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DIRECTOR'S STATUS REPORT

on

GENERIC ACTIVITIES

Action Plans

Generic Communication and Compliance Activities

OCTOBER 2004

Office of Nuclear Reactor Regulation

INTRODUCTION

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

This report includes two attachments: 1) action plans, and 2) generic communications under development and other generic compliance activities.

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

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

ATTACHMENT 1

NRR ACTION PLANS

TABLE OF CONTENTS

OFFSITE POWER CONCERNS
(INITIAL UPDATE)

TAC No. MC3380

Last Update: Initial Update
Lead Division: DE
Supporting Divisions: DLPM, DSSA, and DIPM
Supporting Office: RES

GROUP ONE CONCERNS TO BE RESOLVED BY TEMPORARY INSTRUCTION 2515/156

Milestones	Responsibility	Estimated Completion Date
1. Issue tasking memorandum.	ADPT	01/08/04 (C)
2. Establish interoffice coordination and areas of responsibility between NRR and RES.	NRR/DLPM	01/16/04 (C)
3. Collect and prioritize grid issues for review.	NRR/DE	01/30/04 (C)
4. Identify the applicable licensing basis assumptions that were evaluated in determining reasonable assurance of adequate protection of public health and safety for the offsite AC power requirements in GDC-17.	NRR/DE	01/30/04 (C)
5. Short-term risk insights. a. Develop the risk significance of the identified issues. b. Draft of preliminary ASP results available (internal). c. Preliminary ASP analysis available for technical review (internal/external).	 NRR/DSSA RES/DRAA RES/DRAA	 05/12/04 (C) 02/06/04 (C) 02/27/04 (C)
6. Using the licensing basis information and risk, determine and reconfirm if any immediate safety concerns exist that require the staff to take immediate action (or before summer 2004) and initiate action, as appropriate. a. Issue RIS b. Issue TI c. Receipt of TI responses (key questions) d. Receipt of TI responses (remainder)	NRR/DE NRR/DIPM NRR/DIPM Regions Regions	05/21/04 (C) 04/15/04 (C) 04/29/04 (C) 06/01/04 (C) 06/30/04 (C)
7. Public Meeting with Industry.	NRR/DLPM NRR/DE	03/05/04 (C) 04/15/04 (C)
8. Regulatory Information Conference - Plenary Session.	NRR/DLPM	03/10/04 (C)

Milestones	Responsibility	Estimated Completion Date
9. Using risk significance of each issue as a guide, develop an overall project strategy, evaluate the identified issues, and determine any corrective actions and the processes to attain implementation. Update the action plan as necessary.	NRR/DE	07/27/04 (C)
10. Commission meeting on grid reliability issues.	NRR/DLPM NRR/DE	05/10/04 (C) 04/01/05 (T)
11. Establish interfaces with grid reliability organizations.	NRR/DE	on-going
12. Inform the Commission of the status of the Action Plan prior to the summer peak season.	NRR/DLPM NRR/DE	05/10/04 (C) 08/06/04 (C)
13. Evaluate Station Blackout Implications		
a. Using data from recent LOOP events, update the SBO LOOP frequency and duration(draft report for internal/external review).	RES/DRAA	10/29/04 (T)
b. Re-evaluate SBO risk (CDF) with updated SPAR models for spectrum of plants (draft report for internal/external review).	RES/DRAA	01/28/05 (T)
c. Review SBO considerations and determine if regulatory actions are needed.	NRR/DE/EEIB	06/01/05 (T)
14. Incorporate unresolved concerns into Group Three concerns	NRR/DE/EEIB	07/27/04 (C)

GROUP TWO CONCERNS TO BE RESOLVED BY ACTIONS IDENTIFIED IN 2004 NERC AUDIT REPORTS

Milestones	Responsibility	Estimated Completion Date
1. Receive NERC report.	NRR/DE	06/30/04 (C)
2. Preliminary review of available reports to determine if all concerns have been addressed in the report.	NRR/DE	07/02/04 (C)
3. Assess the information provided in the report to ascertain whether any concerns have been addressed.	NRR/DE	07/09/04 (C)
4. Incorporated results into paper (See Activity 12, page 6) to Commission.	NRR/DE	07/14/04 (C)
5. Inform the Commission of the status of the Action Plan prior to the summer peak season.	NRR/DE	08/06/04 (C)
6. Develop additional requests for information to address any short falls in the report (send to NERC).	NRR/DE	08/06/04 (C)
7. Meet with NERC to discuss their response.	NRR/DE	08/06/04 (C)
8. Re-assess any additional NERC input.	NRR/DE	08/06/04 (C)
9. Develop Group Two disposition document if different from item 5.	NRR/DE	08/06/04 (C)
10. Incorporate unresolved concerns into Group Three concerns.	NRR/DE	10/22/04 (T)

GROUP THREE CONCERNS TO BE RESOLVED BY NRR LED REVIEW GROUPS

Milestones	Responsibility	Estimated Completion Date
1. Assess input from TI responses and NERC report for possible resolution to any Group Three concerns.	NRR/DE	10/22/04 (T)
2. Organize concerns by topic as described in Activity 9 for the Group One concerns.	NRR/DE	07/27/04 (C)
3. Determine staff to be included in review groups.	NRR/DE	09/17/04 (C)
4. Determine NRR leads for review groups	NRR/DE	08/02/04 (C)
5. Incorporate Group Two concerns not resolved in Group One or Two assessments into Group Three concerns.	NRR/DE	08/06/04 (C)
6. Develop schedule for review groups to review concerns.	NRR/DE, RES, (Stakeholders input)	10/15/04 (T)
7. Review groups obtain information necessary to address concerns.	NRR/DE, RES, (Stakeholders input)	11/30/04 (T)
8. Review groups assess concerns.	NRR/DE, RES, (Stakeholders input)	02/28/05 (T)
9. Review group members develop regulatory position to present to Commission.	NRR/DE, RES	03/31/05 (T)
10. Commission briefing	NRR/DE	04/14/05 (T)
11. Final status of action plan on grid concerns to Commission.	NRR/DE	06/30/05 (T)

Description: The power blackout event on August 14, 2003, highlighted the fact that the Nation's electric grid is no longer being operated in the manner that it was considered when it was designed and constructed. An unreliable grid cannot ensure the availability of the offsite power system (preferred power supply), which is essential to ensure the safe operation of nuclear power plants (NPPs).

The plan describes the methods for resolving the concerns related to the loss of power to nuclear power plants. The plan will guide the reviews and assessments of the staff's efforts as we proceed on a resolution path of 48 concerns related to the reliability of offsite power to nuclear power plants. These concerns have been divided into three groups to be resolved.

To resolve Group One concerns the staff developed a three pronged approach. First, the staff raised awareness of the concerns by developing and issuing a Regulatory Issue Summary (RIS) 2004-05 highlighting the significance of the concerns with the reliability of offsite power to nuclear power plants. Second, the staff assessed the licensees readiness to manage any degraded or losses of offsite power through inspection and interview using Temporary Instruction TI 2515/156. Lastly, the staff maintained cognizance of conditions and events through the summer of 2004 and assessed findings to develop any proposals for long-term regulatory actions.

Concerns in Group Two may be addressed by a report to be published by North American Electric Reliability Council (NERC) assessing the grid operators implementation of the U.S. and Canada joint task force recommendations regarding the August 14, 2003, loss of electrical power outage. NERC's mission is to ensure that the bulk electric system in North America is reliable, adequate and secure. Since its formation in 1968, NERC has operated successfully as a voluntary organization, relying on reciprocity, peer pressure and the mutual self-interest of all those involved.

Group Three concerns are the remaining concerns not addressed by the other two approaches and also include those issues from two Staff Requirements Memoranda from the Commission. These concerns will be organized by topic and addressed by safety significance and the need for outside stakeholder input.

Historical Background: In 1992, the National Energy Policy Act (NEPA) encouraged competition in the electric power industry, which it defined as open generator access to the transmission system and statutory reforms to promote the wholesale of electricity. Built on that premise, in 1996, the Federal Energy Regulation Commission (FERC) issued its landmark Order 888 requiring open access to the Nation's electric power transmission system.

In 1997, the U.S. Nuclear Regulatory Commission (NRC) staff and representatives from the U.S. Department of Energy (DOE), FERC, and the electric industry briefed the NRC on the issues related to electric grid reliability and utility restructuring. In response to the staff briefing, the NRC asked the staff to give greater urgency to ensuring that health and safety issues within the NRC's jurisdiction are addressed, particularly in reviewing the terms of the licensing basis and validating assumptions about grid reliability.

In 1998 and 1999, the NRC staff evaluated the impact of deregulation on the reliability of the electric grid. This evaluation led to recommendations to confirm the licensing basis of the nuclear power plants and to reevaluate the under frequency protection trip settings.

In 2000, the NRC asked Nuclear Energy Institute (NEI) and other industry representatives to take the initiative to address the adequacy of reliable offsite power to nuclear power plants. A key aspect of that initiative was the use of recommendations contained in a Significant Operating Experience Report (SOER) on the "Loss of Grid," which Institute of Nuclear Power Operations (INPO) issued in December 1999. In that report INPO called for establishment of communication protocols between the nuclear power plant operator and the grid operator.

In December 2003, the NRC Chairman directed the Office of the Executive Director of Operations (EDO) to conduct a review of the issues raised in a report entitled "State of U.S. Power Grid from a NPP Perspective." Following deterministic and risk evaluations, it was concluded that there was certain urgency to address, before the Summer of 2004, those significant issues manifested by the August 14, 2003, event.

Proposed Actions: The staff has identified 48 concerns with the reliability of offsite power to nuclear power plants that need to be resolved. These concerns have been divided into three groups to be resolved.

Group One contains 10 concerns that the staff has determined need to be addressed in the short-term. Short-term is defined as the next potentially stressful electrical grid period (i.e., Summer 2004). To resolve Group One concerns the staff developed a three pronged approach. First, the staff raised awareness of the concerns by developing and issuing a Regulatory Issue Summary (RIS) 2004-05, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," highlighting the significance of grid reliability with respect to the operability of the offsite power system for nuclear power plants. Second, the staff assessed the licensees readiness to manage any degraded or losses of offsite power through inspections and interviews using Temporary Instruction (TI) 2515/156, "Offsite Power System Operational Readiness." Lastly, the staff monitored and reviewed conditions and events through the summer of 2004 and assessed any finding to develop any proposals for long-term regulatory actions.

Group Two has 21 concerns most of which are beyond the statutory authority of the NRC and fall within FERC's and NERC's purview. These concerns may be addressed by a report to be published by NERC assessing the grid operators implementation of the U.S. and Canada joint task force recommendations regarding the August 14, 2003, loss of electrical power outage. The staff will assess the information in this report and other NERC corrective actions to ascertain whether the Group Two concerns have been addressed by NERC.

Group Three has 17 remaining concerns not addressed by the other two approaches. These concerns cannot be addressed without further research and evaluation. Group Three concerns will be organized by topic and addressed by safety significance and the need for outside stakeholder input. An NRC review group will be assembled with the appropriate staff from the Office of Nuclear Reactor Regulation (NRR) and the Office of Research (RES) to address these concerns.

Originating Document: The originating document was a memorandum (ML033650075) to Dr. William Travers (EDO) from Chairman Nils Diaz, Chairman, dated December 16, 2003, regarding the "State of U.S. Power Grid from a Nuclear Power Plant Perspective."

Regulatory Assessment: The loss of all alternating current (AC) power at nuclear power plants involves the loss of offsite power (LOOP) combined with the loss of the onsite emergency power supplies (typically emergency diesel generators [EDGs]). This is also referred to as a station blackout (SBO). Risk analyses performed for nuclear power plants indicate that the loss of all AC power can be a large contributor to the core damage frequency, contributing up to 74 percent of the overall risk at some plants. Although nuclear power plants are designed to cope with a LOOP event through the use of onsite power supplies, LOOP events are considered to be precursors to an SBO. An increase in the frequency or duration of LOOP events increases the risk of core damage.

The staff has developed three technical papers on the safety significance of this issue: one on deterministic evaluation, another on risk, and the third incorporating deterministic and risk results. The staff has not identified any safety issues warranting immediate regulatory action. However, since the

underlying assumptions in support of the licensing basis have changed, these assumptions will need to be investigated in order to establish a new baseline. The 2004 summer peak season allowed the staff to gain information regarding the licensees capabilities to cope with a loss-of-offsite power event similar to the August 14, 2003, power outage.

Current Status: New Report.

NRR Technical Contacts:	James Lazevnick, DE/EEIB, Offsite Power System Availability Topical Area, 415-2782 Amritpal Gill, DE/EEIB, Station Blackout Review Topical Area, 415-3316 George Morris, DE/EEIB, Risk Insights Topical Area, 415-4074 Thomas Koshy, DE/EEIB, Interactions with Stakeholders Topical Area, 415-1176 Martin Stutzke, DSSA/SPSB, Risk, 415-4105
NRR Lead PM:	John G. Lamb DE/EEIB, 415-1446
RES Contact:	Dale Rasmuson, RES/DRAA, 415-7571

DAVIS-BESSE LESSONS LEARNED TASK FORCE RECOMMENDATIONS REGARDING ASSESSMENT OF BARRIER INTEGRITY REQUIREMENTS

Last Update: 09/30/04
Lead Division: RES/DET
Supporting Divisions: DRAA, DSARE
Supporting Offices: NRR, Regions

TAC No.	Description
KC0042	Develop and implement action plans based on recommendations of the Davis-Besse reactor vessel head degradation Lessons-Learned Task Force (LLTF)
MB7287	NRR support for development of action plan
MC0036	NRR Support to RES for action plan activities

Milestone	Date (T=Target) (C=Complete)	Lead	Support
Part I: Leakage			
1. a. Review PWR TS to identify plants that have non-standard RCPB leakage requirements	7/03 (C) ML031980277	NRR/DIPM	
b. Take appropriate action to make TS consistent among all plants. [LLTF 3.3.4(9):High]	9/04 (C) ML042110336		
2. Inspect plant alarm response procedure requirements for leakage monitoring systems to assess whether they provide adequate guidance for the identification of RCPB leakage. [LLTF 3.2.1(3)]			
a. Revise inspection procedures.	05/04 (C)	NRR/DIPM	RES/DET NRR/DE NRR/DSSA
b. Assess adequacy of licensee procedure requirements based on results of inspections.	05/05 (T)	NRR/DIPM	Regions
3. Develop inspection guidance pertaining to RCS unidentified leakage that includes action levels to trigger increasing levels of NRC interaction with licensees in response to increasing levels of unidentified RCS leakage [LLTF 3.2.1(2):High]	1/05 (T)	NRR/DIPM	RES/DET NRR/DE NRR/DSSA

Milestone	Date (T=Target) (C=Complete)	Lead	Support
4. Evaluate RCS leakage requirements and leakage detection systems.			
a. Perform research study to reevaluate basis for RCS leakage requirements and assess the capabilities of currently used and state-of-the-art leakage detection systems.			
(1) Provide initial draft report for internal staff comment	07/04 (C)	RES	NRR
(2) Provide revised report for internal staff comment	10/04 (T)	RES	NRR
(3) Issue final report as NUREG/CR	12/04 (T)	RES	NRR
b. Form working group to review report and make recommendations.	08/04 (C)	NRR/RES	
c. Working group to complete a white paper to address:	02/05 (T)	NRR/RES	
(1) Determine whether PWR plants should install on-line enhanced leakage detection systems on critical plant components, which would be capable of detecting leakage rates of significantly less than 1 gpm. [LLTF 3.1.5(1):High]			
(2) Recommend improvements in the requirements pertaining to RCS unidentified leakage and RCPB leakage to ensure that they are sufficient to: (1) provide the ability to discriminate between RCS unidentified leakage and RCPB leakage; and (2) provide reasonable assurance that plants are not operated at power with RCPB leakage. [3.2.1(1):High]			
(3) Evaluate appropriate regulatory tools to implement recommendations, if necessary.			

