NRR TASK ACTION PLANS

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OFFSITE POWER ISSUES

TAC No. MC3380

Last Update: 09/30/05
Lead Division: DE
Supporting Divisions: DLPM, DSSA, and DIPM
Supporting Office: RES

GROUP ONE ISSUES 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)
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	07/30/04 (C)

Milestones	Responsibility	Estimated Completion Date
10. Commission meeting on grid reliability issues.	NRR/DLPM NRR/DE	05/10/04 (C) 04/26/05 (C)
11. Establish interfaces with grid reliability organizations.	NRR/DE	12/06
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). b. Re-evaluate SBO risk (CDF) with updated SPAR models for spectrum of plants (draft report for internal/external review). c. Review SBO considerations and determine if regulatory actions are needed.	 RES/DRAA RES/DRAA NRR/DE/EEIB	 11/16/04 (C) 02/28/05 (C) 08/15/04 (C)
14. Incorporate unresolved concerns into Group Three concerns	NRR/DE/EEIB	08/15/04 (C)

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

Milestones	Responsibility	Estimated Completion Date
1. Receive NERC audit reports.	NRR/DE	06/30/04 (C)
2. Assess the information provided in the audit reports to ascertain whether NRC issues have been addressed.	NRR/DE	07/09/04 (C)
3. Incorporated results into paper (See Activity 12, page 6) to Commission.	NRR/DE	07/14/04 (C)
4. Inform the Commission of the status of the Action Plan prior to the summer peak season.	NRR/DE	08/06/04 (C)
5. Develop additional requests for information to address any short falls in the report (send to NERC).	NRR/DE	08/06/04 (C)
6 Meet with NERC to discuss their response.	NRR/DE	08/20/04 (C)
7. Re-assess any additional NERC input.	NRR/DE	09/03/04 (C)
8. Develop Group Two disposition document if different from item 5 - Group Two disposition was the same.	NRR/DE	N/A

Milestones	Responsibility	Estimated Completion Date
9. Incorporate unresolved concerns into Group Three concerns.	NRR/DE	09/17/04 (C)

GROUP THREE ISSUES 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 issues.	NRR/DE	12/22/04 (C)
2. Organize issues by topic as described in Activity 9 for the Group One concerns.	NRR/DE	08/30/04 (C)
3. Determine staff to be included in review groups.	NRR/DE	08/30/04 (C)
4. Determine NRR leads for review groups	NRR/DE	08/30/04 (C)
5. Incorporate Group Two issues not resolved in Group One or Two assessments into Group Three issues .	NRR/DE	09/17/04 (C)
6. Develop schedule for review groups to review issues .	NRR/DE, RES, (Stakeholders input)	02/03/05 (C)
7. Review groups obtain information necessary to address issues .	NRR/DE, RES, (Stakeholders input)	01/31/05 (C)
8. Review groups assess issues including management briefings on LOOP risk analyses and EDG reliability and issue draft LOOP risk report for review.	NRR/DE, RES/DRAA, (Stakeholders input)	02/28/05 (C)
9. Review group members develop regulatory position to present to Commission - Determined a regulatory position in not necessary at this time.	NRR/DE, RES	04/12/05 (C)
10. Commission briefing	NRR/DE	04/26/05 (C)
11. Final status of action plan on grid issues to Commission.	NRR/DE	07/13/05 (C)
12. Resolve public comments and issue as final - Generic Letter (GL) 2005-XX; "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power."	NRR/DE/EEIB	12/15/05 (T)
13. Evaluate the results of the final GL to determine which regulatory requirements need to be revised to ensure safe NPP operation, including rulemaking if appropriate.	NRR/DE/EEIB	12/06 (T)
14. Collection and on-going assessment of impact of grid operating experience and data by NERC and the NRC.	NRR/DE/EEIB RES/DSARE	12/06 (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).

In December 2003, the NRC Chairman directed the NRC Executive Director of Operations to conduct a review of the issues raised in a report entitled "State of U.S. Power Grid from a Nuclear Power Plant Perspective." Following deterministic and risk evaluations, it was concluded for that for the following reasons, that there was certain urgency to address, before the Summer of 2004, plant operational readiness for the possibility that an event similar to the August 14, 2003, event occurs: (1) Long duration Loss of Offsite Power events are safety significant, (2) Risk increases when the plant's ability to cope with event is decreased due to online equipment outages, and (3) Grid is less reliable during the Summer period .

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.

<u>Proposed Actions</u>: 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.

<u>Originating Document</u>: 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."

<u>Regulatory Assessment</u>: 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: The NRC staff established a Memorandum of Agreement (MOA) between the NRC and NERC and a MOA between the NRC and FERC. NERC and FERC signed the MOAs on August 27, and September 1, 2004, respectively (ADAM Accession Nos. ML042520329 and ML042580167). In the MOAs, NERC , FERC, and NRC have agreed to consult with each other with regard to the availability of technical information that would be useful in the areas of mutual interest, and to promote and encourage a free flow of such information pertaining to electrical grid reliability, security, and integrity. The staff met with FERC on October 26, 2004, as part of the MOA (ADAMS Accession No. ML043090122). The staff and FERC has also communicated via e-mail and phone to exchange grid-related information impacting nuclear power plants. The staff met with NERC on November 16 and 17, 2004 (ADAMS Accession No. ML043270359), to work on the four Appendices to the MOA: Appendix I - Coordination plan for communications and information sharing during emergencies, Appendix II - Coordination plan for event analysis and follow-up review activities, Appendix III - Coordination plan for the exchange of operational experience data and information, Appendix IV - Coordination plan for participation by NRC staff in NERC committee and subgroup activities. The Appendices (ADAM Accession No. ML051750337) were signed by both the NRC and NERC on June 2, 2005.

[Deleted for Archive Purposes see previous reports]

In January 2005, the staff began processing a generic communication in order to meet SRM M041209. The generic communication is contained in ADAMS Accession No. ML050310139 and the Committee To Review Generic Requirements (CRGR) memorandum is contained in ADAMS accession No. ML050390189. On April 12, 2005, in accordance with SRM M041209, the staff issued the draft generic letter for public comment in the *Federal Register* (70 FR 19125) to request that licensees submit information to the staff concerning the status of their compliance with certain NRC regulations (ADAMS Accession No. ML050810504). . The public comment period closed on June 13, 2005. A total of 14 comments were received. The staff is currently reviewing the comments to meet the December 15, 2005, issuance of the final GL.

On February 28, 2005, the staff issued in the *Federal Register* (70 FR 9682) a notice of availability of the RES draft report entitled, “Station Blackout Risk Evaluation for Nuclear Power Plants,” for comment. The comment period expires on April 15, 2005. This report is an update of several previous reports analyzing the risk from loss of offsite power and subsequent station blackout events at U.S. commercial nuclear power plants. The risk measure used is core damage frequency (CDF). Standardized plant analysis risk (SPAR) models developed by the U.S. Nuclear Regulatory Commission, covering the 103 operating commercial nuclear power plants, were used to evaluate the risk. CDF results indicating contributions from station blackout (SBO) scenarios and other LOOP scenarios are presented for each of the 103 plants, along with plant class and industry averages. In addition, a comprehensive review of emergency diesel generator (EDG) performance was performed to obtain current estimates for input to the SPAR models. Overall results indicate that CDFs for LOOP and SBO are lower than previous estimates. Contributing to this risk reduction is an improvement in EDG performance.

