 3 other draft guides and the draft of the generic letter remain under revision.

Potential Problems: None.

Proposed Resolution of Potential Problems: None.

Schedule Changes Since Last Update: Resources were diverted from development of the draft regulatory guides and draft generic letter due to the staff being tasked to work on iodine spiking issues for the steam generator action plan, issues related to the terrorist attacks on September 11, 2001, as well as the regular full load of licensing issues. Because the draft regulatory guides other than DG-1111 were unable to be completed on schedule due to comment resolution and closer inspection of materials to be released to the public, the updated schedule has been changed to accommodate the requirements for public participation in the process.

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

References:

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

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

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

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

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

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

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

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

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

**GENERIC COMMUNICATION AND COMPLIANCE
ACTIVITIES**

JANUARY 2002 DIRECTOR'S QUARTERLY STATUS REPORT
Open Generic Communications and Compliance Activities

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

January 16, 2002

16-Jan-02

DIRECTOR's QUARTERLY STATUS REPORT

January 2002

Open Generic Communication and Compliance Activities

Sorted by Lead Technical Division and Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
Division of Regulatory Improvement Programs						
Events Assmt, Gen Comms & Non-Power Reactor Branch						
MB0371	IN	ENFields	--/--		02/20/2002 T IN: Debris in Standby Liquid Control System Storage Tanks	Fragments of plastic bags used for chemicals were left in SLIC tanks and might disable SLIC pumps.
MB0703	RI	CVHodge	--/--		03/31/2002 T RIS: On Improvements in Distribution of Generic Communications (GC)	Staff's proposal to use email messages with hyperlinks to disseminate GCs and to ask addressees to voluntarily inform NRC of their willingness to accept electronic msgs linked to new generic comms on the NRC web, instead of paper or electronic copies.
MB0858	RI	JWShapaker	--/--		01/30/2002 T RIS: Submitting Security Plan Changes	Proposed RIS clarifying the correct regulatory process for submitting security plan changes.
MB1120	IN	IJDozier	--/--		01/31/2002 T IN: Deficiencies in Work Packages Under Sec. 11, ASME Code	Level II examiner had not reviewed and signed work packages as required by ASME Code, Section 11.
MB1537	IN	ENFields	12/30/01	12/30/2002	T IN: Fitness-For-Duty Performance Data - Year 2000	Summarizing fitness-for-duty program performance reports for CY 2000
MB1622	IN	ICJung	02/28/02	03/03/2002	T IN: Guide Tube Failures In Westinghouse Lopar Fuel Assemblies	Guide tube failures in Westinghouse lopar fuel assemblies.
MB2112	RI	ENFields	--/--		01/30/2002 T RIS: Lessons Learned - Decommissioning/License Termination	Provides licensees with information that may help them develop more complete decommissioning plans and license termination plans.
MB2509	RI	JWShapaker	--/--		01/30/2002 T RIS: Measurement Uncertainty Recapture Power Uprate Submittals	Staff will provide guidance on the scope of information needed to conduct an efficient review of applications for power uprates based on improvements in feedwater measurement techniques.
MB2529	RI	JWShapaker	--/--		02/08/2002 T RIS: Decommissioning Funding Calculations for Power Uprates-Dusaniwsky	Will remind licensees that if they incorporate a power uprate at their facilities, that increases the thermal output of the reactor, they may be subject to an increase in decommissioning funding as stated in 10 CFR 50.75.
MB2530	RI	JWShapaker	--/--		12/31/2002 T RIS: Part 9900 Revision	Staff will inform power and nonpower reactor licensees about the availability of revised NRC inspection guidance on the resolution of degraded and nonconforming conditions.