Milestone	Date (T=Target) (C=Complete)	Lead	Support
d. Prepare a memorandum from division directors to office management to disposition conclusions in staff white paper. e. Implement approved changes in RCS and RCPB leakage requirements using appropriate regulatory tools. [3.2.1(1):High]	03/05 (T) TBD	NRR/RES	
Part II. Performance Indicators (PI)			
1. Continue ongoing efforts to review and improve the usefulness of the barrier integrity PIs. Evaluate the feasibility of establishing a PI which tracks the number, duration, and rate of primary system leaks that have been identified but not corrected. [LLTF 3.3.3.(3):High]	5/05 (T)	NRR/DIPM	RES/DRAA RES/DET NRR/DE NRR/DSSA Regions
Part III. Risk Associated with Passive Component Degradation			
1. Form working group to address recommendation LLTF 3.3.7(3).	08/04 (C)	RES	NRR
2. Working group to complete a white paper to address: - Evaluate the adequacy of analysis methods involving the assessment of risk associated with passive component degradation, including the integration of the results of such analyses into the regulatory decision-making process. [LLTF 3.3.7(3)]	02/05 (T)	RES/NRR	
3. Prepare a memorandum from division directors to office management to disposition conclusions in staff white paper.	03/05 (T)	RES/NRR	

Description: The Reactor Pressure Vessel Head degradation event at the Davis Besse Nuclear Power Station has many safety implications. One concern is the integrity of the reactor coolant pressure boundary. This action plan was developed to improve some of the requirements intended to ensure an effective barrier to the release of radioactivity. This plan describes the required actions, establishes milestone schedules, identifies responsible parties, and estimates resource requirements.

Historical Background: In March, 2002, while conducting inspections in response to Bulletin 2001-01, the Davis-Besse Nuclear Power Station identified three control rod drive mechanism (CRDM) nozzles with

indications of axial cracking, which were through-wall, and resulted in reactor coolant pressure boundary leakage. During the nozzle repair activities, the licensee removed boric acid deposits from the RVH, and conducted a visual examination of the area, which identified a 7 inch by 4-to-5 inch cavity on the downhill side of nozzle 3, down to the stainless steel cladding. The extent of the damage indicated that it occurred over an extended period and that the licensee's programs to inspect the reactor pressure vessel (RPV) head and to identify and correct boric acid leakage were ineffective.

One of the NRC follow-up actions to the Davis-Besse event was formation of a Lessons Learned Task Force (LLTF). The LLTF conducted an independent evaluation of the NRC's regulatory processes related to assuring reactor vessel head integrity in order to identify and recommend areas of improvement applicable to the NRC and the industry. A report summarizing their findings and recommendations was published on September 30, 2002. The report contains several consolidated lists of recommendations. The LLTF report was reviewed by a Review Team (RT), consisting of several senior management personnel appointed by the EDO. The RT issued a report on November 26, 2002, endorsing all but two of the LLTF recommendations, and placing them into four overarching groups. On January 3, 2003, the EDO issued a memo to the Director, NRR, and the Director, RES, tasking them with developing a plan for accomplishing these recommendations. This action plan addresses the Group 4 recommendations of the Davis-Besse Lessons Learned Task Force Review Team regarding the Assessment of Barrier Integrity Requirements. The 6 high priority recommendations in the "Assessment of Barrier Integrity Requirements" grouping are included in this Action Plan. The LLTF recommendations are listed in the attached Table 1, and have been identified under the appropriate milestone(s).

Proposed Actions: The specific LLTF recommendations within this category are focused on reviewing and improving leakage detection requirements. However, simply improving leakage detection and lowering allowable leakage may not be sufficient to provide increased assurance of reactor coolant pressure boundary (RCPB) integrity. Leakage monitoring assumes that the pressure boundary will fail only under a leak-before-break (LBB) scenario. Small leak rates associated with tight stress corrosion cracks or cracks which may be partially plugged are not necessarily associated with small flaws in the RCPB. Therefore, the scope of this action plan also includes methods which may be capable of detecting crack initiation and monitoring crack growth before a through-wall crack develops and leakage occurs. Other degradation modes, such as boric acid corrosion and erosion-corrosion, which can lead to failure without leakage as a precursor will also be considered.

To support the decision for revising requirements, a comprehensive review and evaluation of plant experiences and current leakage detection systems will be performed. A similar study was performed by Argonne National Laboratory in the late 1980's. This task would essentially be to update that work. The technical bases for the current requirements on leak rates will also be reviewed. If changes should be made to leak rate limits, the impacts of these changes to other plant systems and analyses need to be identified. An evaluation of state-of-the-art systems capable of detecting leaks and cracks will also be completed. This evaluation will include, but is not limited to, acoustic emission technology. An evaluation will also be done to determine if leak rates can be correlated to unacceptable levels of degradation. It should be noted that this evaluation will be more difficult for tight stress-corrosion cracks which typically have low leak rates. Results of these reviews and analyses will then be used to develop an updated basis for leak rate requirements. Once this basis is complete, recommendations will then be made for improving leak rate limits, plant alarm response procedures, TS, and inspection guidance. Then a determination will be made to select which recommendations should be imposed as new requirements. The appropriate regulatory tools and procedures will be used to develop and implement these new requirements. A regulatory analysis will probably be needed to help establish the appropriate leakage criteria. It may not be possible or practical to implement leakage requirements small enough to preclude failure. Therefore, a regulatory impact analysis will be necessary to establish appropriate risk-informed leakage limits.

In addition to the broad study described above, some other specific activities will be implemented. First, PWR TS will be reviewed to identify plants that have non-standard RCPB leakage requirements (based on current standard TS), and appropriate action will be taken to make TS consistent. Second, inspection guidance for evaluating plant alarm response procedures will be developed and the adequacy of licensee procedure requirements will be evaluated. Finally, inspection guidance will be developed to trigger increasing levels of NRC interaction with licensees in response to increasing levels of unidentified RCS leakage.

The second group of milestones relate to LLTF recommendation 3.3.3(3) regarding the review and improvement of barrier integrity performance indicators (PI). The NRC/Industry ROP Working Group will review the feasibility of establishing a PI that tracks the number, duration and rate of primary system leaks that have been identified but not corrected, as well as other possible PIs that could monitor RCPB leakage.

Completion of this action plan may require participation in public meetings and establishing communications with stakeholders. These items will be scheduled as needed.

A working group consisting of RES and NRR staff will evaluate the adequacy of risk analysis methods for passive component degradation, including how such analysis results could be incorporated into the regulatory decision making process.

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

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

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

Regulatory Assessment: The reactor coolant pressure boundary forms one of the 3 defense-in-depth barriers to the release of radioactive products. General Design Criteria 14, 30, and 32 of Appendix A to 10 CFR Part 50 specify requirements for the reactor coolant pressure boundary.

- GDC 14 states in part that "[t]he reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage."

- GDC 30 states in part that "[m]eans shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage."

- GDC 32 states in part that "[c]omponents which are part of the reactor coolant pressure boundary shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity."

In addition, the NRC has developed Regulatory Guide 1.45 "Reactor Coolant Pressure Boundary Leakage Detection Systems."

From a practical standpoint, it was recognized that the RCPB cannot be made completely leaktight since some leakage is to be expected from equipment such as pump and valve seals. Therefore, it becomes important to identify the source of any leaks. Identified leaks, such as from valves or pump seals, should be measured, collected, and isolated so as not to interfere with detection of leakage from an unknown

source which could indicate a breach of the RCPB. Specific limitations on leakage are stated in the Technical Specifications (TS) for each plant. In general, the TS place a limit on unidentified leakage (usually to 1 gpm) and state that continued operation with RCPB leakage is not allowed. In addition Title 10, Section 50.55a of the *Code of Federal Regulations* requires plants to meet the requirements of the ASME Boiler and Pressure Vessel Code. Section XI (Inservice Inspection of Nuclear Power Plant Components) of this code provides acceptance criteria for flaws found during inspection and evaluation procedures for determining the acceptability of flaws exceeding these standards.

Since the vessel head penetration (VHP) nozzles are considered part of the RCPB and significant degradation of the RPV head occurred at Davis-Besse, the issues raised by this event extend beyond problems of stress corrosion cracking in CRDM nozzles to issues of RCPB integrity in general. Primary water stress corrosion cracking of the VHP nozzles and their associated welds has been experienced by both U.S. and foreign plants. In addition, the degradation mechanism that occurred at Davis-Besse was also known. Therefore, one of the conclusions from the LLTF report was that this incident was preventable, but occurred because of a failure to follow-up and integrate relevant operating experience and other available information.

The TS for Davis Besse set a 1 gpm limit for unidentified leakage. In general, unidentified leakage was kept below 0.2 gpm. Despite this conservatism, the leakage eventually caused the degradation found in the vessel head. Therefore, the requirements associated with RCS leakage need to be reviewed and improved as warranted.

Current Status: To address the first milestone in Part I, NRR completed a review of PWR plant TS in July 2003 and identified plants with nonstandard RCS leakage requirements. The comparison of the PWRs to the STS identified two distinct levels of non-standard reactor coolant pressure boundary TS requirements: 1) units with no TS leakage limit requirement and, 2) units with a TS leakage limit but the TS Action requirements were non-standard.

Only one PWR plant did not have TS for reactor coolant pressure boundary leakage. This licensee submitted a license amendment request to make its TS consistent with the improved STS, and the staff issued the amendment in May 2004. For the other PWR units, the action requirements and completion times when the TS limit is not met are not identical to the STS. However, these plants have a reactor coolant pressure boundary TS leakage limit that is equivalent to the STS, in that the plants will take appropriate conservative actions in the time frame specified in the STS. In addition, the TS for these plants are consistent with the requirements of 10 CFR 50.36©)(2) in that the reactor must be shut down and plant cool down must be initiated. Therefore, the staff concluded that the TS are consistent among all plants and no additional actions are required.

The second milestone in Part I calls for an inspection of plant alarm response procedure requirements for leakage monitoring systems to assess whether they provide adequate guidance for identifying RCPB leakage. To address this recommendation, inspection guidance has been revised to verify that licensees have programs and processes in place to (1) monitor plant-specific instrumentation that could indicate potential RCS leakage, (2) meet existing requirements related to degraded or inoperable leakage detection instruments, (3) use an inventory balance check when there is unidentified leakage (4) takes appropriate corrective action for adverse trends in unidentified leak rates, and (5) pays particular attention to changes in unidentified leakage. The revised procedures include Inspection Manual Chapter 2515 Appendix D (Plant Status Review), Inspection Procedure 71111.22, and Inspection Procedure 71111.08. These revisions were issued in May 2004. The assessment of the adequacy of licensee procedure requirements will be completed as part of the annual ROP self assessment process.

The third milestone is also addressed in the revision to IMC 2515, Appendix D. Inspectors are to monitor leakage for adverse trends and notify plant management and regional management if any are noted. Development of additional technical guidance, such as a tool to determine statistically if a trend exists, is under consideration.

The fourth milestone is being addressed by the Barrier Integrity Research Program that is being performed at the Argonne National Laboratory. The objective of this program is to reevaluate the technical basis for RCS leakage requirements. There are 3 main tasks associated with this effort. The first task is an assessment of the leakage associated with the degradation of various RCPB components. This includes a review of leak rate experiments and models to identify correlations between crack size and leak rate. A set of leak rate calculations are also being performed using an updated version of the Seepage Quantification of Upsets in Reactor Tubes (SQUIRT) code developed by the NRC. The second task is a review of leakage operating experience by developing a database of leakage events. The information in this database includes (1) leak location, (2) leak rate, (3) cause of leakage, (4) operation of reactor when leak was detected, and (5) action taken. The third task is an evaluation of the capabilities of various leakage detection systems. To date the systems that have been evaluated included acoustic emission, humidity detection, and localized airborne radioactivity monitoring. In addition, this task is evaluating the capabilities of acoustic emission systems to monitor and detect cracking in RCS components before leakage occurs. On March 24, 2004 a program review meeting was held at headquarters in which Argonne and its subcontractors presented interim results of this program to the staff.

At the end of May 2004, ANL provided a draft NUREG report on barrier integrity research.
This draft report contains an updated review of RCS leak rate experiments and leak rate models and identifies correlations between crack size, crack opening displacement (COD), and leak rate. Although the focus of this work is on components susceptible to stress corrosion cracking (SCC), other types of materials and cracking mechanisms are considered.

A database was developed which identifies the number, source, rate, and resulting actions from RCS leaks discovered in U.S. LWRs. It describes for each incident what equipment detected the leakage, how it was determined that the leakage was through the pressure boundary, the cause of leakage, and comparisons with applicable leakage requirements. If the leakage was from a crack in the pressure boundary, the crack size, crack type, and measured leak rates are also described. For each incident the database notes what, if any, indications in identified or unidentified leakage were present (i.e., change in some measurement when the pressure boundary was breached).

The capabilities of each type of leakage detection system were evaluated to determine their sensitivity, reliability, response time, and accuracy. The scope of technology considered includes the state-of-the-art in this area, but was limited to technology that can be applied to the monitoring of RCPB conditions in U.S. LWRs. The evaluations also included crack monitoring systems capable of detecting crack initiation and growth. In addition, technology that can monitor or detect other (non-cracking) degradation modes such as boric acid corrosion or erosion/corrosion was studied. The sensitivity, reliability, response time, and accuracy of these systems, as well as the feasibility of using this technology in nuclear power plant applications have been considered. The systems, procedures, and equipment used in nuclear power plants of other countries to detect leakage were also evaluated.

The RES and NRR staff reviewed this draft report and provided comments for inclusion in the final report. The revised draft report will be available for staff use in October 2004 and the final report will be issued as a NUREG/CR by December 2004.

A working group has been formed to use the information contained in the ANL report, as well as other pertinent plant information, to (a) determine if PWR plants should install on-line enhanced leakage detection systems [LTTF No: 3.1.5(1)], and (b) recommend improvements in the requirements for RCS unidentified leakage and RCPB leakage [LLTF No: 3.2.1(1)].

The Part II milestones regarding Performance Indicators have been revised to indicate more clearly that the response to LLTF 3.3.3(3) is continuation of the ongoing process of working with the industry to improve the Barrier Integrity PIs and to evaluate the feasibility of a PI that tracks the number, duration and rate of primary leaks that have been identified but not corrected.

Part III milestones were added to track completion of LLTF 3.3.7(3). A working group consisting of RES and NRR staff has been formed and is currently studying the risk assessment methods related to passive component degradation, evaluating their adequacy, and has been periodically meeting and discussing technical challenges and paths to writing a white paper on this issue.

Contacts:

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NRR/DE Lead Contact:	William Bateman, DE, 415-2795

References:

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

Memorandum from Ledyard Marsh, Deputy Director Division of Licensing and Project Management, to John Grobe, Chair, Davis-Besse Reactor Oversight Panel, dated December 6, 2002, "Response to Request for Technical Assistance - Risk Assessment of Davis-Besse Reactor Head Degradation (TIA-2002-01)" [ML023330284]

10 CFR Part 50 Appendix A

NRC Regulatory Guide 1.45 "Reactor Coolant Pressure Boundary Leakage Detection Systems"
NUREG/CR 4813, "Assessment of Leak Detection Systems for LWR's," May 1988, Argonne National Laboratory.

Table 1
LLTF Report Recommendations Included in Barrier Integrity Action Plan

High Priority

RECOMMENDATION NUMBER	RECOMMENDATION
3.1.5(1)	The NRC should determine whether PWR plants should install on-line enhanced leakage detection systems on critical plant components, which would be capable of detecting leakage rates of significantly less than 1 gpm.
3.2.1(1)	The NRC should improve the requirements pertaining to RCS unidentified leakage and RCPB leakage to ensure that they are sufficient to: (1) provide the ability to discriminate between RCS unidentified leakage and RCPB leakage; and (2) provide reasonable assurance that plants are not operated at power with RCPB leakage.
3.2.1(2)	The NRC should develop inspection guidance pertaining to RCS unidentified leakage that includes action levels to trigger increasing levels of NRC interaction with licensees in order to assess licensee actions in response to increasing levels of unidentified RCS leakage. The action level criteria should identify adverse trends in RCS unidentified leakage that could indicate RCPB degradation.
3.2.1(3)	The NRC should inspect plant alarm response procedure requirements for leakage monitoring systems to assess whether they provide adequate guidance for the identification of RCPB leakage.
3.3.3(3)	The NRC should continue ongoing efforts to review and improve the usefulness of the barrier integrity PIs. These review efforts should evaluate the feasibility of establishing a PI which tracks the number, duration, and rate of primary system leaks that have been identified but not corrected.
3.3.4(9)	The NRC should review PWR plant TS to identify plants that have non-standard RCPB leakage requirements and should pursue changes to those TS to make them consistent among all plants.

Medium Priority

3.3.7(3)	Evaluate the adequacy of analysis methods involving the assessment of risk associated with passive component degradation, including the integration of the results of such analyses into the regulatory decision-making process.