During the 520th meeting held on March 3-5, 2005, of the ACRS, the ACRS plans to review the generic communication after the public comment period (ADAMS Accession No. ML050680282).

The staff sponsored a grid reliability breakout session at the Regulatory Information Conference on March 8, 2005. The summary and presentation slides are available on the NRC web page.

NEI made a drop-in visit on April 15, 2005, with Commissioner McGaffigan and Commissioner Lyons regarding grid reliability. The briefing package is located in ADAMS No. ML050830250.

The Commission requested a meeting regarding grid reliability. The Commission meeting was held on April 26, 2005. Documents including the meeting slides and transcripts can be found on the NRC public website.

The staff issued Temporary Instruction (TI) 2515/163, “Operational Readiness of Offsite Power,” on May 2, 2005. The objective of the TI is confirm, through inspections and interviews, the operational readiness of offsite power systems in accordance with NRC requirements. This TI will also ensure the readiness of nuclear power plants to cope with the potential challenges by power outage events during the summer of 2005. All responses were received from the resident inspectors by June 1, 2005. Staff is currently reviewing the results from the TI.

The staff has been following grid reliability activities of the following organizations: North American Electric Reliability Council (NERC), Federal Energy Regulatory Commission (FERC), Department of Homeland Security (DHS), Nuclear Energy Institute (NEI), and Institute of Nuclear Power Operations (INPO).

On September 16, 2005, the staff held a public meeting to discuss the results of Temporary Instruction (TI) 2515/163, "Operational Readiness of Offsite Power." The purpose of the meeting was to share the results of TI 2515/163 with interested stakeholders. The staff ’s presentation concluded that the TI results indicated considerable inconsistency and variability among nuclear power plants (NPP) and that high quality information from licensees to the Grid Reliability Generic Letter (to be issued in December 2005) should address staff concerns regarding the operational readiness of NPP for the summer of 2006, as well as, other long-term issues.

On September 30, 2005, the staff sent a letter to NEI, at their request, that included the plant specific results of TI 2515/163.

<u>NERC</u>
NERC has established a program to audit the readiness of all reliability coordinators and control areas in North America to perform their assigned reliability responsibilities. These audits will give immediate attention to deficiencies in control area and reliability coordinator capabilities identified by the August 14, 2003, blackout investigation. The purpose of the NERC readiness audit is to provide an independent review of control area and reliability coordinator operations to identify areas for improvement and help them achieve excellence from a reliability operations standpoint. The NERC readiness audits initiate a process to ensure that operators of the bulk electric system have the tools, processes, and procedures in place for reliable operation. The NERC readiness audits will help control areas and reliability coordinators recognize and assess their reliability responsibilities and evaluate how their operation supports those responsibilities. NERC intends to use the results of these audits to help champion the changes required to better meet the reliability responsibilities of these entities. The NERC readiness audits will be conducted on a three-year cycle; 64 readiness audits were completed by the end of 2004. Approximately, 60 audits are planned for 2005. The NRC staff has reviewed 20 NERC readiness audits and believes that effective actions are being taken to enhance the availability of offsite power for safe nuclear power plant operation.

In order to prevent and mitigate the impacts of future cascading blackout, NERC requires each regional reliability council to report to NERC quarterly on all violations of NERC and regional reliability council standards.

NERC revised its reliability standards and they were approved by its Board of Trustees on February 8, 2005. The new reliability standards took effect on April 1, 2005.

NERC is laying the groundwork for a system operator training program. The NERC training standards currently is posted on the NERC webpage for comment and sets forth educational requirements for all system operators. The training standards will drive the training programs that NERC will ultimately accredit and require system operators to successfully complete. The individual training programs will then be accredited based on how closely they follow the standards.

NERC approved the extension of the Urgent Action Cyber Security Standard. A permanent Cyber Security Standard is under development by NERC.

In addition, NERC issued its 2004 Long-Term Reliability Assessment in September 2004. According to the NERC report, resource adequacy is expected to be satisfactory throughout North America, provided new generating facilities are constructed as anticipated and NERC reliability rules are followed.

NERC sponsored workshops in April 2005 regarding the implementation of the new reliability standards for electric reliability coordinators, balancing authorities, transmission operators, planning authorities, transmission planners, and compliance mangers.

<u>FERC</u>

On April 19, 2004, the FERC Commission issued a policy statement that, among other things, explained that the FERC Commission interpreted the term "Good Utility Practice" as that term used in the *pro forma* open access transmission tariff (OATT) to include compliance with reliability standards developed by NERC *(Policy Statement on Matters Related to Bulk Power System Reliability*, 107 FERC 61,052 *(Policy Statement), clarified* 108 FERC 61,288 (2004)).

On September 7, 2004, FERC sent "Utility Vegetation Management and Bulk Electric Reliability Report from the Federal Energy Regulatory Commission" to Congress.

FERC issued an Order requiring a response to a survey on operator training practices by control area operators and transmission providers on December 27, 2004.

On February 4, 2005, FERC issued a report titled, "Principles for Efficient and Reliable Reactive Power Supply and Consumption." FERC held a technical conference at its headquarters to discuss this report on March 8, 2005. The goal of the technical conference was to discuss the proper regulatory policy toward reactive power supply and consumption. Both technical and economic issues were discussed. Industry experts presented their views on reactive power supply in three panels. Panel 1 was reliability and technical issues; Panel 2 was the short-term reactive power issues; and Panel 3 was the prospective reactive power solutions. The FERC Commissioners will take the industry expert views into account as it considers the proper regulatory policy toward reactive power supply and consumption. The NRC staff is interested in this issue, because the reactive power is important to the reliability of offsite power to the Nation's nuclear power plants. With a weakened electrical transmission system, loss of the nuclear power plant generating unit could potentially result in a drop in voltage of the offsite power system to a value that would make it inadequate to supply the onsite safety-related loads. The NRC staff will follow this issue as it progresses.

The NERC Board of Trustees approved Version 0 Reliability Standards, to become effective April 1, 2005, which have the goal of restating exiting standards in a manner that is unambiguous and measurable on February 8, 2005. In an order dated February 9, 2005, the FERC Commission supplemented its reliability policy by making clear that the term Good Utility Practice as used in the OATT includes compliance with NERC's Version 0 Reliability Standards.

DHS

The staff met with the DHS and FERC on March 1, 2005. The purpose of the meeting was to discuss with DHS and FERC areas of mutual interest regarding grid reliability. We discussed the electricity and nuclear critical infrastructure, graphic interface modeling of the health of the grid, an Acknowledgment and Agreement form for the NRC/FERC MOA, the benefits of the three agencies (DHS, FERC, and NRC) working together, and cyber security. The meeting summary is contained in ADAMS Accession No. ML050690243.