Sorted by Lead Technical Division and Branch

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MB2534	IN	JWShapaker	--/--/--	01/30/2002	T IN: Protection of Safeguards Information From Compromise	Emphasize the need for licensees to exercise sufficient caution in handling safeguards information.
MB2788	GL	ENFields	--/--/--	01/30/2002	T GL: Revision to NEI 99-03, 5 rem TEDE-Hayes	NRC endorsement of NEI 99-03 regarding 5 rem total effective dose equivalent and the staff's intention to issue four new reg guides.
MB2864	RI	ENFields	--/--/--	01/30/2002	T RIS: Change in NRC Participation in INES-Stransky	Informs addressees of pending changes in the NRC's level of participation in the International Nuclear Events Scale.
MB2932	RI	ENFields	--/--/--	01/30/2002	T RIS: Topical Report Program - Shukla	Informs addressees that information on the NRC's topical report program is available on the NRC's public web page.
MB3005	IN	CVHodge	--/--/--	02/28/2002	T IN: Potentially Submerged Safety-Related Cables	Water found in manways containing safety-related cables at nuclear power plants.
MB3057	IN	RABenedict	12/30/01	03/01/2002	T IN: EDG Piston Wrist Pin Bearing Damage	Apparent inadequate lubrication caused bearing failure.
MB3216	RI	ENFields	--/--/--	01/31/2002	T RIS: Changes to Safety System Unavailability - Sanders	Informs addressees that a 6-month pilot test will be conducted to evaluate changes to the "safety system unavailability indicator" and to construct a reliability performance indicator.
MB3218	IN	TKoshy	--/--/--	02/04/2002	T IN: BWR Level Instrumentation Vulnerabilities	Design vulnerabilities with BWR reactor vessel level instrumentation backfill modification.
MB3246	RI	ENFields	--/--/--	03/31/2002	T RIS: Clarification NRC Req, Worker Fatigue and FFD-Desaulniers	Highlights recent concerns about worker self-declarations of fitness for duty and clarifies applicable regulatory requirements.
MB3345	IN	MSFreeman	--/--/--	04/30/2002	T IN: Use of Sodium Hypochlorite for Cleaning Diesel Fuel Oil Supply Tanks	To alert addressees to the potential problems related to the use of sodium hypochlorite solutions for cleaning diesel fuel oil supply
MB3368	IN	TKoshy	--/--/--	02/11/2002	T IN: Pump Shaft Damage Improper Hardness OG Shaft Sleeve	Pump shaft damage due to improper hardness of shaft sleeve.
MB3553	IN	CDPetrone	--/--/--	06/06/2002	T IN: IN 99-28, Sup 1, Recall of Add'l Star Brand Fire	To provide new information on failures.
MB3554	IN	CDPetrone	--/--/--	06/05/2002	T IN: Potential Problems with the Use of Heat Collectors	To provide information on defective heat collectors.
MB3555	IN	CDPetrone	--/--/--	06/05/2002	T IN: Recent Fires at Nuclear Power Plants	To provide information on recent fires at nuclear power plants.

TAC	Type	Contact	TR Comp	LA Comp	Title	Description
MB3556	IN	CDPetrone	06/01/02	06/05/2002	T IN: Potential Problems with Gaseous Fire Suppression Systems	To provide information on potential problems with gaseous fire suppression systems.

REXB has 25 GCCA(s)
DRIP has a total of 25 GCCA(s)
There are a total of 25 GCCA(s)

NOTES:

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

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

DIRECTOR's QUARTERLY STATUS REPORT
January 2002

Generic Communication and Compliance Activities Added
Since October 11, 2001

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp Title	Reason Added
MB2529	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	02/08/2002 T RIS: Decommissioning Funding Calculations for Power Uprates-Dusaniwsky	7/30/01: TAC approved by C. Petrone.
MB3057	IN	RABenedict	Events Assmt, Gen Comms & Non-Power Reactor Branch	12/30/01	03/01/2002 T IN: EDG Piston Wrist Pin Bearing Damage	10/3/01: TAC approved by C. Petrone.
MB3216	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	01/31/2002 T RIS: Changes to Safety System Unavailability - Sanders	10/18/01: TAC approved by C. Petrone.
MB3218	IN	TKoshy	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	02/04/2002 T IN: BWR Level Instrumentation Vulnerabilities	10/19/01: TAC approved by C. Petrone.
MB3246	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	03/31/2002 T RIS: Clarification NRC Req, Worker Fatigue and FFD-Desaulniers	10/23/01: TAC approved by C. Petrone.
MB3345	IN	MSFreeman	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	04/30/2002 T IN: Use of Sodium Hypochlorite for Cleaning Diesel Fuel Oil Supply Tanks	11/08/01: TAC approved by C. Petrone.
MB3368	IN	TKoshy	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	02/11/2002 T IN: Pump Shaft Damage Improper Hardness OG Shaft Sleeve	11/9/01: TAC approved by C. Petrone.
MB3553	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	06/06/2002 T IN: IN 99-28, Sup 1, Recall of Add'l Star Brand Fire	12/6/01: TAC approved by C. Petrone.
MB3554	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	06/05/2002 T IN: Potential Problems with the Use of Heat Collectors	12/6/01: TAC approved by C. Petrone.