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

TAC No.	Description	
MB7281	Develop Action Plan	Last Update: 09/30/04
MB7726	Evaluation of Inspection and Assessment Guidance	Lead Division: DIPM
		Supporting Division: DLPM
		Supporting Office: Regions

Milestone		Date (T=Target) (C=Complete)	Lead	Support
Part 1 - Evaluation of Inspection Guidance Related To Problem Identification and Resolution				
The NRC should revise its inspection guidance to provide assessments of: (1) the safety implications of long-standing, unresolved problems; (2) corrective actions phased in over several years or refueling outages; and (3) deferred modifications. [LLTF 3.2.5.(2) High] The NRC should revise the overall PI&R inspection approach such that issues similar to those experienced at DBNPS are reviewed and assessed. The NRC should enhance the guidance for these inspections to prescribe the format of information that is screened when determining which specific problems will be reviewed. [LLTF3.3.2.(2) Low] The NRC should provide enhanced Inspection Manual Chapter guidance to pursue issues and problems identified during plant status reviews [LLTF3.3.2.(3) Low] The NRC should revise its inspection guidance to provide for the longer-term follow-up of issues that have not progressed to a finding. [LLTF3.3.2.(4) Low]				
1.	Make changes to IP 71152 to require annual follow-up of three to six issues.	01/02 (C)	DIPM	
2.	PI&R focus group assess lessons learned recommendations.	03/03 (C)	DIPM	Regions
3.	Develop draft procedure changes based on PI&R group recommendations and provide to regions for review.	04/03 (C) ML031390010	DIPM	Regions
4.	Provide training on procedure changes.	09/03 (C)	DIPM	
5.	Issue procedure changes.	09/03 (C)	DIPM	

Milestone		Date (T=Target) (C=Complete)	Lead	Support
Part 2 - Evaluation of IMC 0350 Guidance				
The NRC should develop guidance to address the impacts of IMC 0350 implementation on the regional organizational alignment and resource allocation. [LLTF3.3.5.(4) High]				
1.	Assess past and present IMC 0350 data and associated inspection approaches.	06/03 (C) MI031890873	DIPM	Regions
2.	Develop enhanced structure to the inspection approach used for IMC 0350 plants.	08/03 (C) ML032250336	DIPM	Regions
3.	Develop draft revisions to IMC and issue for regional comment.	10/03 (C)	DIPM	
4.	Issue procedure revisions.	12/03 (C)	DIPM	
5.	Include estimated resources for IMC 0350 plants into budget cycles.	12/03 (C) ML033010385	DIPM	
Part 3 - Evaluation of Project Management Guidance				
The NRC should establish guidance to ensure that *decisions to allow deviations from agency guidelines* and recommendations issued in generic communications are adequately documented. [LLTF 3.3.7.(2) High]				
1.	The DLPM Handbook will be updated with a new section that addresses documenting staff decisions.	02/03 (C)	DLPM	
2.	A training package emphasizing compliance with the requirements of MD 3.53 will be developed and distributed to all Offices and regions.	04/03 (C) ML030300067	DLPM	
3.	Issue Office Instruction on Generic Communications	06/03 (C) ML023170311	DRIP	DLPM

Milestone	Date (T=Target) (C=Complete)	Lead	Support
4. Conduct effectiveness review:			
a. Follow up with Offices and Regions to determine effectiveness of training.	07/04 (C) ML041200528	DLPM	
b. Review sample of generic communication closeouts for appropriate documentation.	06/04 (C) ML041810128	DLPM	
c. Complete additional training and procedure revisions as indicated by effectiveness review.	03/05 (T)	DLPM	

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

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

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

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

Project management guidance regarding documentation when accepting deviations from generic communications recommendations will be incorporated into the DLPM handbook and into training materials to be distributed to all Offices and Regions. An Office Instruction will be issued to provide guidance on preparation, issuance and closeout of generic communications.

Originating Documents:

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

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

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

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

Current Status: Part 1 milestones are complete. The procedure changes have been issued and a training module was placed on the web-based "Read and Sign" training for inspectors. Inspection Procedure (IP) 71152, "Identification and Resolution of Problems," was revised to require the resident inspector to perform a screening review of each item entered into the corrective action program. The intent of this review is to be alert to conditions such as repetitive equipment failures or human performance issues that might warrant additional follow-up through other baseline inspection procedures. IP 71152 was also revised to require a semi-annual review to identify trends that might indicate the existence of a more significant safety issue. Included within the scope of this review are repetitive or closely related issues that may have been documented by the licensee outside the normal corrective action program, such as in trend reports or performance indicators, major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, maintenance rule assessments, or corrective action backlog lists. Finally, IP 71152 was revised to include enhanced requirements regarding routine PI&R reviews conducted by the resident inspectors, biennial reviews of longstanding issues, and biennial reviews of licensees' operating experience issues.

To address the issue of deferred modifications, the staff revised IP 71111.15, "Operability Evaluations." The objective of this procedure is to review operability evaluations affecting mitigating systems and barrier integrity to ensure that operability is properly justified and the component or system remains available, such that no unrecognized increase in risk has occurred. The procedure was revised to include deferred modifications as one of the areas an inspector can assess to ensure that structures, systems, and components are capable of performing their design function.

Part 2 milestone activities have been completed. The Inspection Program Branch completed an evaluation of the IMC 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems," process in June 2003, (ML031890873). It identified the need for specifically budgeting resources for IMC 0350 inspections and providing prescriptive inspection guidelines for the process. The budget estimate was increased for FY2005 and beyond (ML033010385) to account for one IMC 0350 plant per year. IMC 0350 was revised in December 2003, to provide additional inspection guidelines.

The Part 3 milestones associated with issuing guidance were completed as scheduled. The milestones to follow up on the effectiveness of training and to perform a review of a sample of recent generic communications for proper closeout documentation are complete.

The effectiveness evaluations indicated a need for additional action to improve guidance documents and to conduct additional training. This was added to the action plan.

Contacts:

NRR Lead for this action plan: Stuart Richards, DIPM, 415-1257
Overall Lead for DB LLTF response: Brendan Moroney, DLPM, 415-3974

References:

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

Table 1
LLTF Report Recommendations Included in This Action Plan

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

SIGNIFICANCE DETERMINATION PROCESS (SDP) IMPROVEMENT

TAC Nos. MA9164, MB0046, & MB2203

Last Update: 09/30/04
Lead Division: DIPM
Supporting Division: DSSA

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

Coordinator: Peter Koltay, IIPB/DIPM/NRR

Task		Completion Date	Lead	Status
1. Improve Focus on Early Resolution of Specific Technical Questions and Internal Staff Disagreements				
Objective 1.1	Implement a weekly management status report on SDP issues in process. **[SDP 3.9.3(1)]**	04/01/02 (C)	IIPB	SDP Activities Tracking List implemented 2/1/02 to address Objectives 1.1-1.2.
Objective 1.2	Incorporate features to provide for early identification of SDP issues that are likely to become untimely due to technical, policy, or process issues. **[OIG - 6]**	(C)	IIPB	The Active Issues Matrix communicates a running summary of active SDP findings focusing senior HQ and regional managers' attention to on timeliness issues. Intermediate goals leading to 90% within 90 days are being met.
Objective 1.3	Develop and track/trend SDP timeliness metrics within ROP Self-Assessment Process, including the cycle-time calculation for major process steps.	06/28/02 (C)	IIPB	IMC 0307, Reactor Oversight Process Self-Assessment Program, incorporates the relevant timeliness metrics.
Objective 1.4	Implement a requirement to conduct a self-assessment for SDP results that are not timely.	06/28/02 (C)	IIPB	IMC 0307 changed to reflect requirement to conduct self-assessments during annual review of baseline procedures.

Task		Completion Date	Lead	Status
Objective 1.5	Rectify the difference between the NRR Operating Plan and IMC 0307 for SDP timeliness. [SDP 3.9.3(3)]	10/1/03 (C)	IIPB	Timeliness goals in the NRR Operating Plan are referenced in IMC 0609.
Objective 1.6	Incorporate SDP timeliness metrics Into the Regional Operating Plans. **[SDP 3.9.3(1)]**	12/30/03 (C)	IIPB	Memorandum from NRR to the regions explaining timeliness goals are included into the regional operating plans (ML0321602552).
Objective 1.7	Change IMC 0307 "ROP Self-Assessment Program" to improve evaluation of inspection effectiveness in timely identification of performance deficiencies inspection. **[OIG - 5]**	12/30/03 (C)	IIPB	IMC0307, ROP Self Assessment, Section 06.01b6, Timeliness of Identification, addresses the issue.
2. Improve SDP Process				
Objective 2.1	Revise Attachment 1 of IMC 0609 to clarify the roles and responsibilities of the SERP, to include an escalation process for resolution of issues for which the SERP cannot reach a consensus position, and to include process timeliness goals.[1]		IIPB	IMC 0609 Att. 1 was revised April 30, 2002, to incorporate this enhancement to the SERP process.
a.	Clearly define the accounting process of the 90 day time period including: Starting time End time	08/01/02 (C)		Guidance is provided in IMC 0609 Att. 1 and tracked under Objective 1.1 of the Plan.

Task		Completion Date	Lead	Status
b.	Communicate the Agency's timeliness goals to licensees (e.g., Choice Letters, Regulatory Conferences, Reg. Information Conference, etc.). **[SDP 3.9.3(2)]**	12/01/03 (C)		Timeliness goals have been and will continue to be communicated to all stakeholders through all identified interactions. This change will also be incorporated into next revision of IMC 0609.01 (Choice Letter), in FY 2004.
c.	Improve the SERP process: Clearly identify SERP participants and define their respective roles and responsibilities in IMC0609.01.	06/28/02 (C)		IMC 0609 Att. 1 was revised April 30, 2002, to identify SERP participants and their roles and responsibilities.
d.	Outline the escalation process for issues where the SERP fails to reach consensus in IMC0609.01.	06/28/02 (C)		IMC 0609 Att. 1 was revised April 30, 2002, to outline the escalation process when SERP fails to reach Consensus.
e.	Improve the Regulatory Conference process and associated activities: Designation of NRC participants Post conference caucus Post conference re-SERP Post conference SDP and re-SERP.	6/28/02 (C)		IMC 0609 Att. 1 was revised April 30, 2002, to improve the effectiveness of the Regulatory Conference and post-conference caucus.
Objective 2.2	Engage the regions to confirm their understanding and implementation of the expectations regarding use of the SDP, including guidance on the level and type of licensee engagement that is appropriate during the conduct of:[2] **[SDP 3.2.3(2)]**		IIPB Support: SPSB	Routine bi-weekly teleconferences are held with the Regions. NRR emphasizes expectations noted in IMC 0609 and the August 9, 2002, memorandum from S. Collins to the Regional Administrators on "Reactor Oversight Expectations for Inspector Use of the Significant Determination Process". This memo provides specific instructions on the level and type of licensee engagement for each phase of the SDP.

Task		Completion Date	Lead	Status
a.	SDP Phase 2 risk analyses.	08/01/02 (C)		
b.	SDP Phase 3 risk analyses.	08/01/02 (C)		
c.	Communicate expectations for inspector use of the phase 2 notebooks during interim period in which enhanced pre-solved tables are being developed.	08/01/03 (C)		A revised "Expectations" memorandum provides clear instructions regarding the use of the phase 2 risk notebooks during the development of the pre-solved SDP tables. ML031270689
Objective 2.3	Issue guidance on the use of the site specific risk-informed inspection notebooks (hence referred to throughout this document as **the notebooks**) within the overall context of the SDP.[2,3]		IIPB Support: SPSB	Based on experience gained from the initial notebook benchmarking efforts and ROP implementation, additional notebook usage guidelines were developed and presented to SRAs for discussion and comments. The final version of the guidelines were incorporated into "Expectations Memorandum" and IMC 0609.
a.	Use of the revision 0 notebooks (pre-benchmarking).	05/31/02 (C)		
b.	Use of the benchmarked notebooks, revision 1.	05/31/02 (C)		
c.	Guidance when additional analysis beyond the capability of the notebooks needs to be conducted.	05/31/02 (C)		

Task		Completion Date	Lead	Status
Objective 2.4	Evaluate revising the SDP to require that the preliminary characterization of potentially risk significant issues be "potentially greater than green," rather than a specific color.[2] **[SDP 3.9.3(4)]**		IIPB	This issue was presented to the DRP/DRS Division Directors during the August 20-21, 2002, counterpart meeting and the proposed change to 0609 was issued for review and comment. The proposed revision to the 0609 guidance was also discussed during the January 2003 ROP public meeting.
a.	Collect and evaluate regional input.	01/31/03 (C)		
b.	Make final determination on changing the process to preliminary greater than green, or stay with the existing process or preliminary specific color.	04/30/03 (C)		The revision to IMC 0609 Att. 1 was issued on March 21, 2003, to allow for the use of "greater than green" preliminary SDP characterization.
Objective 2.5	Assemble a focus group of internal stakeholders to identify key SDP-related issues going forward and provide recommendations for their resolution, consistent with the ROP principles and objectives.[3]		IIPB Support: SPSB, Regions	The SDP Task Group was formed consisting of regional and headquarters staff. A charter was developed and the SDPTG completed a comprehensive review of the SDP and provided recommendations to enhance the overall effectiveness of the process. The recommendations have been accepted by NRR and incorporated into this Plan, as noted.
a.	Identify focus group members.	05/01/02 (C)		
b.	Develop charter.	06/28/02 (C)		

Task		Completion Date	Lead	Status
c.	Present recommendations.	12/20/02 (C)		
Objective 2.6	Develop a plan for long range improvements to the SDP.[3] **[OIG-1]**	Complete		The SDP Improvement Initiative Task Action Plan is NRR's tool for tracking SDP improvement activities.
a.	Issue the proposed SDP basis document, including the current performance expectations for the notebooks. The notebook "construction rules" should also be included or referenced in the proposed SDP Basis Document. **[SDP 3.2.3(1)] [SDP 3.6.3(4)]**	05/30/04 (C)	IIPB, SPSB	SPSB provided the notebook construction rules, to be incorporated into the basis document, in April 04. The revised Inspection Manual Chapter IMC 0308, "Reactor Oversight Process Basis Document", was signed June 25, 2004, and is going through the issuing process.
b.	Re-evaluate the performance expectations of the SDP tools after completion of the notebook benchmarking and modify program guidance, as appropriate, to reflect any revisions to the expectations. **[SDP 3.2.3(3)]**	03/31/04 (C)	IIPB	Evaluation of SDP effectiveness and performance expectations is conducted as part of the routine annual assessment process outlined in IMC 0307. A re-evaluation of the benchmarked notebooks resulted in the ongoing notebook standardization process identified in Objective 3.1 c.

Task		Completion Date	Lead	Status
3. Improve SDP Tools				
Objective 3.1	Revise IMC 0609 App. A to improve the guidance for conducting a phase 2 analysis to:(3)		SPSB Support: IIPB	
a.	Develop tools and simplify the process of accounting for external initiators in phase 2 of the SDP.	TBD (Multy year effort)	SPSB	Task group will evaluate the IPEEEs of 3 sites before identifying the appropriate methodology to develop risk informed inspection tools. Decision on how to proceed will be made based on the outcome of the assessment. BNL is assisting in the selection of the methodology. The pilot will be completed 09/04 at which time the final direction for this activity will be determined. Draft report based on the pilot, identifies three methodologies as options which may be implemented to facilitate risk assessment. Communications plan under development.
b.	Clarify the guidance on the treatment of concurrent issues.	04/01/02 (C)		Guidance incorporated in March 18, 2002, revision to IMC 0609 App. A Section III.

Task		Completion Date	Lead	Status
c.	Standardize benchmarked notebooks and develop pre-solved risk tables from standardized (re- benchmarked) notebooks. **[SDP 3.1.3(2)] [SDP 3.6.3(1)] [OIG-1]**	10/30/05 (T)	SPSB	Notebooks that were benchmarked during the early stages of the initiative will be revised to incorporate lessons learned from the benchmarking process. This may require approximately 10 additional site visits. Notebooks will be standardized by completing the re- benchmarking process and ready to be issued as revision 2 by 06/30/05. This will form the basis for pre-solved phase 2 SDP worksheets. No funding or FTE has been allocated for the development of pre-solved risk tables.
d.	Evaluate training needs and issue revised guidance for the use of the pre-solved risk tables.	10/30/05 (T)	IIPB	Supplemental training needs will be evaluated prior to issuance of the pre-solved risk tables.
Objective 3.2	Develop a plan to benchmark and revise all of the notebooks (Revision 1). Develop and implement a quality assurance (QA) plan for the development of revision 1 to the notebooks.[2, 3] **[SDP 3.1.3(2)]**		SPSB Support: IIPB	
a.	Schedule and complete benchmarking plan (site visits)	10/01/03 (C)		All benchmarking trips completed. Final Revision 1 notebooks are available to internal stakeholders on the DSSA/SPSB and SRA web pages.