NEI

On October 20, 2004, the NEI Grid Reliability Task Force submitted to NERC a Standard Authorization Request (SAR) for a proposed reliability standard regarding nuclear offsite supply reliability. The first posting of the SAR has been completed and all comments were received by January 7, 2005. A drafting team is being formed by NERC to draft the proposed standard.

The staff participated in the Grid Reliability workshop in Atlanta on February 16-17, 2005, sponsored by NEI, INPO, EPRI, and NERC. The main theme of the meeting was the importance of communication between the nuclear power plant operator and the grid operator. The NRC Deputy Executive Director for Reactor Programs made a presentation regarding the NRC perspective regarding grid reliability. The purpose of the workshop was to make the industry aware of grid reliability issues and share information regarding emerging good practices. The audience consisted of nuclear plant operators, transmission system operators, independent system operators, contractors, consultants, NEI, INPO, Electric Power Research Institute (EPRI), NERC, and NRC. The workshop consisted of presentations and breakout sessions. The presentations were informational and the breakout sessions were discussion-based. The breakout sessions allowed participants to ask questions and to learn best practices from their peers. In addition, the workshop focused on new NERC reliability standards, improvements in operational planning, and communication issues as well as raising awareness to nuclear power plant offsite power requirements. INPO presented information concerning SOER 99-1 analysis of grid related operating experience, and switchyard/substation improvements. The meeting summary is contained in ADAMS Accession No. ML050610414.

INPO

On December 9, 2004, INPO issued World Association of Nuclear Operators (WANO) Significant Operating Experience Report (SOER) 99-1, "Loss of Grid - Addendum." The Addendum to WANO SOER 99-1 included the lessons learned from loss-of-grid and other events experienced by WANO members since 1999. The lessons learned were (1) more formal agreements between the nuclear plants and the grid operators are needed to improve grid reliability and ensure timely restoration of off-site power, (2) procedure guidance is sometimes insufficient to verify that the appropriate grid voltage is being maintained, and (3) grid and switchyard electrical equipment that can affect plant operation needs to be effectively maintained to ensure off-site power is reliable.

The staff met with INPO in Atlanta on February 15, 2005. The purpose of the meeting was to discuss with INPO areas of mutual interest in accordance with the Memorandum of Agreement (MOA) signed December 24, 1996 (ADAMS Accession No. ML011770437). The meeting discussed INPO activities, NRC activities, SOER 99-1, and a process to exchange information under the MOA. INPO also provided a tour of its facility. The meeting summary is contained in ADAMS Accession No. ML050590120.

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SIGNIFICANCE DETERMINATION PROCESS (SDP) IMPROVEMENT

TAC Nos. MA9164, MB0046, & MB2203

Last Update: 09/30/05
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 1.2	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]**	12/31/05 (T)	IIPB	The Active Issues Matrix communicates a running summary of active SDP findings focusing senior HQ and regional managers' attention on timeliness issues. This previously closed issue is being re-opened pending measurable success in meeting the timeliness requirements. See note under the section titled "Potential Problems" at the back of this document. Proposed change to IMC 0609 incorporates requirements for the early stage of the assessment period to convene a pre-SERP to establish scope and schedule for risk-informing complex and potentially safety significant findings.

Task		Completion Date	Lead	Status
Objective 1.8	Change IMC 0307 “ROP Self-Assessment Program” SDP timeliness metrics to better reflect the time spent on findings presented to the SERP.	04/30/06 (T)	IIPB	Findings presented to the SERP and tracked for timely issuance of the final risk-informed determination represents one half of one percent of all findings issued by the ROP. To better represent the time spent on this small but important group the existing metric will be augmented by establishing an average age limit for findings in this category.
2. Improve SDP Process				
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 (Multi-year effort)	SPSB	NRR has completed feasibility studies of developing external events SDP notebook worksheets through a pilot program for Diablo Canyon, Limerick, and Salem. NRR has planned to complete EE worksheets for 8 to 9 plants in fiscal year 2006. The purpose of this notebook effort is to develop worksheets to evaluate proposed SPAR EE models for these facilities as part of proposed internal benchmarking/peer review to assess the efficacy of the SPAR EE models. Following this effort, no further SDP EE notebook work is scheduled since SPAR EE model development should be

Task	Completion Date	Lead	Status
			sufficiently mature to continue, and development and maintenance of 71 EE notebooks would not be practical or efficient use of staff resources. Schedule TBD.
a. 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]**	12/31/05 (T)	SPSB	Each of the Revision 2 notebooks will include the pre-solved tables with the value of each sequence identified. The Revision 2 of the notebooks and the associated pre-solved tables will be issued as scheduled.
b. Evaluate training needs and issue revised guidance for the use of the pre-solved risk tables.	12/31/05 (T)	SPSB/ IIPB	The usage rules associated wit the risk-informed notebooks have not changed therefore, no additional training for the use of the notebooks is needed beyond the training that is routinely provided by the SRAs to the inspectors. The pre-solved tables are spreadsheets and their use and relationship to the notebooks is self-explanatory. However, a website based tutorial is being considered which will provide generic examples using a typical spreadsheet.
c. 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)]	TBD		Based on the late 2005 issue date for Revision 2 of the notebooks no routine maintenance to update the documents should be necessary. Adjustments recommended by the users should be made using the existing feedback process.

Task		Completion Date	Lead	Status
Objective 3.3	Develop or improve existing SDP tools as applicable in the following areas: **[OIG-3]**			
f.	Spent Fuel Storage	06/30/06 (T)	IIPB	This procedure is in development.
Objective 3.4	Improve the physical protection SDP, if necessary, accounting for any safeguards policy changes.	07/05 (C) 06/30/06 (T)	NSIR Support: IIPB	PP SDP was issued 7/05. The staff plans on evaluating findings using both the PP SDP and the NEI recommended SDP for the next 12 months as part of a pilot program to verify comparability.
Objective 3.6	Consider development of analysis criteria and standards for conducting detailed phase 3 analysis.(3) **[SDP 3.5.3(2)] [OIG-4]**			
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)]**	04/30/05 (C) 04/30/06 (T)	RES Support: SPSB/ IIPB	The RASP 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. The Risk Assessment of Operating Events Handbook, Rev. 0, (April 2005), provides state-of-the-art methods, best practices, examples, tips and precautions for applying SPAR models for SDP Phase 3, ASP. MD 8.3, Rev. 1 of the handbook, tentatively scheduled for issuance in the spring of 2006, will expand upon the topics in the current Rev. 0 to address uncertainty analysis, both modeling and

Task		Completion Date	Lead	Status
				parameter uncertainty. This is reflected in the new schedule.
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]**	TBD	SPSB	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. The RASP action plan calls for the development of a checklist, based on RG 1200, allowing inspectors and SRAs to verify the quality of the basis for the licensee's risk analyses. The same Handbook, Rev. 1, will establish a formal method to consider when the SPAR model should consider plant-specific inputs from the licensee. These include, for example, the modification of equipment failure probabilities, updating of initiating event frequencies, and model validation, using Bayesian formulation.
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	