Generic Communication and Compliance Activities Added Since October 11, 2001

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp Title	Reason Added
MB3555	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	--/--	06/05/2002 T IN: Recent Fires at Nuclear Power Plants	12/6/01: TAC approved by C. Petrone.
MB3556	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	06/01/02	06/05/2002 T IN: Potential Problems with Gaseous Fire Suppression Systems	12/6/01: TAC approved by C. Petrone.

NOTES:

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

Total Number of Records = 11

January 2002
Generic Communication and Compliance Activities Closed
Since October 11, 2001

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Closed
MA8819	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	11/08/01 P	11/08/2001	RIS: SG Tube Integrity - Industry	11/8/01: TAC closed. Need for GC is dependent upon outcome of the staff's interaction with the industry.
MA9204	IN	CVHodge	Events Assmt, Gen Comms & Non-Power Reactor Branch	12/31/01 P	01/11/2002	IN: Potential IN on Rigging Problems	1/11/02: TAC closed in lieu of RES closing generic
MA9474	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	01/08/02 P	01/08/2002	RIS: Procedure for Conducting Meetings with Proprietary Content	1/8/02: TAC withdrawn.
MA9992	RI	JWShapaker	Events Assmt, Gen Comms & Non-Power Reactor Branch	01/02/02 P	11/30/2001	RIS: Format and Content of No Significant Hazard	11/20/01: RIS 2001-22 issued.
MB1340	IN	CVHodge	Events Assmt, Gen Comms & Non-Power Reactor Branch	11/28/01 P	11/28/2001	IN: Holtec Part 21 on Excess Weight Found in Spent Fuel Racks	11/28/01: TAC closed.
MB1382	IN	CVHodge	Events Assmt, Gen Comms & Non-Power Reactor Branch	01/11/02 P	01/11/2002	IN: Highly Radioactive Particle Control Problems During Spent Fuel Pool	1/10/02: IN 2002-03 issued.
MB1793	IN	TKoshy	Events Assmt, Gen Comms & Non-Power Reactor Branch	01/10/02 P	01/10/2002	IN: Metalclad Switchgear Failures and Consequent Losses of Offsite Power	1/8/02: IN 2002-01 issued.
MB1952	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	10/29/01 P	10/29/2001	RIS: Deficiencies in the Documentation of DB Radiological Analyses Submitted in Conjunction with Lic Amdmt Reqs	10/18/01: RIS 2001-19 issued.
MB1978	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	11/20/01 P	11/20/2001	RIS: Attributes of a Proposed NSHC Determination	11/20/01: TAC closed. Duplicate of MA9992.