Task	Completion Date	Lead	Status
This activity is incorporated into Objective 2.6 a.			See Objective 2.6 a.
b. Develop and implement QA plan for development of the notebooks.	03/01/02 (C)		QA Plan developed and provided to BNL for development of the notebooks.
c. Implement a process to compare the results of the QA'd SPAR models and benchmarked notebooks. **[SDP 3.6.3(3)]**	09/30/03 (C)		All benchmarking has been completed. Outcomes continue to be verified.
d. Develop notebook maintenance schedules to review and update the phase 2 tools to address licensee PRA changes and/or plant modifications. [SDP 3.6.3(2)]	4/30/05 (T)		This will be accomplished on a case by case basis. No funding or FTE has been allocated for this effort.
Objective 3.3 Develop or improve existing SDP tools as applicable in the following areas: **[OIG-3]**			
a. Fire protection	05/31/04 (C)	SPSB	Three training sessions attended by most fire protection inspections and SRAs.
b. Maintenance rule	12/31/04 (T)	SPSB/ IIPB	Internal stakeholder comments are being incorporated.
c. Containment	05/30/04 (C)	IIPB	Issued to industry and NRC regions for review and comments May 03. Public meeting participation in July. Training has been completed.

Task		Completion Date	Lead	Status
d.	Steam generator tube integrity	04/30/04 (C)	IIPB	Public meeting with NEI participation was held on 09/24/03. Some industry comments have been resolved. A new draft SDP will be issued to stakeholder comments at the October ROP meeting. SPSB will be the primary user of this guidance.
e.	Shutdown	05/30/04 (C)	SPSB/ IIPB	SDP presented to NEI and SRAs July 2002 and October 2002. Workshop held January 2003. Enhanced Appendix G to be issued November 2003. Training has been completed.
f.	Spent Fuel	12/31/04 (T)	IIPB	Under development.
Objective 3.4	Improve the physical protection SDP, if necessary, accounting for any safeguards policy changes.	12/31/04 (T)	NSIR Support: IIPB	Regional comments incorporated into draft baseline SDP ML033520291. Force on force SDP under development. When ready, the two documents will be implemented at the same time. A trial implementation of the Commission approved base line and force on force SDPs is underway.
Objective 3.5	Develop a database of all completed phase 3 analyses.[3]	10/01/02 (C)	SPSB	Database of submitted phase 3 analyses was created and is accessible via the SPSB web page. Continuing to add information. Link at http:nrr.gov/adt/ds/sa/ spsb/webpages/srapa/ ge/phase3_serps/rop-case-status.html

Task		Completion Date	Lead	Status
Objective 3.6	Consider development of analysis criteria and standards for conducting detailed phase 3 analysis.(3) **[SDP 3.5.3(2)] [OIG-4]**	6/28/02 (C)	SPSB Support: RES, Regions	11/26/02, RES developed procedures incorporating high level ASP guidance. The documents were provided to the SRAs for review to determine applicability to the phase 3 SDP.
a.	Identify participating RES and NRR personnel and establish responsibilities and a completion schedule.	8/30/03 (C)		NRR/SPSB and RES initiated the Risk Assessment Standardization Project (RASP) to develop standard methodologies and procedures for conducting phase 3 analyses.
b.	Develop criteria and to allow the staff to recognize situations where "the state of knowledge" correlation, which is described in RG 1.174, might warrant a Phase 3 analysis. **[SDP 3.7.3(1)]**	Milestone schedule due 10/01/04	RES Support: SPSB/ IIPB	The Risk Assessment Standardization Project will evaluate the possibility for developing advanced risk criteria for recognizing when modeling parameter uncertainties warrant a more in-depth analysis to properly characterize the significance of an inspection finding. Schedule and prioritization for the RASP will be developed and concurred on in a RES/NRR Office Agreement by 10/01/04.
c.	Develop guidance to allow the staff to determine whether the results of a licensee's risk analysis of a finding is of sufficient quality to use as an input to the staff's final significance determination. **[SDP 3.11.2.3(1)] [OIG-4]**	12/30/04 (T)	IIPB	The staff determined that guidance incorporated into the ROP documents, IMC 0609.01 and the notebooks, provide the assurance that licensee risk analyses consider the appropriate assumptions and uncertainties. Additionally, Regulatory Guide 1.200 published February 2004 provides an approach for determining the technical adequacy of PRA results for risk-informed activities.

Task		Completion Date	Lead	Status
Objective 3.7	Evaluate accelerating the SPAR Model Development Program (i.e., Revision 3i SPAR models, low power/shutdown models, LERF models, and external events analysis capability).(2)		RES	
a.	Develop Rev. 3i SPAR models.	9/30/02 (C)		Complete.
b.	Complete onsite QA verification (benchmarking) of Rev. 3i SPAR models.	10/31/03 (C)		Complete.
c.	Develop Low Power/Shutdown model.	12/31/05 (T)		RES is developing generic templates for each class of licensed reactor plants. Four models have been completed.
d.	Develop LERF model	12/31/06 (T)		Draft event trees have been developed.
4.	**Improve Staff Training in The Use of SDP Tools**			
Objective 4.1	Develop and conduct training on the use of the notebooks. Develop initial and periodic refresher training on the SDP.(3)		IIPB Support: SPSB/ TTC	
a.	Develop training materials for IMC 0609A revision.	4/15/02 (C)		Complete.
b.	Complete IMC 0609A training at inspector counterpart meetings: **[OIG-3]**	10/01/02 (C)		Complete.
	Region I			
	Region II			

Task		Completion Date	Lead	Status
	Region III			
	Region IV			
c.	Encourage regions to conduct annual SDP refresher training during routine inspector seminars. **[SDP 3.5.3(1)]**	6/30/03 (C)		Refresher training will be provided by regional and headquarters SRAs on an annual basis.
d.	Develop systematic assessment of training needs in the area of risk, with a particular focus on identifying and advancing the knowledge, skills, and abilities (KSAs) for implementing the SDP. **[SDP 3.5.3(3)]**	1/31/04 (C)		NRR's Risk Informed Environment Initiative and IMC 1245 Working Groups are engaged in evaluations of the necessary skills and training needs as they relate to understanding and using risk in regulatory activities. Based on their evaluations, the groups will make recommendations to enhance the training program for inspectors and risk analysts and propose improvements to staff processes, practices, and infrastructure.
Objective 4.2	Increase staffing and/or staff development in the areas of shutdown risk, seismic, fire protection, and containment risk analysis. **[OIG-3]**	6/30/02 (C)	IIPB Support: SPSB	NRR has staffed additional SRA positions within SPSB. The newly hired staff is currently completing required training for SRA certification.

Task		Completion Date	Lead	Status
Objective 5.1	The staff will develop guidance on the termination of ongoing risk evaluations when it is clear that such activity will result in exceeding timeliness guidelines. **[OIG-3]**	12/31/05 (T)	IIPB	The staff determined that a cost-benefit evaluation is not the appropriate measurer to determine when the ongoing refinement of the risk evaluation should be terminated. The staff determined that in order to meet the timeliness guidelines, the termination of ongoing refinement of risk evaluations should be based on management assessment of available information including understanding of uncertainties associated with the issue, and the plausibility of forth coming information within the timeliness guidelines. The change in date reflects the ongoing effort which will be proposed in a SECY paper to the commission.
5.	**Improve Clarity of Risk-Informed ROP Decision Guidance**			
Objective 5.2	Develop guidance that defines the attributes of a minimally acceptable risk-informed decision for use within the ROP. **[OIG-3]**	03/26/03 (C)	IIPB Support: SPSB	The attributes for reaching the minimally acceptable risk-informed decision are described in IMC 0609 Att. 1, Exhibit 4.

	Task	Completion Date	Lead	Status
Objective 5.3	Revise the ROP guidance to explicitly indicate that traditional engineering analysis considerations (e.g., reduction of safety margin, or significant loss of defense-in-depth) should be used to determine an appropriate color to associate with findings where the uncertainty in the risk evaluation arising from the characterization of the impact of the inspection finding is large enough that the color is indeterminate on the basis of the risk analysis. This guidance should promote consistency and be used only where the uncertainty is significant (i.e., when alternate assumptions yield results which vary over more than two orders of magnitude). **[SDP 3.7.3(2)]**	12/31/05 (T)	IIPB Support: SPSB Regions	IIPB is in the process of identifying findings where this could be applicable and developing guidance for evaluating issues when there is a significant reduction of safety margin or loss of defense-in-depth.

Task		Completion Date	Lead	Status
6.	**Clarify Expectations for ASP and SDP Process Coordination**			
Objective 6.1	Issue guidance to delineate the role of the Office of Research in the SDP, *in order to* minimize the potential for unexpected or unreasonable differences in the results of the SDP and ASP processes. Explore efficiencies and quality enhancements that would result in better coordination and/or integration of these two programs. **[SDP 3.11.1.3(1)]**	06/30/04 (C)	IIPB Support: RES	Currently, based on a user need memo, RES reviews all greater than green issues and provides independent reviews a quarterly assessment of the specific implementation of the process. To date, all differences were minor, resulting from variation in risk assessment methodology, and were promptly resolved. This independent review program will continue indefinitely.
a.	NRR and RES should identify avenues to enhance the staff's knowledge of the ASP program, including adding a module to the P-111 course regarding the ASP program. **[SDP 3.11.1.3(2)]**		IIPB Support SPSB /RES	This issue is under Review by the IMC 1245 Working Group.

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

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

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

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

Description: In conjunction with IMC 2515, "The Policy For the Light-Water Operating Reactor Inspection Program", IMC 0609, "The Significance Determination Process (SDP)", was developed to assist the staff in using risk insights, where appropriate, to help NRC inspectors and staff determine the safety significance of inspection findings. The appendices to IMC 0609 support safety cornerstones associated with the strategic performance areas as defined in IMC 2515. The SDP determinations for inspection findings and the Performance Indicator (PI) information are combined for use in assessing licensee performance in accordance with guidance provided in IMC 0305, "Operating Reactor Assessment Program."

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

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

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

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

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

problems with the accuracy of the notebooks and concluded that benchmarking was needed to confirm the adequacy of the notebooks. Using NRC risk analysts and contractor resources, the staff began its efforts to benchmark the notebooks in April 2001.

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

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

Between May and October 2001, the OIG conducted an audit of the SDP. The objectives of the audit, as indicated in the OIG's report (OIG-02-A-15) dated August 21, 2002, were to determine whether (1) the SDP is achieving desired results, (2) NRC staff clearly understand the process, and (3) NRC staff are using [the] SDP in accordance with agency guidance. In its report, OIG concluded that "while the SDP is meeting is objectives and agency staff are using SDP in accordance with guidance, additional refinements are needed." The report provided a number of recommendations, including that the NRC develop an action plan to correct Phase 2 analysis weaknesses or eliminate this portion of the SDP. Objectives in the Plan which address the OIG recommendations are identified by recommendation number. As of September 2004, four recommendations remain resolved but not closed. Next update on the status of the recommendations is due from the staff to the OIG on January 31, 2005.

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

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

The SDP Improvement Task Action Plan (The Plan) was developed to guide staff efforts aimed at implementing the recommendations developed by the SDPTG and lessons learned since initial

implementation of the ROP. The Plan delineates responsible organizations, establishes aggressive completion dates, and provides status updates for each of the specified Plan action items.

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

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

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

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

Current Status: N/A.

Contact:

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

References:

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

Status Summary: N/A

STEAM GENERATORS

TAC Nos.	Description	
M88885	Steam Generator (SG) Integrity Rulemaking	Last Update: 9/30/04
M99432	GL: SG Tube Integrity	Lead Division: DLPM
MA4265	NEI 97-06	Supporting Divisions: DE, DIPM, DSSA
MA5037	SG Action Plan	Supporting Office: RES
MA5260	DPO on SG Issues	
MA7147	GSI-163	
MA9881	Regulatory Issue Summary - IP2 SG Tube Failure	
MB0258	SG Action Plan Administration	
MB0553	SG Inspection Program	
MB0576	Licensee SG Inspection Results Summary Reports & SG Tube Integrity Amendment Review Guidance	
MB0631	SG Workshop	
MB0633	OL No. 803 Revisions per SG Action Plan	
MB0737	IIPB SG Action Plan Activities	
MB2446	SG Risk Communication	
MB3794	SG Communication Plan	
MB7216	SG DPO Followup	
MB7842/3	Catawba Pilot Plant Application (Fee billable, not added to AGAP total)	
MC1550	NEI 97-06 Review	
MC2470	SG Tube Integrity & Associated Technical Specifications	

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

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

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.11 (MB0553)	Review the NRC inspection program and, if necessary, revise guidance to inspectors on overseeing facility eddy current inspection of steam generators. This involves the following major substeps:			
	a) review and revise the baseline inspection program.	04/30/01 (C) ML011210293	DE C. Khan	DIPM DSSA S. Long
	b.1) review how ISI results/degraded conditions should be assessed for significance by a risk-informed SDP and define needed revisions to the SDP	09/21/01 (C) ML012680252	DSSA S. Long	DE C. Khan DIPM P. Koltay
	b.2) develop and issue draft revision of risk-informed SDP using information identified in b.1 above	02/21/02 (C) ML020730318	DIPM P. Koltay	DSSA S. Long DE C. Khan
	c) review and revise the training program for inspectors	ML020560366 ML012970361	DIPM E. Kleeh	DE C. Khan
	c.1) Provide IP training material to Regions	10/11/01 (C)		
	c.2) Formal training to inspectors	02/01/02 (C)		
	(Attachment 3 to Ref. 1; TG: 5a, 5b, 5c, 5d, 5f, 6c)			
1.12 (MB0576)	Determine need for formal written guidance for technical reviewers to utilize in performing steam generator tube integrity license amendment reviews (TG: 5c, 6a)	04/30/01 (C) ML011220621	DE S. Coffin	
1.13 (MB0258)	Staff provides EDO with update on status of action plan (page 8 of Ref. 1)	05/17/01 (C) ML011720125	DLPM R. Ennis	
1.14 (MB7842/3)	Staff completes review and issues safety evaluation on pilot plant application (NEI 97-06, TG: 2, 3, 4, 7)	TBD Note 12	DE E. Murphy	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
1.15 (MB0631)	Hold steam generator workshop with stakeholders (page 2 of Ref. 1; page 2 of Ref. 2)	02/27/01 (C) ML010820457	DE R. Rothman	
1.16	Staff completes review of generic package and issues model SE for TSTF in *FR* for public comments (NEI 97-06)	TBD (T) Note 12	DRIP K. Kavanagh	DE E. Murphy
1.17	Publish Notice of Availability of TSTF in *FR* (NEI 97-06)	TBD (T) Note 12	DRIP K. Kavanagh	
1.18 (MA4265)	Staff briefs the Commission on regulatory framework (NEI 97-06, and WITS Item 199400048)	05/29/03 (C)	K. Karwoski	
1.19	Issue generic communication related to steam generator operating experience and status of steam generator issues	10/31/01 (C) ML020230299	DE Z. Fu	
1.20 (MA4265)	Staff issues a Commission Paper on regulatory framework (NEI 97-06, and WITS Item 199400048)	05/16/03 (C) ML023540491	DE L. Lund	
1.21 (MC2470)	Staff issues a Generic Letter requesting PWR licensees to address adequacy of their technical specifications to ensure tube integrity between inspections and how bending loads are assessed in their tube integrity evaluations	03/31/05 (T) Note 12	DE L. Lund	
2.1	Evaluate the need for a new communication protocol with the U.S. Secret Service that would cover emergency situations at all NRC licensed facilities (Attachment 3 of Ref. 1)	12/05/00 (C) ML010460485 ML010820457	IRO F. Congel	
2.2 (MB0258)	Establish NRC web site for Steam Generator Action Plan	01/16/01 (C) ML010820457	DLPM R. Ennis	

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. Klingler	
2.6 (MB2446) (MB3794)	Incorporate experience gained from the IP2 event and the SDP process into planned initiatives on risk communication and outreach to the public (TG: 9) 1. Issue NRR input for incorporation into OEDO initiative 2. Address SRM dated 12/26/01	 01/31/02 (C) ML020590125 12/24/02 (C) ML023440202	PMAS M. Kotzalas	
2.7 (MB0258)	Investigate possibility of establishing protocol with OIG regarding review of draft reports for factual/contextual errors (page 8 of Ref. 1)	06/18/01 (C) ML011720125	DLPM R. Ennis	
2.8 (MB0633)	Review and revise, as appropriate, the amendment review process, including concurrence responsibilities, supervisory oversight, and second-round requests for additional information.			