Task		Completion Date	Lead	Status
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. Ten models have been completed, with an additional 5 models expected to be completed by 5/31/06. Quality assurance review of the models is contingent on the availability of licensee PRA staffs. This effort has been slowed by continuing potential Organizational Conflict of Interest issues at Idaho National Laboratory.
d.	Develop LERF model	12/31/06 (T)		Draft event trees have been developed. As part of the SPAR model enhancement program, RES is developing large early release frequency (LERF) PRA models. As of this date, 1 model has been completed (PWR with large dry containment), with an additional 6 models expected to be completed by 6/30/06.
4.	**Improve Staff Training in The Use of SDP Tools**			
5.	**Improve Clarity of Risk-Informed ROP Decision Guidance**			

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.	12/31/05 (T)	IIPB	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 entire issue of SDP timeliness is under review. The change in date reflects the ongoing effort which is detailed in a SECY paper, ROP Self Assessment for 2004. Also see section titled “Potential Problems” at the end of this document. IMC 0609, Significance Determination Process, is being updated to incorporate new guidance which will allow a pre-SERP panel to extend the SDP completion requirement beyond 90 days based on the complexity and potential high safe significance of the finding, and decide on the scope of the evaluation for reactor safety issues. SDP metrics will also augmented to provide a more realistic status of the SDP timeliness and quality. This will give credit to the regions for early completion of the process.

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)]**	TBD	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. Internal stakeholder input is being solicited in this area to help develop appropriate guidance. The date for finalizing the process will be determined after receiving input from the regional offices.

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. While this independent verification was completed by RES for FY 2005 and the infrastructure to continue is in place, the need for this independent evaluation is under review.
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)]**	TBD	IIPB Support SPSB /RES	This issue is under Review by the IMC 1245 Working Group. The new version of the P-111course is will be introduced in FY 2007.

(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. In a memorandum dated February 23, 2005, the OIG accepted the staff January 26, 2005, responses to the remaining recommendation and declared all recommendations closed.

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

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

Last Update: 09/30/05
Lead Division: DLPM
Supporting Divisions: DE, DIPM, DSSA
Supporting Office: RES

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
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	TBD (T) Note 12	DE L. Lund	
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

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.1 (continued)	b) Perform T-H assessment of flow-induced vibrations during MSLB. Using the T-H conditions calculated during the transient, generate a conservative estimate of flow-induced vibration displacement and frequency assuming steady state behavior.	12/31/02 (C) ML023650132	RES W. Krotiuk	DSSA W. Jensen
	c) Perform additional sensitivity studies as needed.	06/30/03 (C)	RES W. Krotiuk	DSSA W. Jensen
	d) Obtain information from existing analyses related to loads and displacements (axial, bending, cyclic) experienced by SG structures under MSLB conditions.	12/31/02 (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).	TBD (T) Note 14	DSSA S. Long	DE E. Murphy RES J. Muscara H. Woods
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 (C)	RES R. Lee	DSSA S. Long
	a) RES will continue to follow results of ARTIST program, especially the results of integrated testing of particulate removal during passage through the full secondary flow path. From the integral results, determine whether the decontamination factor should be revised for radionuclides transport in the secondary side of the SGs during a SGTR. Based on this, provide a letter report to closeout SGAP item 3.3.1. (Note: Schedule for release of data is decided by a foreign entity - not controlled by NRC)	12/31/07	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.	05/31/06 (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.	01/25/05 (C) 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 C. Hammer 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 C. Hammer DSSA S. Long
	h.3) Determine the feasibility of extending the Rhodes RCP seals leakage/failure model to severe accident conditions.	03/30/06 (T) Note 20	RES J. Page	DE E. Murphy C. Hammer DSSA S. Long
	h.4) Conduct large scale tests if needed.	11/30/06 (T) Note 21	RES J. Page	DE E. Murphy C. Hammer 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 19	DSSA S. Long	DE E. Murphy RES J. Muscara C. Boyd H. Woods J. Page

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 See Note 16	RES M. Junge	DSSA S. Long
	e) Extend the 3.5(b) methods and (c) model logic to include CE plants, and document them.	TBD (T) See Note 16	RES M. Junge	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 M. Junge	DSSA S. Long

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.5 (continued)	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 M. Junge	DSSA S. Long
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:		DSSA M. Hart	
	a) Assess Adams and Atwood and Adams and Sattison spiking data with respect to the ACRS comments.	08/09/01 (C)		
	b) Based upon the assessment performed in task 3.9a, develop a response to the ACRS comments.	TBD (T) Note 11		
	c) Publish in the *Federal Register* for public comment, the response to ACRS' comments.	TBD (T) Note 11		
	d) Complete review of public comments.	TBD (T) Note 11		
	e) Based upon task 3.9d, determine if additional work needs to be performed.	TBD (T) Note 11		

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.10 (MB7216)	To address concerns in the ACRS report regarding our current level of understanding of stress corrosion cracking, the limitations of current laboratory data, the difficulties with using the current laboratory data for predicting field experience (crack initiation, crack growth rates), and the notion that crack growth should not be linear with time while voltage growth is, the following tasks will be performed: (Pgs. 20-29) (See last sentence in Note 3) Specific tasks include:			
	a) Conduct tests to evaluate crack initiation, evolution, and growth. Tests to be conducted under prototypic field conditions with respect to stresses, temperatures and environments. Some tests will be conducted using tubular specimens.	12/31/05 (T)	RES J. Muscara	DE E. Murphy
	b) Using the extensive experience on stress corrosion cracking in operating SGs, and results from laboratory testing under prototypic conditions, develop models for predicting the cracking behavior of SG tubing in the operating environment.	12/31/06 (T)	RES J. Muscara	DE E. Murphy
	c) Based on the knowledge accumulated on stress corrosion cracking behavior and the properties of eddy current testing, attempt to explain the observed relationship between changes in eddy current signal voltage response and crack growth.	12/31/05 (T)	RES J. Muscara	DE E. Murphy

Item No. (TAC No.)	Milestone	Date (T=Target) (C=Complete)	Lead	Support
3.11	In order to resolve GSI 163, it is necessary to complete the work associated with tasks 3.1 through 3.5 and 3.7 through 3.9. Upon completion of those tasks, develop detailed milestones associated with preparing a GSI resolution document and obtaining the necessary approvals for closing the GSI, including ACRS acceptance of the resolution. (See Note 9)	TBD Note 9	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)	TBD Note 9	DSSA S. Long	DE E. Murphy

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).
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 is affected by the uncertainty in completion of previous 3.x milestones, and will be determined when those dates are established.

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 ongoing. A schedule will be developed once screening is completed.

12. A draft version of the generic letter (GL 2004-xx, Steam Generator Tube Integrity and Associated Technical Specifications) was issued for a 60 day comment period in the *Federal Register* (FR) in October 2004, and public comments have been received. This GL will request licensees (1) to discuss the adequacy of their steam generator tube integrity program and their plans for modifying their TS to ensure they are representative of their program and (2) to discuss how bending loads are assessed in their evaluations of tube integrity. The licensees that have adopted the new version of the TS (published in the *Federal Register* as NRC approved Technical Specification Task Force (TSTF) traveler TSTF-449 on 5/6/05) will not be required to respond to the GL. A CRGR meeting was held on 9/27/05 on the draft GL on steam generator technical specifications. The staff is incorporating CRGR comments into the draft GL. The staff will be preparing a Commission Information Paper notifying the Commission of its plan of issuing the GL.