Since October 11, 2001

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Closed
MB2400	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	12/23/01 P	12/23/2001	RIS: Industry Initiative Fee Issue	12/23/01: TAC cancelled.
MB2403	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	10/12/01 P	10/12/2001	RIS: Scram Performance Indicator (Whitney)	10/12/01: TAC cancelled.
MB2418	IN	CDPetrone	Events Assmt, Gen Comms & Non-Power Reactor Branch	10/31/01 P	10/31/2001	IN: Recent Foreign & Domestic Experience w/Degradation of Steam Generator Tubes & Internals	10/31/01: IN 2001-16 issued.
MB2454	IN	CVHodge	Events Assmt, Gen Comms & Non-Power Reactor Branch	10/30/01 P	10/30/2001	IN: Non-Conservative Errors in Minimum Critical Power Ratio Limits	10/29/01: IN 2001-15 issued.
MB2745	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	11/16/01 P	11/16/2001	RIS: Licensing Action Estimates for Operating Reactors	11/16/01: RIS 2001-21 issued.
MB2863	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	12/03/01 P	12/03/2001	RIS: Resetting Fault Exposure Hours PI - Sanders	12/3/01: RIS 2001-23 issued.
MB3043	IN	OYTabatabai	Events Assmt, Gen Comms & Non-Power Reactor Branch	12/17/01 P	12/17/2001	IN: Inadequate Repair Renders Oil Bubblers Inoperable	12/17/01: IN 2001-19 issued.
MB3217	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	11/21/01 P	11/21/2001	RIS: Pilot Test Results on Unplanned Scrams, PI, etc. - Sanders	11/21/01: TAC cancelled.
MB3245	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	11/14/01 P	11/14/2001	RIS: Revised Guidance on NRC Policy on NOEDs	11/14/01: RIS 2001-20 issued.
MB3346	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	12/12/01 P	12/12/2001	RIS: NEI 99-02, Rev. 2 Voluntary Submission of PI Data	12/12/01: RIS 2001-25 issued.

Since October 11, 2001

TAC	Type	Contact	Lead Tech Branch	TR Comp	LA Comp	Title	Reason Closed
MB3369	IN	TKoshy	Events Assmt, Gen Comms & Non-Power Reactor Branch	01/14/02 P	01/14/2002	IN: Wire Degradation at Breaker Cubicle Door Hinges	01/10/02: IN 2002-04 issued.
MB3376	IN	ICJung	Events Assmt, Gen Comms & Non-Power Reactor Branch	01/11/02 P	01/11/2002	IN: Recent Experience With Plugged Steam Generator Tubes	01/08/02: IN 2002-02 issued.
MB3506	RI	ENFields	Events Assmt, Gen Comms & Non-Power Reactor Branch	11/30/01 P	12/06/2001	RIS: Status of Receipt of NRC Mail Following The Closing of the Brentwood Postal Facility	12/6/01: RIS 2001-24 issued.

NOTES:

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

Total Number of Records = 22

ATTACHMENT 3

RISK-INFORMED INITIATIVES

RISK-INFORMED INITIATIVES

A. CURRENT INITIATIVES			
INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
<p>1. Revised Oversight Process</p> <ul style="list-style-type: none"> - Enhanced performance indicators (PIs) - Plant & system reliability studies Industry-level Performance Indicators - Significance determination process (SDP) 	<ul style="list-style-type: none"> - analysis of PIs - piloted replacement scram and loss of normal heat removal PIs - published Risk-Based PI (RBPI) Phase 1 Report - joint NRC/industry working group met periodically to develop consistent approach for safety system unavailability reporting - developed databases to track LERs and common-cause failures (CCFs) - posted industry indicators on NRC web site - updated data for initiating events indicators - developed SDP - ROP action matrix - issued 72 plant specific SDP notebooks 	<ul style="list-style-type: none"> - developing ment of enhanced (risk-based)PIs for unreliability and unavailability - analysis/trending of PIs - developing plant-specific, risk-informed thresholds for PIs using SPAR models - working with industry to develop consistent approach for safety system unavailability reporting - developing risk-informed thresholds for ex-AEOD PIs and ROP PIs - implementing/improving SDP - revise ALARA, physical protection, SDP - evaluate fire protection, shutdown, external events, concurrent deficiencies 	<ul style="list-style-type: none"> - continue development and possible implementation of enhanced (risk-based) PIs - pilot program for unavailability and unreliability PIs - update data for operating experience studies, including system reliability - analyze data on reliability and CCFs - assess feasibility of enhanced (risk-based) PIs for containment using LERF models - develop additional risk-informed indicators and thresholds