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
2.8 (continued)	a. Issue OI LIC-101 b. Issue procedure for NRR and RES interactions (Attachment 3 of Ref. 1; TG: 6b, 6d, 6e; page 6 of Ref. 1)	08/31/01 (C) 02/27/02 (C) ML020580484	DLPM M. Banerjee DLPM M. Fields	
3.1 (MB7216)	In order to address ACRS comments on current risk assessments, develop a better understanding of the potential for damage progression of multiple steam generator (SG) tubes due to depressurization of the SGs (e.g., during a main steam line break (MSLB) or other type of secondary side design basis accident). (Pgs. 46, 8-12) (See Notes 4, 5, and 6) Specific tasks include:			
	a) Perform thermal-hydraulic (T-H) calculations and sensitivity studies using the 3-D hydraulic component of TRAC-M to assess the loads on the tube support plate and SG tubes during main steam line break (MSLB). Perform sensitivity studies on code and model parameters including numerics. Develop conservative estimate of loads and evaluate against similar analyses.	12/31/02 (C) ML023650132	RES W. Krotiuk	DSSA W. Jensen
	b) Perform T-H assessment of flow-induced vibrations during MSLB. Using the T-H conditions calculated during the transient, generate a conservative estimate of flow-induced vibration displacement and frequency assuming steady state behavior.	12/31/02 (C) ML023650132	RES W. Krotiuk	DSSA W. Jensen

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.1 (continued)	c) *Perform additional sensitivity* studies as needed.	06/30/03 (C)	RES W. Krotiuk	SSA W. Jensen
	d) Obtain information from existing analyses related to loads and displacements (axial, bending, cyclic) experienced by SG structures under MSLB conditions.	12/31/02 (C) ML030230822 Non-public	RES J. Muscara	
	e) Using information from tasks 3.1a, 3.1b, and 3.1d, estimate upper bound loads and displacements.	12/31/02 (C) ML030230822 Non-public	RES J. Muscara	DE E. Murphy
	f) Estimate crack growth, if any, for a range of crack types and sizes using bounding loads from task 3.1e in addition to the pressure stresses. Include the effects of TSP movement in these evaluations and any effects from cyclic loads.	12/31/02 (C) ML030230822 Non-public	RES J. Muscara	DE E. Murphy
	g) Estimate the margins to crack propagation for a range of crack sizes for MSLB types loads and displacements in addition to the pressure stress.	12/31/02 (C) ML030230822 Non-public	RES J. Muscara	DE E. Murphy
	h) Based on the margins calculated in task 3.1g over and above the bounding loads, decide if more refined TH analyses need to be conducted to obtain forces and displacements of structures under MSLB conditions.	12/31/02 (C) ML030230822 Non-public	RES J. Muscara	DE E. Murphy

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.1 (continued)	i) Conduct tests of degraded tubes under pressure and with axial and bending loads to validate the analytical results from above tasks.	06/30/03 (C) ML032080002 (Non-public)	RES J. Muscara	DE E. Murphy
	j) Conduct analyses similar to above with refined load estimates if necessary.	06/30/04 (C) ML042720174	RES J. Muscara	DE E. Murphy
	k) Use information developed in tasks 3.1a through 3.1j to evaluate the conditional probabilities of multiple tube failures for appropriate scenarios in risk assessments for SG tube alternate repair criteria (ARC).	02/28/05 (T)	DSSA S. Long	DE E. Murphy RES J. Muscara H. Woods
3.2	Confirm that damage progression via jet cutting of adjacent tubes is of low enough probability that it can be neglected in accident analyses. (Pgs. 10-11) (See Notes 3 and 5) Specific tasks include:			
	a) Complete tests of jet impingement under MSB conditions.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
	b) Conduct long duration tests of jet impingement under severe accident conditions.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
	c) Document results from tasks 3.2a and 3.2b.	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
3.3 (MB7216)	When available, use data from the ARTIST program (planned in Switzerland) to develop a better model of the natural mitigation of the radionuclide release that could occur in the secondary side of the SGs. (Pgs. 12-13) (See Notes 3 and 5)	09/30/05 (T) See Note 2	RES R. Lee	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (MB7216)	In order to address ACRS criticism of current risk assessments, develop a better understanding of RCS conditions and the corresponding component behavior (including tubes) under severe accident conditions in which the RCS remains pressurized. (Pgs. 46-47, 12-15) (See Notes 3 and 5) Specific tasks include:			
	a) Perform system level analyses to assess the impact of plant sequence variations (e.g., pump seal leakage and SG tube leakage).	09/28/01 (C) ML012720004	RES C. Tinkler	DSSA W. Jensen S. Long
	b.1) Re-evaluate existing system level code assumptions and simplifications.	04/12/02 (C)	RES D. Bessett	DSSA W. Jensen S. Long
	b.2) Following the results from 3.4.a and 3.4.b.1, perform additional analysis to: include modeling of heat transfer enhancement from radiation heat transfer in the hot leg and steam generator; suppress unphysical numerically driven flows in the calculations; and investigate the sensitivity of calculated results to bypass flows and other key parameters.	04/01/04 (C) ML040910022 (Non-public)	RES C. Boyd	DSSA W. Jensen
	c) Examine 1/7 scale data to assess tube to tube temperature variations and estimate variations for plant scale.	08/31/02 (C)	RES D. Bessett	DSSA W. Jensen S. Long
	d) Perform more rigorous uncertainty analyses with system level code to address the uncertainty caused by key governing parameters. Distribution functions will be developed for key parameters. Peer review.	03/31/05 (T) Note 13	RES C. Boyd	DSSA W. Jensen S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	e) Examine SG tube severe accident T-H conditions using computational fluid dynamics (CFD) methods. This includes the following:			
	e.1) Benchmark CFD methods against 1/7 scale test data.	08/31/01 (C) NUREG 1781 ML033140399	RES C. Boyd	DSSA W. Jensen S. Long
	e.2) Perform full scale plant calculations (hot leg and SG) for a 4 loop Westinghouse design. Evaluate scale effects.	03/28/02 (C) NUREG 1788 ML041820075 (Non-public)	RES C. Boyd	DSSA W. Jensen S. Long
	e.3) Perform plant analysis to address the effects on inlet plenum mixing resulting from tube leakage and hot leg orientation (CE design impact).	12/30/02 (C) NUREG 1788 ML041820075 (Non-public)	RES C. Boyd	DSSA W. Jensen S. Long
	f) Examine the uncertainty in the T-H conditions associated with core melt progression.	03/31/05 (T) Note 13	RES C. Boyd	DSSA W. Jensen S. Long
	g) Perform experiments to develop data on inlet plenum mixing impacts due to SG tube leakage and hot leg/ inlet plenum configuration.	03/31/03 (C) See Note 15	RES D. Bessett	DSSA W. Jensen S. Long
	h) Perform a systematic examination of the alternate vulnerable locations in the RCS that are subject to failure due to severe accident conditions. This includes the following:			
	h.1) Evaluate the creep failure of primary system passive components such as pressurizer surge line and the hot leg taking into account the material properties of the base metal, welds, and heat affected zones in the presence of residual and applied stresses, in addition to the pressure stress, and the presence of flaws.	TBD (T) See Note 18	RES J. Page	DE E. Murphy DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.4 (continued)	h.2) Evaluate the failure of active components such as PORVs, safety valves, and bolted seals based on operability and "weakest link" considerations for these components.	TBD (T) Note 18	RES J. Page	DE E. Murphy DSSA S. Long
	h.3) Conduct large scale tests if needed.	11/30/05 (T)	RES J. Page	DE E. Murphy DSSA S. Long
	i) Use existing international data and develop analyses for predicting leak rates of degraded tubes in restricted areas under design basis and severe accident conditions.	05/28/04 (C) ML042720174	RES J. Muscara	DSSA S. Long DE E. Murphy
	j) Put the information developed in task 3.4i into a probability distribution for the rate of tube leakage during severe accident sequences, based on the measured and regulated parameters for ARCs applied to flaws in restricted places (e.g., drilled-hole TSPs and the unexpanded sections of tubes in tube sheets).	TBD (T) Note 17	DSSA S. Long	DE E. Murphy RES J. Muscara
	k) Integrate information provided by tasks 3.4a through 3.4j and 3.5 to address ACRS criticisms of risk assessments for ARCs that go beyond the scope and criteria of GL 95-05 (e.g., ARCs that credit "indications restricted against burst") as well as dealing with other SG tube integrity and licensing issues (e.g., relaxation of SG tube inspection requirements).	TBD (T) Note 17	DSSA S. Long	DE E. Murphy RES J. Muscara C. Boyd H. Woods

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.5 (MB7216)	Develop improved methods for assessing the risk associated with SG tubes under accident conditions. (Pgs. 47, 16-20) (See Note 5) Specific tasks include:			
	a) Development of an integrated framework for assessing the risk for the high-temperature/high-pressure accident scenarios of interest.	04/01/02 (C) ML020910624	RES H. Woods	DSSA S. Long
	b) Issue report describing improved methods and appropriate treatment of uncertainty for identifying severe accident scenarios that lead to challenges of the reactor coolant pressure boundary.	06/28/03 (C) ML031810770	RES H Woods	DSSA S. Long
	c) Develop logic framework for improved PRA models of the scenarios identified above, including the impact of operator actions.	04/06/04 (C) ML041400397 (Non-public)	RES H. Woods	DSSA S. Long
	d) Using the 3.5(b) methods and (c) logic framework, identify scenarios, calculate the frequency of containment bypass events at an example plant, make indicated method improvements, and document the improved methods and results.	TBD (T) See Note 16	RES H. Woods	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.5 (continued)	e) Extend the 3.5(b) methods and (c) model logic to include CE plants, and document them.	TBD (T) See Note 16	RES H. Woods	DSSA S. Long
	f) Extend the 3.5(b) methods and (c) model logic to include consideration of external events as initiators, and low power and shutdown as initial conditions, and document them.	TBD (T) See Note 16	RES H. Woods	DSSA S. Long
	g) Extend the 3.5(d), (e), and (f) improved methods and logic to include consideration of core damage sequences initiated by secondary depressurization events (such as MSLB design basis accident scenarios) that induce tube rupture.	TBD See Note 16	RES H. Woods	DSSA S. Long
3.6	To address an ACRS report conclusion that improvements can be made over the current use of a constant probability of detection (POD) for flaws in SG tubes, RES has recently completed an eddy current round robin inspection exercise on a SG mock-up as part of NRC's research to independently evaluate and quantify the inservice inspection reliability for SG tubes. This research has produced results that relate the POD to crack size, voltage, and other flaw severity parameters for stress corrosion cracks at different tube locations using industry qualified teams and procedures. Complete analysis of research results and prepare topical report to document the results. (Pgs. 47, 33)	12/31/01 (C) ML021910311	RES J. Muscara	DE E. Murphy
3.7 (MB7216)	Assess the need for better leakage correlations as a function of voltage for 7/8" SG tubes. (Pgs. 48, 28-29) (See Note 5)	04/26/03 (C) ML031150674	DE J. Tsao	RES J. Muscara
3.8	Develop a program to monitor the	01/03/02 (C)	DE	

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.9 (MB7216)	Develop a more technically defensible position on the treatment of radio nuclide release to be used in the safety analyses of design basis events. (Pgs. 48, 38-44) (See Note 5) Specific tasks include: a) Assess Adams and Atwood and Adams and Sattison spiking data with respect to the ACRS comments. b) Based upon the assessment performed in task 3.9a, develop a response to the ACRS comments. c) Publish in the *Federal Register* for public comment, the response to ACRS' comments. d) Complete review of public comments. e) Based upon task 3.9d, determine if additional work needs to be performed.	 08/09/01(C) TBD (T) Note 11 TBD (T) Note 11 TBD (T) Note 11 TBD (T) Note 11	DSSA M. Hart	
3.10 (MB7216)	To address concerns in the ACRS report regarding our current level of understanding of stress corrosion cracking, the limitations of current laboratory data, the difficulties with using the current laboratory data for predicting field experience (crack initiation, crack growth rates), and the notion that crack growth should not be linear with time while voltage growth is, the following tasks will be performed: (Pgs. 20-29) (See last sentence in Note 3)			

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

Notes:

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

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

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

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

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

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

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

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

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

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

11. The scope of the work is being re-evaluated. In SECY-04-0156, dated August 27, 2004, Iodine Spiking Phenomena was identified as candidate generic safety issue (GSI) 197 with the Office of Nuclear Regulatory Research (RES) listed as the lead organization. Initial screening of the candidate GSI is to be completed by January 2005.

12. The NRC received the steam generator license amendment submittal for a lead plant (Catawba) on February 25, 2003, and the generic submittal as a Technical Specification Task Force (TSTF) Traveler on March 14, 2003. Based on staff comments, the Catawba submittal was revised on July 30, 2003. By letter dated September 9, 2003, the industry submitted a revised TSTF package to be consistent with the July 30, 2003 submittal for Catawba. The staff issued an RAI to Catawba on September 17,

2003. Based on interactions with the licensee on this RAI, the industry embarked on an effort to develop a methodology for plants to use in determining when bending loads are significant and for determining the structural limit for flawed tubes when bending loads are significant.

Since September 2003, the industry has been working on this issue. After several meetings with the industry, the staff reached a tentative agreement on the wording of the structural integrity performance criterion. Catawba, the lead plant, application review is ongoing, while a similar TS change request by Farley was approved on September 10, 2004. The staff expects to issue the generic safety evaluation six months after receipt of the final generic submittal (which the staff expects to receive near the completion of the Catawba review).

The industry is currently developing/documenting a methodology to address the effects of bending on flawed steam generator tubing and the results are expected in October 2004. Plants with newer steam generators, including the pilot plant with 600TT or 690TT tubing (and Farley with 690TT tubing), are expected to have no problem adopting new technical specifications while the industry work is on-going (since the lack of tube cracking in these plants make the effects of bending a non-issue in the near-term). In the meantime, the staff issued a draft generic letter (GL) in the *Federal Register* for public comments in October 2004, that requests licensees (1) to discuss the adequacy of their steam generator tube integrity program and their plans for modifying their technical specifications (TSs) to ensure they are representative of their program in ensuring tube integrity, which should result in licensee's proposing revisions to their TSs that reflect the guidance in NEI 97-06 and (2) to discuss how bending loads are assessed in their evaluations of tube integrity. The licensees that have adopted the new versions of the TS will not be required to respond to the GL.

13. This task has been delayed so that resources could be used to address emerging issues raised by the PRA analysis. In addition, contractor resources have been temporarily prioritized towards completion of more critical time sensitive work.

14. Note 14 no longer exists.

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

16. Lessons learned from the work completed so far necessitated several modifications and additions to the tasks, milestones, and target completion dates that are being formalized in the RES operating plan and in this SG Action Plan. Scheduled completion date for item 3.5.d thru g will be provided when the present workscope is expanded.

17. The results from this item feed into the task for calculating the severe accident induced steam generator containment bypass probabilities. New completion dates need to be developed based on scheduled completion of 3.4 and 3.5 milestones.

18. Additional analyses will need to be performed to support the development of a robust probabilistic risk assessment.

Description: Steam generator tube integrity issues continue to arise. As a result, many organizations within the NRC have evaluated portions of the regulatory process associated with steam generator tube integrity and have made some insightful observations and/or recommendations. To ensure safety from a steam generator tube integrity standpoint is maintained, that public confidence in the steam generator tube integrity area is improved, and the NRC and stakeholder resources are effectively and efficiently utilized, the steam generator action plan was developed. The action plan is intended to direct and monitor the

NRC's effort in this area and to ensure the issues are appropriately tracked and dispositioned. The action plan is also intended to ensure the NRC's efforts result in an integrated steam generator regulatory framework (license review, inspection and oversight, research, etc.) which is effective, efficient, and realistic.