13. Although milestone 3.4.f has been completed as planned in the RES Operating Plan, the core melt progression will be revisited under 3.4.d during the full evaluation of uncertainty. After an initial peer review of this work carried out for the NRC, some additional effort was deemed necessary. This task has moved back to 5/06 to accommodate the additional efforts.

14. Task completion is delayed due to assignment of staff to higher priority work on PTS.

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. The NRR and RES staff are currently reviewing the results of the work completed under 3.5(d). Discussions are underway to decide future actions needed to complete items 3.5(d) thru (g).

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. The results of the PRA work described in 3.5.d and approval by RES management of the details associated with completing this work scope, will help identify the level of effort needed to complete this work and the associated schedules.

19. The task is dependent on completion of preceding 3.4 subtasks and all 3.5 subtasks. New

completion dates need to be developed based on scheduled completion of 3.4 and 3.5 milestones.

20. Based on preliminary evaluation of RCP seal behavior, it was determined to further investigate the potential for excessive seal leakage or failure.

21. At this time, the only potential large-scale tests will probably be associated with RCP seal leakage/failure. The estimated date is tentative until more is known about the need for full-scale testing.

Description: 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).
- 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.
- 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.
- 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)

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

Last Update: 09/30/05
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	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	06/05 (C) 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 (C) NUREG-1823 ML050690012	RES/DET	DE

Milestone		Date (T=Target) (C=Complete)	Lead	Support
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 (C) ML043010675	DE	RES/DET
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 (C) ML043000274	RES/DET	DE
2.	a. Evaluate individual plant responses to Bulletin 2002-01 regarding Boric Acid Inspection Programs (60-day responses and necessary follow-up) b. Issue public document to summarize evaluation of plant responses.	06/03 (C) ML031760568 07/03 (C) ML032100653	DE DE	DLPM DLPM DRIP
3.	Participate in meetings and establish communications with appropriate stakeholders (e.g.,MRP, ASME).	Ongoing	DE	RES/DET DLPM DRIP DSSA industry
4.	Evaluate need to take additional regulatory actions and determine appropriate regulatory tool(s).	06/03 (C) 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

Milestone		Date (T=Target) (C=Complete)	Lead	Support
8.	Review and evaluate the adequacy of revised ASME Code Requirements for Pressure Testing/Leakage Evaluation being developed by the ASME Code, Section XI, Task Group on Boric Acid .	12/05 (T) Note (1)	DE	RES/DET
Part III - Inspection Programs				
1.	Develop inspection guidance or revise existing guidance to ensure that VHP nozzles and the RPV head area are periodically reviewed by the NRC during licensee ISI activities. [LLTF 3.3.4(3)-High]	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] b. Perform follow-up evaluation of inspection guidance and licensee program acceptability after conducting inspections for approximately one year.	06/04 (C) 05/05 (C) ML051360392	DIPM DIPM	DE RES/DET Regions 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.

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

Historical Background: In March 2002, while conducting inspections in response to Bulletin 2001-01, the Davis-Besse Nuclear Power Station identified three CRDM nozzles with indications of axial cracking, which were through-wall, and resulted in reactor coolant pressure boundary leakage. During the nozzle repair activities, 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).

<u>Proposed Actions</u>: 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.

<u>Originating Documents</u>:

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

<u>Regulatory Assessment</u>: 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 report is publicly available in ADAMS (ML032461221).

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 report is publicly available in ADAMS (ML040910354).

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). The staff participated in ASME Code Committee development of revised inspection requirements. In June 2005, the ASME Board on Nuclear Codes and Standards approved Code Case –729, which provides additional inspection requirements for RPV upper heads. Therefore, Part I, item 5 is considered complete. Once Code Case –729 is formally published (expected in December 2005), the staff will evaluate endorsing it in a revision to 10 CFR 50.55a.

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 (ML032100653). The evaluation of responses to Bulletin 2002-01, which included audits of BACC programs at five plants, determined that the plants complied with requirements at the programmatic level. In general, the results indicated weaknesses in the licensees' BACC and ASME Section XI programs. The weaknesses identified in the RIS included identifying pressure boundary leakage and potential leakage paths, looking for boric acid crystals, walking down systems when the plant is entering or leaving the hot shutdown mode, and detecting small leaks during normal power operation. Based on this review and the discovery of leakage on vessel bottom penetrations at South Texas Project, Bulletin 2003-02 was issued.

The staff collected information on available worldwide operating experience on boric acid corrosion of pressure boundary materials. The staff also contracted Argonne National Lab to conduct a test program on boric acid corrosion of light-water reactor pressure vessel materials. The results were published in NUREG/CR-6875. This information and the information previously collected on nozzle cracking along with the staff evaluation of the SCC models have been incorporated into NUREG-1823, "U.S. Plant Experience with Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials" (ML050690012).

The staff used the information collected on boric acid corrosion and the information previously collected regarding Alloy 600, Alloy 690 and other nickel-based alloy nozzle cracking to develop a course of action and an implementation schedule to address LLTF 3.1.1(1). The staff met with industry representatives on March 24, 2005, to discuss their activities for addressing PWSCC in nickel based alloy butt welds and in other locations in the reactor coolant system. The presentations were high level and lacked the technical details and schedular commitments the staff were expecting. The staff also held a meeting with industry representatives on September 29, 2005, during which representatives of the Materials Reliability Program indicated that their inspection guidelines for this issue would not be available until the end of 2006. Based on the results of these meetings and interactions with NRR senior management, the staff is evaluating Code Case N-722 for incorporation into 10 CFR 50.55a. It contains inspection rules for boric acid corrosion and cracking of nickel-based alloy nozzles and addresses the course of actions associated with the closure of LLTF 3.1.1(1).

In Part III activities, inspection procedure revisions addressing RPV head inspection and boric acid corrosion control programs were issued. Temporary Instruction (TI) 2515/150, issued on October 18, 2002, provides guidance for assessing the licensees' RPV head inspections pursuant to Order

EA-03-009. The TI also includes instructions for follow-up on findings of boric acid accumulation. Inspection Procedure (IP) 71111.08, “Inservice Inspection Activities,” dated May 14, 2004, provides periodic inspection requirements and guidance for boric acid corrosion control. The Regions provided feedback regarding the implementation of TI 2515/150 and IP 71111.08 since October 2002. In addition, the Inspection Program Branch (IIPB) reviewed inspection results from TI 2515/150 and IP 71111.08. As a result of the licensees’ visual and non-visual inspections and NRC direct observations and oversight of licensees’ activities, a number of facilities have made repairs to their vessel heads and some have replaced the vessel heads. In some cases, repairs were required; in others the licensee took actions voluntarily. Feedback from each Region and IIPB staff review indicates that the licensees’ programs are generally adequate for locating and evaluating and/or correcting boric acid leaks. Although several inspection findings were identified, none were of greater than very low significance. The staff will continue to evaluate the effectiveness of this IP as part of annual ROP self-assessment and make appropriate improvements as needed.