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

A. CURRENT INITIATIVES

INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
<p>3. Risk-informed technical specifications</p>	<ul style="list-style-type: none"> - Working with NSSS owners groups and NEI to coordinate submittals - Goal is to reflect safety significance of the condition or requirement - Eight industry initiatives <ol style="list-style-type: none"> 1. modified end states 2. missed surveillance 3. flexible mode restraints 4. risk-informed AOTs with a backstop 5. optimize surveillance frequencies 6. modify LCO 3.0.3 to about 24 hours 7. define actions to be taken when equipment is not operable but functional 8. risk-inform the scope of the TS rule 	<p>Initiative 2 complete and available using a Consolidated Line Item Improvement Process</p> <p>Reviewing submittals for initiatives 1&3</p> <p>Reviewing industry concepts for initiatives 4 and 7.</p>	<p>Continue reviews of initiatives</p> <p>Define "pilot" effort to support initiative 4</p>

A. CURRENT INITIATIVES			
INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
4. Fire protection	<p>- NFPA-805 national standard was issued in April 2001</p> <p>- NFPA-805 is an alternative performance-based risk-informed fire protection standard for nuclear power plants.</p> <p>- Circuit Analysis Resolution Program</p>	<p>- Staff working on proposed rulemaking that would endorse NFPA 805 as a voluntary alternative to NRC existing fire protection regulations. Draft rule language was posted on the NRC Regulatory Forum web site for public comment in December 2001. Separately, NEI is interacting with the staff regarding its effort to separately develop implementation guidance for NFPA-805. NRC plans to endorse the guidance via Regulatory Guide.</p> <p>- staff working with industry to develop risk-informed post-fire safe shutdown methodology document. Staff is reviewing NEI 00-01 Draft Rev. C and will forward its comments in January 2002.</p>	<p>- Over the next 9 months, the staff will develop proposed rule language and associated rulemaking package, solicit public input in the NRC web's Rulemaking Forum, obtain Office concurrences, brief AGRS and CRGR, and provide proposed rule to Commission for notation vote. Brief AGRS and CRGR, and resolve comments by May 2002. Proposed rule to EDO with Office concurrences by July 2002. Provide proposed rule to Commission for notation vote in July 2002.</p> <p>- NEI is proceeding to pilot its methodology at nuclear powerplants but has not yet provided the completed methodology to the NRC staff. NEI plans to provide a final version of NEI 00-01 to the staff in the first quarter of CY 2002 for formal staff review.</p>

A. CURRENT INITIATIVES			
INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
5. Safeguards	<p>- Proposed revisions to 10 CFR 73.55 sent to Commission 6/4/01. Proposal requires that licensees' security programs employ risk insights in identifying based on risk-informed target sets of equipment necessary to prevent core damage and/or spent fuel sabotage and create creates a more performance oriented basis for security regulations.</p> <p>Proposed 73.55 returned by Commission to staff for rework to reflect lessons learned from September 11, 2001, events.</p>	<p>- Subsumed by staff efforts on post-September 11, 2001, Response to Terrorist Activities.</p>	<p>- Subsumed by staff efforts on post-September 11, 2001, Response to Terrorist Activities.</p>

A. CURRENT INITIATIVES			
INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
6. RIP50/Option 2 (risk-informing scope of special treatment requirements)	<ul style="list-style-type: none"> - Published ANPR 3/00 - STPNOC exemptions issued August 2001 - Conceptual rule language made public September 2001 - Public workshop on treatment alternatives held on November 7, 2001 - Draft rule language made available for public comment on NRC web site. (Notice of Availability published in November 29, 2001, Federal Register) 	<ul style="list-style-type: none"> - Reviewing industry guidance documents - Pilot plants conducting IDP review of categorization - Public workshop on treatment requirements in November 2001 - Reviewing public comments on draft rule language 	<ul style="list-style-type: none"> - Complete review of industry guidance documents - Review pilot plants results - Publish proposed and final rules (10 CFR 50.69)