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

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

Historical Background: The NRC originally planned to develop a rule pertaining to steam generator tube integrity. The proposed rule was to implement a more flexible regulatory framework for steam generator surveillance and maintenance activities that allows a degradation specific management approach. The results of the regulatory analysis suggested that the more optimal regulatory approach was to utilize a generic letter. The NRC staff suggested, and the Commission subsequently approved, a revision to the regulatory approach to utilize a generic letter. In SECY-98-248, the staff recommended to the Commission that the proposed GL be put on hold for 3 months while the staff works with NEI on their NEI 97-06 initiative. In the staff requirements memorandum dated December 21, 1998, the Commission did not object to the staff's recommendation. In late 1998 and 1999 the NRC and industry addressed NRC technical and regulatory concerns with the NEI 97-06 initiative, and on February 4, 2000, NEI submitted the generic licensing change package for NRC review. The generic licensing change package included NEI 97-06, Revision 1, proposed generic technical specifications, and a model technical requirements manual section. SECY-00-0078 outlines the staff's proposed review process associated with the revised steam generator tube integrity regulatory framework described in NEI 97-06. This review process was subsequently revised as described in SECY-03-0080 (see Note 12).

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

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

Current Status:

- November 1, 2000	Issuance of "Indian Point 2 Steam Generator Tube Failure Lessons-Learned Report" via memorandum from W. Travers to the Commission (Accession No. ML003765272).
- November 3, 2000	Issuance of "Staff Review of OIG Report on the NRC's Response to the Steam Generator Tube Failure at Indian Point 2 and Related Issues" via memorandum from W. Travers to the Commission (Accession No. ML003753067).
- November 16, 2000	Issuance of "Steam Generator Action Plan" via memorandum from B. Sheron/J. Johnson to S. Collins (Accession No. ML003770259).

- February 1, 2001	ACRS Ad Hoc Subcommittee report related to SG DPO issued (NUREG-1740).
- May 11, 2001	Issuance of a memorandum providing a revision to the SG Action Plan to address the issues related to the DPO on SG tube integrity issues (Accession No. ML011300073).
- August 2, 2001	Issuance of a letter to NEI transmitting a draft NRC paper on NEI 97-06 SG generic change package (Accession No. ML012200349).
- September 26, 2001	Staff briefing of ACRS subcommittee on Materials and Metallurgy regarding SG action plan status.
- September 26, 2001	Staff briefing of ACRS Subcommittee on Materials and Metallurgy on SG action plan.
- October 4, 2001	Staff briefing of ACRS full-committee on SG action plan status.
- October 18, 2001	ACRS letter to the Chairman documenting their comment on staff action plan to address the SG DPO (ML012960166).
- November 29, 2001	Staff briefing of ACRS Subcommittee on Materials and Metallurgy on NEI 97-06.
- December 3, 2001	Staff briefing of the Commission on the status of SG action plan.
- December 06, 2001	Staff briefing of ACRS on NEI 97-06.
- September 9, 2002	Issuance of a letter to NEI transmitting staff comments on the draft generic license change package (ML022520413)
- February 25, 2003	Duke Power submits lead plant (Catawba) SG technical specification amendment application.
- March 14, 2003	NEI submits TSTF-449, Revision 0, SG Program Generic License Change Package.
- May 16, 2003	Issuance of SECY-03-0080, "Steam Generator Tube Integrity (SGTI) - Plans for Revising the Associated Regulatory Framework."
- May 29, 2003	Staff briefing of the Commission on the status of SG Regulatory Framework Modifications. An industry briefing preceded the staff briefing.
- September 4, 2003	Public meeting between NRC, Duke Power, and NEI on lead plant submittal.
- February 3-5, 2004	Staff briefing of the joint ACRS Subcommittee on Materials/Metallurgy and Thermal/Hydraulics, and the Full Committee on SG DPO related action items.
- May 21, 2004	ACRS letter to the EDO documenting their comment on staff action plan to address the SG DPO (ML041420237).
- August 25, 2004	Response to ACRS from the EDO on their comments on staff action plan to address the SG DPO (ML042400055)

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

Last Update: 09/30/04
Lead Division: DLPM
Supporting Divisions: DE, DSSA, DIPM, & DRIP
Supporting Offices: RES & Regions

TAC No.	Description
MB2916	Non plant-specific activities for Bulletin 2001-01
MB3567	VHP Action Plan (Coordination and Administration)
MB3954	Development of CRDM NUREGs (Bulletin 2001-01)
MB4495	Lead PM Activities for Bulletin 2002-01
MB4603	Non plant-specific activities for Bulletin 2002-01
MB5465	Lead PM Activities for Bulletin 2002-02
MB6218	Inspection TI for Bulletin 2002-02
MB6220	Review of NEI/MRP Crack Growth Rate Report (MRP-55)
MB6221	Development of Alternate (to ASME Code) RPV Head and VHP Inspection Requirements
MB6222	Review of NEI/MRP RPV Head and VHP Inspection Plan (MRP-75)
MB7182	Orders for Interim Inspection Guidelines
MB9522	Review of Bulletin 2002-01 Responses
MB8915	Generic Activities for Lower Head Inspection
MB9891	Develop Bulletin 2003-02
MC0590	Develop Technical Issues Related to Incorporating RCPB Inspection Requirements into 50.55a
MC1036	Develop/Revise Inspection Guidance for ISI and BACC

Milestone	Date (T=Target) (C=Complete)	Lead	Support
Part I - Reactor Pressure Vessel Head Inspection Requirements			
1. Collect and summarize information available worldwide on Alloy 600, Alloy 690 and other nickel based alloy nozzle cracking for use in evaluation of revised inspection requirements. [LLTF 3.1.1(1)-High]	03/04 (C) ML040920026	RES/DET	DE
2. Critically evaluate existing SCC models with respect to their continuing use in the susceptibility index. [LLTF 3.1.4(1)-Medium]	07/03 (C) ML032461221 ML032461224	RES/DET	DE

Milestone	Date (T=Target) (C=Complete)	Lead	Support
3. a. Complete initial evaluation of individual plant inspections in response to Bulletins and Orders. b. Continue to review future inspection results until permanent guidelines are issued.	05/04 (C) ML041560306 Ongoing	DE DE	DLPM Regions DLPM Regions
4. Incorporate Order EA-03-009 requirements into 10 CFR 50.55a 1. Develop technical basis 2. Develop rulemaking plan 3. Commission decision	Note (2) 04/04 (C) ML040920628 ML040920638 07/04 (C) 08/04 (C) ML042190072	 DE DRIP	 DRIP DE
5. Monitor and provide input to industry efforts to develop revised RPV Head inspection requirements (ASME Code Section XI). [LLTF 3.3.4(8)-High	TBD Note (1)	DE	RES/DET DSSA Regions Industry
6. Participate in meetings and establish communications with appropriate stakeholders (e.g., MRP, ASME). [LLTF 3.3.4(8)-High]	Ongoing	DE	RES/DET DLPM DRIP DSSA industry
7. Review and evaluate revised ASME Code requirements when issued. [LLTF 3.3.4(8)-High]	TBD Note (1)	DE	RES/DET
8. If revised ASME Code requirements are acceptable, establish schedule to incorporate by reference into 10 CFR 50.55a. [LLTF 3.3.4(8)-High]	TBD Note (1)	DE	DRIP DIPM DSSA RES/DET industry public
9. Publish a NUREG report summarizing findings from Part I, items 1 and 2, and Part II, item 1.	03/05 (T)	RES/DET	DE
10. Propose a course of action and implementation schedule to address the results of the analysis of Part I, item 1, and Part II, item 1. [LLTF 3.1.1(1)-High]	10/04 (T)	DE	RES/DET

Milestone		Date (T=Target) (C=Complete)	Lead	Support
Part II - Boric Acid Control				
1.	Collect and summarize information available worldwide on boric acid corrosion of pressure boundary materials for use in evaluation of revised inspection requirements. [LLTF 3.1.1(1)-High]	10/04 (T)	RES/DET	DE
2.	a. Evaluate individual plant responses to Bulletin 2002-01 regarding Boric Acid Inspection Programs (60-day responses and necessary follow-up)	06/03 (C) ML031760568	DE	DLPM
	b. Issue public document to summarize evaluation of plant responses.	07/03 (C) ML032100653	DE	DLPM DRIP
3.	Participate in meetings and establish communications with appropriate stakeholders (e.g.,MRP, ASME).	Ongoing	DE	RES/DET DLPM DRIP DSSA industry
4.	Evaluate need to take additional regulatory actions and determine appropriate regulatory tool(s).	06/03 (C) ML031760568	DE	DLPM DRIP DIPM DSSA Regions
5.	Issue Bulletin 2003-02 on Reactor Vessel Lower Head inspection	08/03 (C) ML032320153	DE	DLPM
6.	Develop milestones for additional regulatory actions, as necessary.	07/03 (C)	DE	DLPM DSSA DRIP
7.	Complete and evaluate the results of ongoing research on materials degradation, engage external stakeholders and develop a plan to implement a proactive approach to manage degradation of the RCPB.	06/06 (T)	DE	RES
8.	Review and evaluate the adequacy of revised ASME Code Requirements for Pressure Testing/Leakage Evaluation being developed by the ASME Code, Section XI, Task Group on Boric Acid .	01/05 (T) Note (1)	DE	RES/DET

Milestone	Date (T=Target) (C=Complete)	Lead	Support
Part III - Inspection Programs			
1. Develop inspection guidance or revise existing guidance to ensure that VHP nozzles and the RPV head area are periodically reviewed by the NRC during licensee ISI activities. [LLTF 3.3.4(3)-High]	06/04 (C)	DIPM	DE Regions
2. Develop inspection guidance that provides for timely, periodic inspection of PWR plant BACC programs. [LLTF3.3.2(1)-High]	06/04 (C)	DIPM	DE Regions
3. a. Develop inspection guidance for assessing the adequacy of PWR plant BACC programs (implementation effectiveness, ability to identify leakage, adequacy of evaluation of leaks). [LLTF 3.2.2(1)-High]	06/04 (C)	DIPM	DE RES/DET Regions
b. Perform follow-up evaluation of inspection guidance and licensee program acceptability after conducting inspections for approximately one year.	05/05 (T)	DIPM	DE RES/DET Regions

Notes: (1) Milestone dates are dependent upon issuance of industry proposals.

(2) Requirements for inspection of only the upper head will be the subject of this rulemaking.

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

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

One of the NRC follow-up actions to the Davis-Besse event was formation of a Lessons Learned Task Force (LLTF). The LLTF conducted an independent evaluation of the NRC's regulatory processes related

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

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

The Part I milestones deal with development of improved inspection requirements for the RPV head and VHP nozzles. Interim inspection guidelines for the RPV upper head have been issued via Order EA-03-009 and associated temporary inspection guidelines (TI-150) have been issued for use by NRC inspectors. These will be updated as needed based on inspection results. The staff will monitor and assess the adequacy of revisions to the ASME Boiler and Pressure Vessel Code regarding RPV head inspection, which will be based on the inspection program developed by the EPRI MRP. If the revised ASME Code requirements are acceptable, based on the staff's technical evaluations, the NRC will initiate action to incorporate them by reference in a revision to 10 CFR 50.55a.

The Part II milestones evaluate whether industry BACC programs are meeting NRC expectations and whether additional inspection guidance should be issued. First, the staff will establish a technical basis for BACC program requirements through ongoing and planned research programs. This will include evaluation of boric acid events in past reports and in responses to Bulletin 2002-01, and studies of rates of reactor pressure boundary materials in boric acid solutions. The staff is also monitoring development of revised ASME Code requirements by the Section XI Task Group on Boric Acid. If the staff determines that additional interim guidelines are needed prior to issuance of the revised Code requirements, they will be issued by an appropriate regulatory tool. When the ASME Code requirements are revised, the NRC will initiate action to endorse them, if acceptable. If the revised ASME code requirements cannot be made acceptable to the NRC, then alternate requirements would have to be developed and implemented by an appropriate regulatory tool. Based on the leaks discovered in lower vessel head penetrations at South Texas Project, the staff issued Bulletin 2003-02 regarding RPV lower head inspections. Associated temporary inspection guidelines (TI-152) were issued for use by NRC inspectors. The staff will complete and evaluate the results of ongoing research on materials degradation, engage external stakeholders and develop a plan to implement a proactive approach to manage degradation of the RCPB.

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

Originating Documents:

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

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

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

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

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

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

Current Status: Part I activities included continued monitoring of outage inspection results, follow-up with plants discovering defects, and evaluation of requests for relaxation from First Revised Order EA-03-009.

The staff evaluated the existing SCC models and determined that they are acceptable for use in prioritizing RPV head inspections.

The staff collected information on Alloy 600, Alloy 690 and other nickel-based alloy nozzle cracking and issued a summary report for internal use.

The staff developed a rule plan to incorporate the inspection requirements for the RPV upper head into 10 CFR 50.55a. This was submitted for Commission approval in July 2004. The Commission decided not to proceed with this rulemaking and directed the staff to continue to work with the industry to incorporate revised inspection requirements into the ASME code (SRM-SECY-04-0115, August 6, 2004).

In Part II activities, the review and evaluation of licensee responses to Bulletin 2002-01 regarding BACC have been completed. A summary of the evaluation was published in RIS 2003-13. Based on this review and the discovery of leakage on vessel bottom penetrations at South Texas Project, Bulletin 2003-02 was issued.

The staff has produced a draft report, which is being reviewed, on available worldwide operating experience on boric acid corrosion of pressure boundary materials. Following internal review, this information and the information previously collected on nozzle cracking will be issued in a NUREG. The NUREG will also include the staff evaluation of the SCC models.

Using the information collected on boric acid corrosion and the information previously collected regarding Alloy 600, Alloy 690 and other nickel-based alloy nozzle cracking, the staff is also considering a proposed course of action and an implementation schedule to address LLTF 3.1.1(1).

In Part III activities, inspection procedure revisions addressing RPV head inspection and boric acid corrosion control programs were issued.

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	Terrence Reis, IROB, 415-3281

References:
First Revised Order EA-03-009 establishing interim inspection requirements for reactor pressure vessel heads at pressurized water reactors, February 20, 2004.

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

NRC Regulatory Issue Summary 2003-13, "NRC Review of Responses to Bulletin 2002-01."

Order EA-03-009 establishing interim inspection requirements for reactor pressure vessel heads at pressurized water reactors, February 11, 2003.

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

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

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

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

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

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

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

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

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

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

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

Table 1
LLTF Report Recommendations Included in SCC Action Plan

High Priority

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

Medium Priority

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

Low Priority

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

PWR SUMP PERFORMANCE

TAC Nos. MA6454, MA2452, MA4014, MA0704, M95473, MA6204, MA0698, MB4047, MB6411, MB3103, MB8052, MB7776, MB9470, MB4864, MB9931, MC0307, MC1154, MB5625, MB4865, MC0725/6, MB5221, MB5964, MB6589, MB7228, MC1627, MB5334, MC2628, MB6946 MB9549, MC4272

Last Update: 09/30/04
Lead NRR Division: DSSA
Supporting Divisions: DE, DRIP, DLPM, 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"	
o Complete review of licensee responses	03/00 (C)
o Complete revision of RG 1.1/RG 1.82, R3	11/03 (C)
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	
o Preliminary (qualitative) risk assessment (NRR)	03/99 (C)
o Complete collection of plant data to support research program	06/99 (C)
o Integrate industry activities into this Action Plan	04/00 (C)
o Complete research program on PWR sump blockage	09/01 (C)
o Evaluate need for regulatory action based on research program results (NRR)	03/02 (C)

MILESTONES	DATE (T/C)
2. Resolve ECCS suction clogging issue for PWRs (Regulation/Guidance Development and Issuance Stages of GSI process in MD 6.4))	
o Brief NRR ET to obtain approval to prepare a generic letter (GL)	02/02 (C)
o Public meeting with NEI, WOG, B&WOG, CEOG	03/02 (C)
o ACRS Briefing on proposed draft GL	02/03 (C)
o CRGR Briefing on proposed Bulletin 2003-01	04/03 (C)
o Information Paper to Commission, Issue Bulletin 2003-01	06/03 (C)
o NEI publish PWR Industry Evaluation Guidelines (Draft)	10/03 (C)
o CRGR Briefing on proposed draft GL	02/04 (C)
o Proposed draft GL issued for Public Comment	03/04 (C)
o Send Information Paper to Commission regarding GL issuance	08/04 (C)
o Issue Safety Evaluation on Methodology	10/04 (T)
o NRC starts Reviews of GL Responses and Selective Audits	06/05 (T)
o Licensees start modifications, if needed, using approved guidelines	04/06 (T)
o NRC closes GSI-191	12/07 (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; the first three have been completed. First, for boiling-water reactors (BWRs), this issue has been addressed by licensee responses to NRCB 96-03. Second, the adequacy of licensee (both PWR and BWR) net positive suction head (NPSH) calculations was evaluated through NRR review of licensee responses to GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," dated October 7, 1997. The third part of the plan assessed the adequacy of the implementation and maintenance of 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 remaining part of the action plan is an evaluation of the potential for clogging of PWR ECCS recirculation sumps during a LOCA. RES completed its assessment of the potential for debris clogging to support the resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance." RES performed a parametric evaluation to demonstrate whether sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation was ill-suited for determining whether sump blockage will impede or prevent long-term recirculation at a specific plant. By memorandum dated September 28, 2001, RES transferred the lead for GSI-191 to NRR.