Schedule Changes Since Last Update: None.

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PWR SUMP PERFORMANCE

TAC Nos. MA6454, MA2452, MA4014, MA0704, M95473, MA6204, MA0698, MB4047, MB6411, MB3103, MB8052, MB7776, MB9470, MB4864, MB9931, MC0307, MC1154, MB9549, MC4272, MC5881, MC6467, MC6470, MB5625, MB4865, MC0725/6, MB5221, MB5964, MB6589, MB7228, MC1627, MB5334, MC2628, MB6946, MC6659, MC6661, MC6730, MC6731, MC7565, and MC7564

Last Update: 09/30/05
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” ○ Complete review of licensee responses ○ Complete revision of RG 1.1/RG 1.82, R3	 03/00 (C) 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)
3.	Coatings Condition Assessment Guidance	TBD
4.	Confirmatory Coatings Transport Testing	TBD
PART IV:	**GSI 191, “ASSESSMENT OF DEBRIS ACCUMULATION ON PRESSURIZED WATER REACTOR (PWR) SUMP PERFORMANCE”**	
1.	NRC-sponsored research program on the potential for loss of ECCS NPSH during a LOCA due to clogging by debris ○ Preliminary (qualitative) risk assessment (NRR) ○ Complete collection of plant data to support research program ○ Integrate industry activities into this Action Plan ○ Complete research program on PWR sump blockage ○ Evaluate need for regulatory action based on research program results (NRR)	 03/99 (C) 06/99 (C) 04/00 (C) 09/01 (C) 03/02 (C)

MILESTONES		DATE (T/C)
	○ Chemical effects: Determine if sump pool environment generates by-products which contribute to sump clogging	04/06 (T)
	○ Debris Transport & Head loss: Confirmatory research on debris transport of coatings and head losses associated with PWR containment materials with and without chemical effects	04/06 (T)
	○ Downstream effects: Confirmatory research on the effect of injected debris on HPSI throttle valve performance	04/06 (T)
2.	Resolve ECCS suction clogging issue for PWRs (Regulation/Guidance Development and Issuance Stages of GSI process in MD 6.4))	
	○ Brief NRR ET to obtain approval to prepare a generic letter (GL)	02/02 (C)
	○ Public meeting with NEI, WOG, B&WOG, CEOG	03/02 (C)
	○ ACRS Briefing on proposed draft GL	02/03 (C)
	○ CRGR Briefing on proposed Bulletin 2003-01	04/03 (C)
	○ Information Paper to Commission, Issue Bulletin 2003-01	06/03 (C)
	○ NEI publish PWR Industry Evaluation Guidelines (Draft)	10/03 (C)
	○ CRGR Briefing on proposed draft GL	02/04 (C)
	○ Proposed draft GL issued for Public Comment	03/04 (C)
	○ GL issuance	09/04 (C)
	○ Issue Safety Evaluation on Methodology	12/04 (C)
	○ NRC starts Reviews of GL Responses and Selective Audits	09/05 (C)
	○ GL date for licensees to start modifications, if needed, using approved guidelines	04/06 (T)
	○ 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; two of which 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. This part of the plan is being reopened to track development of guidance for coatings condition assessment and the NRC confirmatory coatings transport testing program.

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." 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." GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," dated December 3, 1985, documented the NRC's resolution of USI A-43. 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, including events and research, challenged the adequacy of the NRC's conclusion that no new requirements were needed to prevent clogging of ECCS strainers in BWRs. 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.

Events at the Perry Nuclear Power Plant demonstrated high strainer pressure drop caused by the filtering of suppression pool particulates (corrosion products or "sludge") by fibrous materials adhering to the ECCS strainer surfaces. The effect of particulate filtering on head loss had been previously unrecognized and therefore its effect had not been considered.

An event at Limerick Unit 1 demonstrated the importance of foreign material exclusion practices to ensure adequate suppression pool and containment cleanliness. In addition, the event 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. 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. 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.

Events at a number of plants 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 research aimed at providing technical information regarding the failure of coatings . 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.

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 of NRCB 96-03. RES conducted a program to evaluate PWR sump designs and their susceptibility to blockage by debris. Risk insights supported the conclusions drawn relative to the need for licensees to address the potential for ECCS suction clogging. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs. As a result of research work and plant experience, the NRC is additionally requesting that PWRs evaluate potential downstream and chemical effects as part of the resolution of GSI-191.

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 considers continued operation of PWRs during the implementation of this action plan to be acceptable because the probability of the most challenging 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 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. 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 is that in many cases, 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. 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, compartments, 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, NRC'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 whether the ECCS or containment spray pumps would be able to continue operating if 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 is considered to be appropriate.

Current Status: The staff continues to hold regular public meetings with stakeholders including PWR licensees and the NEI sump performance task force on the progress toward resolving GSI-191. On June 30, 2005, the NRC held a public meeting with the Nuclear Energy Institute (NEI) and other stakeholders to discuss chemical effects, staff expectations in regards to the responses to Generic Letter (GL) 2004-02, and also informed licensees of expectations regarding containment coatings qualifications programs.

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 completing the review of licensee's responses and issuing closeout letters. NEI submitted the second guidance document, "PWR Containment Sump Evaluation Methodology on May 28, 2004. 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 staff Safety Evaluation (SE), issued December 6, 2004, provides 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". The initial 90-day GL responses have been received from all PWR licensee's and reviewed by the staff. A pilot program to audit the evaluations requested in the generic letter was initiated. The first audit of Crystal River Unit 3 was completed in June 2005, and the second audit of Fort Calhoun Station was completed in September 2005.

The staff continues to work with the Office of Nuclear Regulatory Research and the NRR Division of Engineering involving a new User Need for coatings condition assessment and the NRC confirmatory coatings transport testing program. The staff participated in a coatings transport kickoff meeting and test facility tour of the Naval Surface Warfare Center, Carderock Division, on October 4, 2005.

NRR Lead PMs: Michael Webb, LPD 4, 415-1347 (GL 2004-02)
Alan Wang, LPD 4, 415-1445 (Bulletin 2003-01)

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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)
Paul Klein, EMCB, 415-4030 (Chemical Effects)
Matt Yoder, EMCB, 415-4017 (Coatings)
Steve Unikewicz Engineering, 415-3819 (Downstream Effects)
Ruth Reyes, SPLB, 415-3249 (Upstream)

<u>RES Technical Contacts</u>: B. P. Jain, ERAB, 415-6303
Rob Tregoning, ERAB, 415-6657

<u>References</u>:

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 04-07, PWR Sump Performance Evaluation Methodology, December 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.

GSI-191 Safety Evaluation, "Pressurized Water Reactor Containment Sump Evaluation Methodology," dated December 6, 2004.

Report on Results of Staff Pilot Audit - Crystal River Analyses Required for the Response to Generic Letter 2004-02 and GSI-191 Resolution, dated June 29, 2005.

NRC Information Notice 05-26, "Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment" issued September 16, 2005.