A. CURRENT INITIATIVES			
INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
7. RIP50/Option 3 (risk-informing technical requirements)	<ul style="list-style-type: none"> - Developed framework document to guide Option 3 efforts - Completed detailed technical review and proposed changes to 10 CFR 50.44 - Notice published in November 14, 2001, <u>Federal Register of availability of draft 10 CFR 50.44 rule language for public comment on the NRC web site</u> - Completed feasibility study of risk-informed changes to 10 CFR 50.46 	<ul style="list-style-type: none"> - Reviewing public comments and developing proposed rule changes for 50.44 - Developing technical basis for proposed changes to 50.46 and associated rules - Developing technical basis for risk-informed changes to 10 CFR 50.61 	<ul style="list-style-type: none"> - Publish final revisions to 50.44 - Publish proposed and final rule changes to 50.46 - Publish proposed and final rule changes to 50.61
8. PRA standards	<ul style="list-style-type: none"> - Working with ASME on internal events standard - Working with ANS on low power and shutdown and external events standards - Industry developing guidance on peer reviews 	<ul style="list-style-type: none"> - Continuing work with ASME and ANS - Reviewing industry guidance on peer reviews 	<ul style="list-style-type: none"> - Develop regulatory guidance which endorses industry standards generically or for specific applications (e.g., Option 2) and industry guidance on peer review

A. CURRENT INITIATIVES			
INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
9. Creating a risk-informed environment	<p>=Began effort within NRR to create environment in which risk-informed methods are fully integrated into staff activities</p> <ul style="list-style-type: none"> - Conducted 12 individual interviews and 13 focus group discussions with about 100 staff and management to gather information on current environment. - Completed draft report (December 2001) on current environment based on interviews and focus group discussions. 	<ul style="list-style-type: none"> - Evaluating current environment based on results of interviews and focus group discussions. - Developing framework for risk knowledge and information system. 	<ul style="list-style-type: none"> - Establish target environment - Implement target environment - Assess effectiveness
10. Pebble Bed Modular Reactor licensing approach	<ul style="list-style-type: none"> - Exelon submitted risk-informed, top-down approach for licensing pebble bed modular reactors (PBMR) similar to General Atomics MHTGR approach in early 1990's. 	<ul style="list-style-type: none"> - RES/NRR working group evaluating Exelon proposal - Ongoing meetings with Exelon - Commission paper providing staff assessment of Exelon approach to EDO for concurrence 	<p>-Commission paper planned in November on Exelon's approach</p> <ul style="list-style-type: none"> - RES/NRR staff will continue to assess Exelon proposal and identify policy issues in a Commission paper to be provided at the end of the PBMR pre-application review.

A. CURRENT INITIATIVES			
INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
11. Advanced Reactor Regulatory Framework	<ul style="list-style-type: none"> - NEI indicated desire to discuss advanced reactor regulatory framework with staff - Staff identified possible need for advanced reactor regulatory framework in Future Licensing and Inspection Readiness Assessment 	<ul style="list-style-type: none"> - Staff meeting internally to discuss options for an advanced reactor risk-informed regulatory framework. - Staff met with NEI in November 2001 	<ul style="list-style-type: none"> - NEI working with Exelon and developing a more generic approach for any new plant (framework modeled on ROP) NEI expects to submit a white paper to the NRC in April 2002. - Staff will review NEI proposal and other regulatory framework options in 2002. - Staff meeting with NEI in November
12. Construction Inspection Program reactivation	<ul style="list-style-type: none"> - Use of risk insights in the Construction Inspection Program is being proposed by NEI. 	<ul style="list-style-type: none"> - Ongoing meetings with NEI 	

B. COMPLETED INITIATIVES			
INITIATIVE	RECENT ACTIVITIES	CURRENT ACTIVITIES	FUTURE ACTIVITIES
1. Maintenance Rule	<ul style="list-style-type: none"> - New section (a)(4) effective 11/28/00 - RG 1.182 endorses industry guidance document for managing risk during maintenance activities 	Coordinating implementation with risk-informed technical specifications	Effectiveness review
2. Reporting Rules	<ul style="list-style-type: none"> - 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	<ul style="list-style-type: none"> - 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.