Historical Background: During licensing of most domestic power plants, consideration of the potential for loss of adequate NPSH due to blockage of the ECCS suction by debris generated during a LOCA was inadequately addressed by both the NRC and licensees. The staff first addressed ECCS clogging issues in detail during its review of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." The NRC staff's concerns related to the potential loss of post-LOCA recirculation capability due to insulation debris were discussed in GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," dated December 3, 1985. This generic letter documented the NRC's resolution of USI A-43. The staff concluded at that time that no new requirements would be imposed on licensees; however, the staff did recommend that Regulatory Guide 1.82, Revision 1, "Water

Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," be used as guidance for the conduct of 10 CFR 50.59 reviews dealing with change out and/or modification of thermal insulation installed on primary coolant system piping and components. NUREG-0897, Revision 1, "Containment Emergency Sump Performance" (October 1985), contained technical findings related to USI A-43, and was the principal reference for developing the revised regulatory guide.

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

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

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

NRCB 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," was issued on May 6, 1996, requesting BWR licensees to implement appropriate procedural measures and plant modifications to minimize the potential for clogging of ECCS suction strainers by

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

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

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

The NRC has developed web pages to keep the public informed of regulatory and research activities related to PWR sump performance:

http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance.html

These web pages provide links to information regarding NRC interactions with industry (industry submittals, meeting notices, presentation materials, and meeting summaries) and publically available regulatory and research documents. The NRC will continue to update these web pages as new information becomes available.

Proposed Actions: This action plan involves an evaluation of PWR sumps based on new information learned during the development of the staff's resolution for NRCB 96-03. RES conducted a program to evaluate PWR sump designs and their susceptibility to blockage by debris. This evaluation included a risk assessment. Risk insights support the conclusions drawn relative to the need for licensees to address the potential for ECCS suction clogging. The research program needed plant data to bound the problem to be evaluated. The Nuclear Energy Institute (NEI) conducted a survey of PWR licensees and provided the information needed by RES. The staff is coordinating its work with industry to eliminate duplication of effort

and to ensure effective utilization of resources. RES parametrically evaluated whether sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs.

Originating Document: Not Applicable.

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

The staff believes that there is sufficient new information and concerns raised relative to the potential for debris clogging in PWRs that establish the need to address PWR sump blockage concerns. As noted above, RES's parametric evaluation demonstrated that sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill-suited for making a determination that sump blockage will impede or prevent long-term recirculation at a specific plant. Therefore, it is not clear how significant a threat to PWR ECCS operation exists.

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

As part of the GSI-191 study, RES's contractor, Los Alamos National Laboratory (LANL), performed a generic risk assessment to determine how much core damage frequency (CDF) is changed by the findings of the parametric analysis. Utilizing initiating event frequencies that consider LBB credit consistent with NUREG/CR-5750, LANL calculated an overall CDF of 3.3E-06 when debris clogging as a failure

mechanism is not considered, and an overall CDF of 1.5E-04 when debris clogging is considered. However, these CDFs were calculated without giving any credit for operator action, and without consideration to whether the ECCS or containment spray pumps would be able to continue operating after the headloss across the sump screen exceeds the calculated licensing basis NPSH margin. The change in CDF is also dominated by the small and very small break LOCAs which are events where there are significant operator actions that can be taken to prevent core damage. The risk benefit of certain interim compensatory measures is demonstrated by the NRC-sponsored technical report LA-UR-02-7562, "The Impact of Recovery from Debris-Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," dated February 2003. On this basis, the schedule for issuing generic communications and followon actions to address the PWR sump clogging issue outlined above is considered to be appropriate.

Current Status: The staff continues to hold regular public meetings with the PWR owners groups and NEI sump performance task force on the progress toward resolving GSI-191.

The PWR Industry is implementing a two-step program to assess the current conditions and evaluate sump recirculation performance. The first guidance document, NEI 02-01, "Condition Assessment Guidelines: Debris Sources inside Containment," was published in September 2002. Consistent with the risk significance of the PWR sump-clogging concern, the staff issued Bulletin 2003-01 on June 9, 2003, requesting information on compliance within 60 days or information on interim compensatory measures to reduce risk until an evaluation to determine compliance is completed. The staff has issued RAIs for the bulletin as needed, and is in the process of reviewing licensee's responses and issuing closeout letters. On October 31, 2003, NEI submitted a draft of the second guidance document, "PWR Containment Sump Evaluation Methodology. This document recommends methodologies for evaluating a PWR's susceptibility to sump clogging based upon the information collected in accordance with NEI 02-01. The final version of the baseline guidance was issued on May 28, 2004, with a refinements table and the risk-informed section, Section 6.0, provided to the staff on July 7, 2004, and July 13, 2004, respectively. The staff is preparing a Safety Evaluation Report (SER) which will provide licensees an NRC-approved methodology to complete the site-specific evaluations as required in Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors", which was issued on September 13, 2004, following CRGR endorsement on August 10, 2004. The Generic Letter had previously been presented to the ACRS sub-committee on June 22-23, 2004, and to the ACRS full-committee on July 7-9, 2004. The staff presented the SER to the ACRS T-H Sub-committee on September 22, 2004, the ACRS Full-committee on October 7, 2004, and the CRGR on October 12, 2004. The staff intends to conduct a public meeting on the generic letter and the safety evaluation, and support the NEI workshop on the safety evaluation scheduled for December 2-3, 2004.

NRR Lead PMs: Mark Giles, DSSA/SPLB, 415-2016

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Leon Whitney, DSSA, 415-3081 (Bulletin 2003-01)
David Cullison, SPLB, 415-1212 (Generic Letter)
Mark Kowal, SPLB, 415-1663 (Risk-informed approach)

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References:

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

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

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

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

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

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

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

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

Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated November 2003.

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

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

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

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

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

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

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

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

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

NEI PWR Containment Sump Evaluation Methodology, letter dated May 28, 2004.

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

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

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

Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, dated September 13, 2004.

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

TAC No. MB7245

Last Update: 09/30/04
Lead NRR Division: DSSA
Supporting Division: DLPM
Supporting Office: RES

MILESTONES	DATE (T/C)
1. Transfer GSI from RES to NRR. Issue Resolution Process letter from J. Zwolinski, NRR, to F. Eltawila, RES.	12/2002 (C)
2. Issue Task Action Plan - First draft for issuing Order. Final draft ready for issuing an Order. New draft for Rulemaking.	03/14/03 (C) 04/30/03 (C) 06/30/03 (C)
3. Engage the affected stakeholders: BWROG Management Meeting, ICUG, and NEI.	02/19/03 (C)
4. Review RES and contractor Cost and Benefit Analyses, technical assessment, and supporting/reference material. Conduct additional analyses if required.	02/28/03 (C)
5. Determine best solution and course of action (order, rule making, generic letter, severe accident management guidelines, etc.) Order initially selected.	02/12/03 (C)
6. Prepare regulation and guidance development memoranda and provide results and recommendations to NRR management.	03/05/03 (C) 03/05/03 (C)
7. Brief DLPM Management.	03/06/03 (C)
8. Brief LT and obtain approval for Order.	03/13/03 (C)
9. Distribute Draft Order and draft SECY Letter.	03/26/03 (C)
10. Provide Draft Order to OGC.	03/28/03 (C)
11. Brief ET.	03/19/03 (C)
12. Brief NRR/D.	03/19/03 (C)
13. Draft SECY Letter to EDO.	03/27/03 (C)
14. Finalize CRGR Package.	03/26/03 (C)
Course of action changed per OGC and ET - Will conduct a Public Meeting and pursue Rulemaking	
15. Meet with Rulemaking Committee.	05/05/03 (C)
16. Schedule Public Meeting.	05/14/03 (C)
17. Issue Press Release regarding Public Meeting.	05/29/03 (C)
18. Public Meeting.	06/18/03 (C)

MILESTONES	DATE (T/C)
19. Conduct Post Public Meeting Debrief and determine course of action.	06/18/03 (C)
20. Meet with OPA to develop Communications Plan and Website.	06/24/03 (C)
21. Complete Communications Plan Draft and route for approval.	07/10/03 (C)
22. Meeting with ACRS and Second Public Meeting to address issues regarding design criteria of backup power supply and cost/benefit analysis refinements.	11/06/03 (C)
23. Complete Stage 4, Regulation and Guidance Development of Management Directive 6.4 and enter Stage 5, Regulation and Guidance Issuance.	01/31/04 (C)
24. Public Meetings with BWROG and NEI regarding hydrogen igniter back-up power supply design criteria.	02/03/04 (C) 03/31/04 (C)
25. Commissioner Merrifield brief.	03/04/04 (C)
26. NRR Rulemaking Board approval.	Fall 2004
27. Develop Rulemaking Plan (Action by DRIP, Policy and Rulemaking Program Section). The staff is also discussing to explore other options for GSI-189 resolution.	Fall 2004
28. Draft design criteria issued to the NRR division directors for cognizant NRR staff's comment and approval.	08/13/04 (C)
29. Updated Communication Plan to issue and route for approval.	to be issued
30. Public meeting w/external stakeholders to get stakeholders' input on the draft design criteria and implement of backup power supply to H_2 igniters.	09/21/04 (C)
31. Work on NRR action plans to provide options including rulemaking and industry initiatives and to resolve issues resulting from public meeting.	Fall 2004

Description: Following a severe accident concurrent with station blackout (SBO), the PWR ice condenser containments and BWR Mark III containments are vulnerable to failures from hydrogen (H_2) deflagrations and detonations. To resolve GSI-189, NRR is recommending the addition of a backup power supply for the combustible gas igniters for licensees with Ice Condenser or Mark III containments. The generic issue was proposed in response to SECY 00-198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)." There are 13 susceptible plants involved. They are the 4 dual-unit PWR nuclear stations with ice condenser containments - McGuire, Catawba, DC Cook, and Sequoyah; a single-unit PWR nuclear station with ice condenser containment - Watts Bar; and 4 single-unit BWR nuclear plants with Mark III containments - Grand Gulf, River Bend, Clinton, and Perry.

Historical Background: The generic issue was proposed (Memorandum to John Flack, Chief, Regulatory Effectiveness and Human Factors Branch, Division of Systems Analysis and Regulatory Effectiveness, RES, from Mark Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, RES, "Information Concerning Generic Issue on Combustible Gas Control for PWR Ice Condenser and BWR Mark III Containment Designs," August 15, 2001, ML012330522) in response to SECY-00-198. This SECY paper explored means of making 10 CFR 50.44 risk-informed , and the paper recommended that safety enhancements that have the potential to pass the backfit test be assessed for mandatory application through the generic issue program.

Following a severe accident, the PWR ice condenser and BWR Mark III containments are vulnerable to failures from hydrogen (H2) deflagrations or detonations, because the existing H_2 igniters which are used to prevent hydrogen accumulation in large quantities cannot be energized due to lack of onsite and offsite AC power under SBO conditions.

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

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

The technical assessment quantified the reduction in the conditional containment failure probability associated with combustible gas (H_2) control being available during SBO events, which was then converted to a dollar value based on the expected values for averting public exposure and offsite property damage associated with the availability of combustible gas control. These averted costs (benefits) were then compared to the overall cost for the implementation and maintenance of several alternative safety enhancements to determine if there was a potential cost beneficial back-fit.

The RES analyses were based on consideration of internal events only. However, sufficient information was provided in the RES analyses associated with external events for some of the plants to evaluate the impact external events could have on the analyses. When considering external events, averted costs increase substantially. Though the backup power system would not be required to be designed to withstand the external events that could be precursors of the SBO, it is expected that the small, backup power supply will be located in an area capable of withstanding those external events. However, NRR did not believe that addition of the backup power supply provided a substantial safety enhancement at a justifiable cost. NRR is refining the cost-benefit analysis with the contractor, ICF, for the regulatory analysis.

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

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

For most accident sequences, the hydrogen igniters can deal with the potential threat from combustible gas buildup. In the beyond-design-accident analysis, station blackout was postulated concurrent with a severe accident that would cause significant releases of radioactive material to the environment. The situation of interest for this generic safety issue only occurs during severe accident sequences associated with station blackouts, where the igniter system is not available because they are AC-powered.

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

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

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

NRR staff recommendations were presented to the ACRS on November 6, 2003, citing the results from recent studies which identify a near certainty of containment failure without the use of igniters during this severe accident. The ACRS recommended that NRR pursue upgrading the igniters through Rulemaking, as well as providing guidance via SAMGs or EOPs. NRR recommended that backup power be provided to one train of the hydrogen igniter system and met with the Boiling Water Reactor Owners' Group (BWROG) prior to making a decision to pursue Rulemaking. NRR staff discussed alternatives with the BWROG for the four affected BWR plants.

Proposed Actions: NRR developed draft design criteria for the backup power supply, and discussed with the industry in the public meetings on February 3, and March 31, 2004, to work on draft design criteria. NRR is pursuing rulemaking and voluntary license initiatives as an alternative to rulemaking, and is refining the cost-benefit analysis for the regulatory analysis to determine whether the cost is justifiable for safety enhancement.

Meet with Rulemaking Committee to finalize course of action.

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

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

Regulatory Assessment: Defense-in-Depth - As pointed out in the analyses, NRR technical staff recognized that there are significant uncertainties in both the cost and benefit calculations done by the RES which can shift the benefit to cost from a net negative number to a net positive number. NRR technical staff agreed with RES and ACRS that the defense-in-depth philosophy is applicable for the reason to manage uncertainties.
Backfit Rule - NRR technical staff believes that adding backup power provides a safety enhancement that yields a substantial increase in the overall protection of the public health and safety. NRR is performing additional analyses to determine whether the implementation costs are justified in view of this increased protection, and will perform a backfit analysis to confirm that backfit is justified.

Current Status: NRR staff is currently pursuing rulemaking and voluntary license initiatives as an alternative to rulemaking. Currently, NRR is working on finalizing the design criteria for the backup power supply, and is administering a contract with ICF to merge and enhance the existing technical assessment into a regulatory analysis, and is performing a cost-benefit analysis to support a possible rulemaking effort. The NRR held a public meeting with the public and industry on September 21, 2004, to get stakeholders' input on the design criteria. Representatives of the PWR ice condenser utilities, the BWROG of the BWR Mark III utilities, and the Nuclear Energy Institute (NEI) discussed the proposed design criteria. The representatives of PWR ice condenser containment utilities considered that the draft design criteria are generally acceptable. However, the BWROG representatives stated that the 1-hour time limit is insufficient for the BWR Mark III containment to connect backup power source to the hydrogen igniters without making the system automatic, and manually hookup the backup power source is required. The BWROG is willing to make hardware modifications to supply backup power from the existing HPCS diesel system, and agreed to provide additional information regarding implementation costs and the relative risk contribution from fast-SBO and slow-SBO at each of the Mark III plants. The BWROG requested that NRC provide feedback whether the 2-hour power supply solution is viable.

At the public meeting, Duke power, representing two PWR ice condenser sites, Catawba-1&2, McGuire-1&2, agreed to make modifications on an existing safe-shutdown diesel generator that can manually hookup to the H_2 igniters as needed. AEP representative agreed to provide backup power for D.C. Cook-1&2 from new, large diesel generators which are already planned for installation to support increased allowed outage time. TVA, representing two PWR ice condenser containment sites, Sequoyah-1&2, Watts Bar-1, was willing to provide new design for the backup power supply as the standard emergency power on the 69Kv board. These utilities will provide their proposal to NRC for staff review in the near future.