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: 09/30/05
Lead NRR Division: DSSA
Supporting Division: DLPM, DRIP
Supporting Office: RES

	MILESTONES	DATE (T/C)
1.	SPLB staff briefed DRIP/DSSA management to obtain management endorsement for actions to implement voluntary industry initiatives through issuance of generic letter.	11/15/04 (C)
2.	SPLB staff, with assistance from SPSB & RPRP, briefed the ET/LT on closure plans for GSI-189. A consensus was reached at the ET/LT meeting to go forward with letters (in lieu of a generic letter) to the owners to capture voluntary licensee initiatives for providing backup power sources to the hydrogen igniters.	11/29/04 (C)
3.	The NRC staff met with senior representatives of the six affected PWR and BWR licensees to discuss safety-security-related insights and briefed the industry regarding potential changes to the voluntary actions.	03/30/05 (C)
4.	A memorandum to the Commissioners from the EDO was issued to inform the Commission of the regulatory analysis results and staff's plans for resolution of GSI-189.	06/14/05 (C)
5.	Issue letters to owners of affected plants requesting information about voluntary actions to provide backup power to hydrogen igniters at affected units and how these features will be captured within the plants licensing basis.	01/31/06 (T)
6.	Transfer resolution responsibility to Generic Safety Issues Branch in NRR.	02/28/06 (T)
7.	Receive responses from affected owners regarding voluntary actions.	05/30/06 (T)

Description: Following a severe accident concurrent with station blackout (SBO), the PWR ice condenser containment and BWR Mark III containment are vulnerable to failures from hydrogen deflagrations and detonations. To resolve the generic safety issue, GSI-189, NRR recommended the addition of a backup power supply for the combustible gas igniters for the plants with Ice Condenser or Mark III containments. The generic safety 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. The affected plants are the four 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 four single-unit BWR nuclear plants with Mark III containments - Grand Gulf, River Bend, Clinton, and Perry.

Historical Background: In response to SECY-00-198, a 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). 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 associated with a station blackout, PWR ice condenser and BWR Mark III containments are vulnerable to failures from hydrogen deflagrations or detonations, because the existing hydrogen 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 Office of Nuclear Regulatory Research (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 also worked with the 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 cost-benefit analysis enveloping all the susceptible plants.

The technical assessment quantified the reduction in the conditional containment failure probability associated with combustible gas 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.

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 early containment failure probability is dominated by non-DCH hydrogen combustion events for ice condenser containments due to relatively low containment free volume and low containment structural strength in these designs. These containments rely upon 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 concentration 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 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 containments are also pressure-suppression designs, but are operated with the containment "inerted," i.e., the drywell and 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 for 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). In response to the ACRS letter, a letter from the EDO stated that 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 the use of SAMGs to be 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 was 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's recommendations were presented to the ACRS on November 6, 2003, citing the results from recent studies which identified a near certainty of containment failure without the use of igniters during the 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 Owner's Group (BWROG) prior to making a decision to pursue rulemaking. NRR staff also discussed alternatives with the BWROG for the four affected BWR plants.

Proposed Actions: To resolve GSI-189, DSSA/SPLB developed a draft of the proposed design criteria for the backup power supply, and discussed it with the industry in the public meetings on February 3, and March 31, 2004. The draft design criteria incorporated the comments received from the industry, and was issued to the division directors of NRR for comment on August 13, 2004.

The NRR staff 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. At the public meeting, Duke power, representing two PWR ice condenser sites, Catawba-1&2, McGuire-1&2, indicated a willingness to modify an existing safe-shutdown diesel generator that can manually hookup to the hydrogen igniters as needed. The American Electric Power (AEP) representative indicated a willingness to provide back up power for D.C. Cook 1&2 from new, large diesel generators which are already planned for installation to support an increased allowed outage time. The Tennessee Valley Authority (TVA), representing two PWR ice condenser containment sites, Sequoyah - 1&2, Watts Bar-1, indicated a willingness to provide new design for the backup power supply as the standard emergency power on the 69Kv board. However, the BWR licensees, BWROG representatives, stated that 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. The BWROG indicated a willingness to make hardware modifications to supply backup power from the existing high-pressure core spray (HPCS, division 3) 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.

Based on stakeholders' responses, a staff meeting was held on October 26, 2004, to discuss issues regarding (1) rulemaking versus other options to resolve GSI-189 and (2) 1-hour versus 2-hour time limits for the BWRs to connect backup power source to hydrogen igniters. At the meeting, the staff agreed to proceed with voluntary industry initiatives through the issuance of a generic letter instead of rulemaking, and leave the design criteria unchanged. The SPLB staff briefed DRIP/DSSA divisional management on GSI-189 status in a meeting dated November 15, 2004, to obtain management endorsement of actions to implement voluntary industrial initiatives by issuing a generic letter, and proposed to notify the Commission of the change from rulemaking. The SPLB staff, with assistance from SPSB and RPRP, briefed the ET/LT in a meeting dated November 29, 2004, on plans to resolve GSI-189. A consensus was reached at the meeting to go forward with letters (in lieu of a generic letter) to the owner's groups to capture voluntary licensee initiatives on providing backup power sources to hydrogen igniters. Additional milestones to implement this approach have been developed.

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:

Current Status: The DRIP/RPRP staff finalized the regulatory analysis. The final regulatory analysis indicated that the backup power modification may provide a substantial safety benefit at a justifiable cost for the PWRs with ice-condenser containment, and the proposed voluntary actions provide the majority of the benefit. The costs exceed the benefits for all BWR regulatory options, and none of the options for the BWRs provides a substantial increase in the overall protection of public health and safety. However, defense-in-depth considerations in improving the balance among accident prevention and mitigation provide an additional un-quantified benefit for both containment types. On June 14, 2005, the EDO issued a memorandum to inform the Commission of the regulatory analysis results and staff's plans to pursue voluntary actions for resolution of GSI-189.

In February and early March 2005, the NRR staff met with representatives of RES, NSIR, and OEDO to develop and understanding of the safety security interface and actions initiated in the security arena that bear on the solution of the safety issue. On March 30, 2005, and again on August 3, 2005, the NRC staff met with senior representatives of the six affected PWR and BWR licensees to discuss safety-security-related insights for consideration in developing plans for voluntary actions. The staff will continue monitoring the safety-security interface and will inform the licensees of new insights related to GSI-189.

<u>Contacts</u>:

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NRR Rulemaking Contact:	Stewart Schneider, DRIP/RPRP, 415-4123
RES Technical Contact:	Allen Notafrancesco, DSARE/SMS, 415-6499

<u>References</u>:

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3. NUREG/CR-4551, Vol. 3, Rev. 1, Part 3, "Evaluation of Severe Accident Risks: Surry Unit 1, External Events," December 1990.
4. NUREG/CR-4551, Vol. 5, Rev. 1, Part 1, "Evaluation of Severe Accident Risks: Sequoyah, Unit 1, Main Report," December 1990.
5. NUREG/CR-4551, Vol. 6, Rev. 1, Part 1, "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1, Main Report," December 1990.
6. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990.
7. Letter from V. Mubayi, Brookhaven National Laboratory, to H. VanderMolen, NRC, "NUREG-1150 Consequence Calculations," July 20, 1994.
8. T. D. Brown *et. al.,* "NUREG-1150 Data Base Assessment Program: A Description of the Computational Risk Integration and Conditional Evaluation Tool (CRIC-ET) Software and the NUREG-1150 Data Base," letter report, March 1995.
9. NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," Final Report, January 1997.
10. 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors," January 1, 2000 (last revised 1987).
11. NUREG/CR-6427, "Assessment of the DCH Issue for Plants with Ice Condenser Containments," April 2000.
12. NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," July 2000.
13. Memorandum to Samuel Collins, Director, Office of NRR, from Ashok Thadani, Director, Office of RES, September 29, 2000, regarding Research Information Letter RIL-0005, "Completion of Research to Address Direct Containment Heating Issue for All Pressurized Water Reactors." (ML003755724).