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NRR Technical Contact:	Bob Palla, DSSA/SPSB, 415-1095
RES Technical Contact:	Allen Notafrancesco, DSARE/SMS, 415-6499

References:
1. SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50.
2. NUREG/CR-4551, Vol. 3, Rev. 1, Part 1, "Evaluation of Severe Accident Risks: Surry Unit 1, Main Report," October 1990.
3. NUREG/CR-4551, Vol. 3, Rev. 1, Part 3, "Evaluation of Severe Accident Risks: Surry Unit 1, External Events," December 1990.
4. NUREG/CR-4551, Vol. 5, Rev. 1, Part 1, "Evaluation of Severe Accident Risks: Sequoyah, Unit 1, Main Report," December 1990.
5. NUREG/CR-4551, Vol. 6, Rev. 1, Part 1, "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1, Main Report," December 1990.
6. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990.
7. Letter from V. Mubayi, Brookhaven National Laboratory, to H. VanderMolen, NRC, "NUREG-1150 Consequence Calculations," July 20, 1994.
8. T. D. Brown *et. al.*, "NUREG-1150 Data Base Assessment Program: A Description of the Computational Risk Integration and Conditional Evaluation Tool (CRIC-ET) Software and the NUREG-1150 Data Base," letter report, March 1995.
9. NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," Final Report, January 1997.
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13. Memorandum to Samuel Collins, Director, Office of NRR, from Ashok Thadani, Director, Office of RES, September 29, 2000, regarding Research Information Letter RIL-0005, "Completion of Research to Address Direct Containment Heating Issue for All Pressurized Water Reactors." (ML003755724).
14. Memorandum to Ashok Thadani, Director, Office of RES, to Samuel Collins, Director, Office of NRR, November 22, 2000, regarding Research Information Letter RIL-0005, "Completion of Research to Address Direct Containment Heating Issue for All Pressurized Water Reactors." ML003761979).
15. NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program, Main Report," Draft Report for Public Comment, April 2001.
16. Memorandum to John Flack, Chief, Regulatory Effectiveness and Human Factors Branch, Division of Systems Analysis and Regulatory Effectiveness, RES, from Mark Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, RES, "Information Concerning Generic Issue on Combustible Gas Control for PWR Ice Condenser and BWR Mark III Containment Designs," August 15, 2001 (ML012330522).
17. Memorandum to M. Snodderly (NRC) from M. Zavisca and M. Khatib-Rahbar (ERI), "Combustible Gas Control Risk Calculations (DRAFT) for Risk-Informed Alternative to Combustible Gas Control Rule for PWR Ice Condenser, BWR Mark I, and BWR Mark III (10 CFR 50.44)," October 22, 2001.
18. Management Directive 6.4 (MD 6.4), "Generic Issues Program," December 4, 2001.
19. Management Directive 6.3 (MD 6.3), "The Rulemaking Process," July 31, 2001.
20. Memorandum from John H. Flack, Chief, REAHFB:DSARE:RES to Jack E. Rosenthal, Chief, SMSAB:DSARE:RES and Mark A. Cunningham, Chief, PRAB:DRAA:RES, dated February 6, 2002, regarding "Panel Review of GSI-189, Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
21. Memo from Farouk Eltawila, Director, RES, to Ashok C. Thadani, Director RES, dated February 13, 2002, regarding RES Task Action Plan for Resolving Generic Safety Issue 189: "Post Accident Combustible Gas Control in Pressure Suppression Containments."
22. Memorandum from William Travers, EDO, to The Commissioners, dated May 13, 2002 (SECY-02-0080), Proposed Rulemaking–Risk Informed 10CFR50.44, "Combustible Gas Control In Containment", (WITS 20010003).

23. Advisory Committee on Reactor Safeguards Meeting Minutes, 493rd Meeting, June 6, 2002, regarding Technical Assessment Generic Safety Issue (GSI)-189.
24. Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment, Rev. 2, September 24, 2002, by Information Systems Laboratories, Inc., Rockville, MD.
25. Hydrogen Control Calculations for the Sequoyah Plant, draft letter report, Rev. 3, September 30, 2002, by Sandia National Laboratories.
26. Memorandum from Ashok Thadani, RES to William Travers, EDO, dated October 1, 2002, regarding, "Revision to NRC's Regulatory Analysis Guidelines [NUREG/BR-0058] and RES Office Letter 1 to Conform to OMB's Information Quality Guidelines."
27. Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice condenser and Mark III Containment Plants, draft letter report, October 4, 2002, by Brookhaven National Laboratory. ADAMS ML022880554.
28. Advisory Committee on Reactor Safeguards Subcommittee on Thermal-Hydraulic Phenomena and Subcommittee on Reliability and Probabilistic Risk Assessment Meeting Minutes, November 5, 2002, regarding Generic Safety Issue (GSI)-189.
29. Advisory Committee on Reactor Safeguards Meeting Minutes, 497th Meeting, November 7, 2002, regarding Technical Assessment Generic Safety Issue (GSI) -189.
30. Memo from George E. Apostolakis, Chairman Advisory Committee on Reactor Safeguards, to the Commission Chairman Richard A. Meserve, dated November 13, 2002, regarding "Recommendations Proposed by the Office of NRR for Resolving Generic Safety Issue -189, Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident. ML023230513
31. Memo from Ashok C. Thadani, Director RES, to Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, dated December 17, 2002, regarding RES Proposed Recommendation for Resolving Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident." ML023510161
32. Attachment to Memo from Ashok C. Thadani, Director RES, to Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, dated December 17, 2002, "Technical Assessment Summary for GSI-189: Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
33. Memo from John A. Zwolinski, Director, Division of Licensing Project Management, NRR to Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, RES, dated January 21, 2003, regarding, "Resolution Process for Generic Safety Issue (GSI) 189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
34. Memo from Jack Rosenthal, Branch Chief, Safety Margins and Systems Analysis Branch, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research to John Hannon, Branch Chief, Plant Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation dated June 19, 2003, regarding, Final Contractor's Reports: Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
35. Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report, Energy Sciences and Technology Department, Brookhaven National Laboratory, December 23, 2002 (ML031700011).
36. Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment, Revision 2, Information Systems Laboratories, Inc., September 24, 2002 (ML031700015).
37. Hydrogen Control Calculations for the Sequoyah Plant, Final Letter Report, March 2003, Prepared By Sandia National Laboratories, March 2003 (ML031700025).

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

TAC No.	Description
MB7280	Develop Operating Experience Action Plan
MB7347	Overall Assessment of Agency's Operating Experience Program
MB8220	Operating Experience Task Force Activities (NRR)
KC0056	Operating Experience Task Force Activities (RES)
MC2066	Operating Experience Task Force Plan Development
MC3378	Operating Experience Program Implementation

Last Update: 09/30/04
Lead Division: DIPM
Supporting Divisions: DE, DSSA, & DLPM
Supporting Offices: RES & Regions

Milestone		Date (T=Target) (C=Complete)	Lead	Support
Part I - Operating Experience Program: Objective Phase				
1.	Form Task Force with Steering Committee and develop Charter.	03/03 (C) ML030900117	NRR/RES	
b.	Identify desirable agency operating experience program objectives and attributes, and	04/03 (C)	Task Force	DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions
2.a.	Provide documented staff proposals of operating experience program objectives and attributes.	04/03 (C) ML031200312 ML031490535		
2.b.	Obtain executive management endorsement.	05/03 (C) ML031350156		
Part II - Operating Experience Program: Assessment Phase				
1.	Define functional needs/areas and processes to meet objectives and attributes.	9/03 (C)	Task Force	DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions

Milestone		Date (T=Target) (C=Complete)	Lead	Support
2.	Review and evaluate current processes. [LLTF 3.1.6(1)] +	11/03 (C) ML033350063	Task Force	DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions
3.	Identify areas for improvements. [LLTF 3.2.4(1)]	11/03 (C) ML033350063	Task Force	DIPM, DLPM, DE, DSSA, DET/RES, DRAA/RES, DSARE/RES, Regions
4.	Task Force issues draft report.	09/03 (C) ML032740058	Task Force	
5.	Task Force provides final report to Steering Committee documenting its specific program improvement proposals.	11/03 (C) ML033350063	Task Force	
6.	Steering Committee sends report back to line management for implementation detail.	01/04 (C) ML040080005	Steering Committee	
6.a	Responsible organizations achieve consensus on proposals to implement.	01/04 (C) ML040560144	NRR/RES	Regions
Part III - Operating Experience Program: Implementation Phase				
1.	Develop plan for program development based on 6.a in Part II.	04/04 (C) ML041180024	NRR/RES	Regions OCIO
1.a	Complete Operating Experience framework (Draft Management Directive/Handbook) [LLTF 3.1.6(2)]	12/04 (T)		
1.b	Other program enhancements: (1) Handling of foreign operating experience [LLTF 3.1.6(3)]	03/03 (C) LIC-401		
	(2) Strengthen inspection guidance [LLTF 3.3.4(2)]	09/03 (C) IP 71152		
2.	Establish processes to monitor effectiveness.	TBD	NRR/RES	Regions
Part IV - Inspection Program Enhancements				
1.	Provide training and reinforce expectations	12/03 (C)	DIPM	DE,

Milestone		Date (T=Target) (C=Complete)	Lead	Support
	to NRC managers and staff members to address the following areas: (1) maintaining a questioning attitude in the conduct of inspection activities; (2) developing inspection insights stemming from the DBNPS event relative to symptoms and indications of RCS leakage; (3) communicating expectations regarding the inspection follow-up of the types of problems that occurred at DBNPS; and (4) maintaining an awareness of surroundings while conducting inspections. Training requirements should be evaluated to include the appropriate mix of formal training and on-the-job training commensurate with experience. Mechanisms should be established to perpetuate these training requirements. [LLTF 3.3.1(1)]			DSSA, DET/RES, Regions
2.	Implement actions to maintain NRC expertise by ensuring that NRC inspector training includes: (1) boric acid corrosion effects and control; and (2) PWSCC of nickel based alloy nozzles. [LLTF 3.3.5(1)]	12/03 (C)	DIPM	DE, DSSA, DET/RES, Regions

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

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

Since this time, both NRR and RES have recognized the need to make operating experience more efficiently available to users. RES has made substantial advances in making existing databases available through the internal web. These databases include licensee event reports (LERs), INPO's EPIX database, and monthly operating reports. RES uses these data to provide initiating event frequencies, safety system reliabilities, component failure probabilities, and common-cause failure parameter estimates, as well as related insights. The RES internal web page, for which significant further advances are already planned,

will allow NRC staff easier and more timely access these estimates, related trends, and insights in a more timely manner. In addition, the RES internal web site will provide a new expanded LER search tool for use by NRC staff. It is planned that in April 2003, the accident sequence precursor (ASP) database will be accessible through the RES internal web site to the NRC staff. In September 2003, this will be followed by an expanded web site that will further integrate presently contained in separate databases and NUREG and NUREG/CR reports. NRR has similarly improved communications of its short term operating experience program outputs through web technology and is currently replatforming its events and assessment database.

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

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

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

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

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

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

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

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

Originating Documents:

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

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

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

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

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

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

Current Status: All Part I (Objective Phase) activities are complete. The Operating Experience Task Force was formed, and completed development of program objectives and attributes, which were endorsed by the Steering Committee.

The Part II (Assessment Phase) activities are complete. The Task Force delivered its draft report to the Steering Committee in September. After incorporating review comments from the Steering Committee, the final report was delivered in November. The Steering Committee sent the report to line management in January with 24 direction setting recommendations for implementation.

The Part III (Implementation Phase) activities are in progress. The plan for program development was completed in April 2004. Development of the Operating Experience framework (draft Management Directive/Handbook) is scheduled for completion by December 2004. Implementation of the new processes by NRR and RES will commence as soon as the framework is approved. Development of detailed operating procedures will be accomplished part of the routine update and improvement process. After reviewing program enhancements instituted to date, it was determined that LLTF recommendations 3.1.6(3) and 3.3.4(2) were adequately addressed and can be considered complete.

Inspection program enhancements in Part IV were completed as scheduled. A web-based training process was initiated, by which inspectors log on and conduct self-paced training. A record of personnel who complete the training is available for management review and follow-up. Training modules on boric acid corrosion and primary water stress corrosion cracking were issued on the system. Also, a training program based on the Columbia shuttle accident, which emphasizes expectations on maintaining a questioning attitude, awareness of surroundings, follow-up to problems, etc., was presented at inspector counterpart meetings and added to the web-based training.

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Regional Offices:	Charles Casto, Region II, 404-562-4600

References:

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

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

Table 1
LLTF Report Recommendations (High Priority)

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

ATTACHMENT 2

GENERIC COMMUNICATION AND COMPLIANCE ACTIVITIES

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

Summary Report as of 9/30/2004

TAC NO.	TAC TITLE	AGE	LEAD ORG
MC2340	BL: Inspection of Primary System Alloy 82/182 Piping Butt Welds (Mitchell-DE/Petrone-DIPM)	7	DE
MC3721	BL: Spent Fuel Rod Accountability (Tuttle-NSIR/Petrone-DIPM)	3	NSIR
MC2590	BL: Ultrasonic Flow Meters (Ahmed-DE/Petrone-DIPM)	6	DE
MC2341	GL: Assmt & Dispos of Impact of PWSCC of Alloy 82/182 Welds on Leak-Before-Break Analyses (Mitchell-DE/Petrone-DIPM)	7	DE
MC2470	GL: Steam Generator Tube Integrity and Associated Technical Specifications (Karwoski-DE/Petrone-DIPM)	7	DE
MC4223	IN: Additional Adverse Effect of Boric Acid Leakage Upset of Post-Accident Coolant Leakage (Hodge-DIPM)	2	DIPM
MC4467	IN: Problems Associated w/ Back-up Pwr Supplies to Emerg Response Facilities & Equipment (Flemming-NSIR/Petrone-DIPM)	1	NSIR
MC3846	RIS: Clarif on Use of Later Editions & Addenda to ASME Sec XI for Repair/Replacement Activities (Tsao-DE/Petrone-DIPM)	3	DE
MC3844	RIS: Clarif Providg Access to Autho Nucl Inservice Insp & NRC Authorized Alter ASME Code Rqmts (Tsao-DE/Petrone-DIPM)	3	DE
MC3977	RIS: Clarifying the Process for Making Emergency Plan Changes (Williams-NSIR/Petrone-DIPM)	2	NSIR
MC3713	RIS: Emergency Preparedness Issues: Post-9/11 (Anderson-NSIR/Petrone-DIPM)	3	NSIR
MC2262	RIS: GL 91-18, Rev 2-Guidance on Operability & Resol of Degraded & Non-Cnfrming Conditions (Kavanagh-DIPM/Petrone-DIPM)	7	DIPM
MC4220	RIS: Guidance for Establishing and Maintaining a Safety Conscious Work Environment (Jarriel-OE/Petrone-DIPM)	2	OE
MC3722	RIS: Operator Medical Issues (Guenther-DIPM/Petrone-DIPM)	3	DIPM
MC3628	RIS: Performance of Manual Actions to Satisfy the Reqmts of 10CFR Part 50, App R, Sec III.G.2 (Klein-DSSA/Petrone-DIPM)	4	DSSA
MC4310	RIS: RIS 2004-003, Rev 1: Risk-Informed Approach For Post-Fire Safe-Shutdown Circuit Insps (Frumkin-DSSA/Markley-DIPM)	1	DSSA

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

Summary Report (7/01/2004 - 9/30/2004)

TAC NO.	TAC TITLE	STATUS	AGE	TAC CLOSED	LEAD ORG
MB7262	GL: Steam Generator Tube/Tubesheet Inspection Issues (Lund-DE/Petrone-DIPM)	C	20	09/13/2004	DE
MB4864	GL: Potential Clogging of Containment Recircul. Sump Screens by Debris Accumulation at PWRs (Cullison-DSSA/Petrone-DIPM)	C	10	09/27/2004	DSSA
MC1960	IN: Dual Unit Trip from Grid Transient (Hodge-DIPM/Perry-R1)	C	5	07/30/2004	DIPM
MC3845	IN: Loose Part Detection and Computerized Eddy Current Data Analysis in Steam Generators (Klein-DE/Hodge-DIPM)	C	3	09/09/2004	DE
MC3242	IN: Spent Fuel Rod Accountability (Karcagi-DLPM/Hodge-DIPM)	C	2	07/14/2004	DLPM
MC2633	IN: Tube Leakage Due to Fabrication Flaw in Replacement Steam Generators at Palo Verde 2 (Karwoski-DE/Rini-DIPM)	C	4	08/17/2004	DE
MC3698	RIS: Clarification On Use of Later Editions and Addenda to the ASME O.M. Code & Sec XI (Hernandez-DE/Petrone-DIPM)	C	1	08/11/2004	DE
MC3243	RIS: Consideration of Sheltering in Licensee's Range of Protective Action Recommendations (Kahler-EPPO/Petrone-DIPM)	C	3	08/16/2004	EPPO
MC2634	RIS: Proprietary Information Submissions (Benney-DLPM/King-DIPM)	C	6	07/14/2004	DLPM
MC3249	RIS: RIS 2003-18, Sup 1: Use of NEI 99-01, Methodology for Development of EALs, Rev 4 (Casto-EPPO/Petrone-DIPM)	C	2	07/28/2004	EPPO