14. Memorandum to Ashok Thadani, Director, Office of RES, to Samuel Collins, Director, Office of NRR, November 22, 2000, regarding Research Information Letter RIL-0005, "Completion of Research to Address Direct Containment Heating Issue for All Pressurized Water Reactors." ML003761979).
15. NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program, Main Report," Draft Report for Public Comment, April 2001.
16. Memorandum to John Flack, Chief, Regulatory Effectiveness and Human Factors Branch, Division of Systems Analysis and Regulatory Effectiveness, RES, from Mark Cunningham, Chief, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Applications, RES, "Information Concerning Generic Issue on Combustible Gas Control for PWR Ice Condenser and BWR Mark III Containment Designs," August 15, 2001 (ML012330522).
17. Memorandum to M. Snodderly (NRC) from M. Zavisca and M. Khatib-Rahbar (ERI), "Combustible Gas Control Risk Calculations (DRAFT) for Risk-Informed Alternative to Combustible Gas Control Rule for PWR Ice Condenser, BWR Mark I, and BWR Mark III (10 CFR 50.44)," October 22, 2001.
18. Management Directive 6.4 (MD 6.4), "Generic Issues Program," December 4, 2001.
19. Management Directive 6.3 (MD 6.3), "The Rulemaking Process," July 31, 2001.
20. Memorandum from John H. Flack, Chief, REAHFB:DSARE:RES to Jack E. Rosenthal, Chief, SMSAB:DSARE:RES and Mark A. Cunningham, Chief, PRAB:DRAA:RES, dated February 6, 2002, regarding "Panel Review of GSI-189, Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
21. Memo from Farouk Eltawila, Director, RES, to Ashok C. Thadani, Director RES, dated February 13, 2002, regarding RES Task Action Plan for Resolving Generic Safety Issue 189: "Post Accident Combustible Gas Control in Pressure Suppression Containments."
22. Memorandum from William Travers, EDO, to The Commissioners, dated May 13, 2002 (SECY-02-0080), Proposed Rulemaking–Risk Informed 10CFR50.44, "Combustible Gas Control In Containment", (WITS 20010003).
23. Advisory Committee on Reactor Safeguards Meeting Minutes, 493rd Meeting, June 6, 2002, regarding Technical Assessment Generic Safety Issue (GSI)-189.
24. Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment, Rev. 2, September 24, 2002, by Information Systems Laboratories, Inc., Rockville, MD.
25. Hydrogen Control Calculations for the Sequoyah Plant, draft letter report, Rev. 3, September 30, 2002, by Sandia National Laboratories.
26. Memorandum from Ashok Thadani, RES to William Travers, EDO, dated October 1, 2002, regarding, "Revision to NRC's Regulatory Analysis Guidelines [NUREG/BR-0058] and RES Office Letter 1 to Conform to OMB's Information Quality Guidelines."
27. Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice condenser and Mark III Containment Plants, draft letter report, October 4, 2002, by Brookhaven National Laboratory. ADAMS ML022880554.
28. Advisory Committee on Reactor Safeguards Subcommittee on Thermal-Hydraulic Phenomena and Subcommittee on Reliability and Probabilistic Risk Assessment Meeting Minutes, November 5, 2002, regarding Generic Safety Issue (GSI)-189.
29. Advisory Committee on Reactor Safeguards Meeting Minutes, 497th Meeting, November 7, 2002, regarding Technical Assessment Generic Safety Issue (GSI) -189.
30. Memo from George E. Apostolakis, Chairman Advisory Committee on Reactor Safeguards, to the Commission Chairman Richard A. Meserve, dated November 13, 2002, regarding "Recommendations Proposed by the Office of NRR for Resolving Generic Safety Issue -189, Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident. ML023230513
31. Memo from Ashok C. Thadani, Director RES, to Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, dated December 17, 2002, regarding RES Proposed Recommendation for Resolving Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident." ML023510161
32. Attachment to Memo from Ashok C. Thadani, Director RES, to Samuel J. Collins, Director, Office of

Nuclear Reactor Regulation, dated December 17, 2002, "Technical Assessment Summary for GSI-189: Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."

33. Memo from John A. Zwolinski, Director, Division of Licensing Project Management, NRR to Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, RES, dated January 21, 2003, regarding, "Resolution Process for Generic Safety Issue (GSI) 189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
34. Memo from Jack Rosenthal, Branch Chief, Safety Margins and Systems Analysis Branch, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research to John Hannon, Branch Chief, Plant Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation dated June 19, 2003, regarding, Final Contractor's Reports: Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
35. Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants, Final Letter Report, Energy Sciences and Technology Department, Brookhaven National Laboratory, December 23, 2002 (ML031700011).
36. Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment, Revision 2, Information Systems Laboratories, Inc., September 24, 2002 (ML031700015).
37. Hydrogen Control Calculations for the Sequoyah Plant, Final Letter Report, March 2003, Prepared By Sandia National Laboratories, March 2003 (ML031700025).
38. Memorandum from Luis A. Reyes, EDO, to The Commissioners, dated June 14, 2005, "Status of Staff Activities to Resolve Generic Safety Issue 189, 'Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident' (WITS 20010144) (ML051440875).
39. Attachment to Memorandum from Luis A. Reyes, EDO, to The Commissioners, dated June 14, 2005, "Background - Status of Staff Activities to Resolve Generic Safety Issue 189, 'Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident' (WITS 20010144) (ML051590380).
40. Regulatory Analysis for Proposed Action to Address Generic Safety Issue 189: Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident, May 24, 2005 (ML051450060).

CONTROL ROOM HABITABILITY

TAC No. MC0021

Last Update: 09/30/05
Lead NRR Division: DSSA
Supporting Division: TBD
CTL: N/A
GSI No.: N/A

MILESTONE	DATE (T/C)
Staff review of NEI 99-03 and redline and strikeout version provided to NEI Control Room Habitability task force	04/17/01 (C)
Staff will prepare Generic Letter and develop draft Regulatory Guides on Control Room Habitability at Nuclear Power Reactors (DG-1114), Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (DG-1115), Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light Water Nuclear Power Reactors (DG-1113), and Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (DG-1111)	07/01/01 (C)
Office review of draft Regulatory Guides DG-1111 and DG-1113	12 /31/01 (C)
Office review of draft Regulatory Guides DG-1114 and DG-1115 and draft Generic Letter	03/01/02 (C)
Brief CRGR on draft Regulatory Guides DG-1111 and DG-1113	12/31/01 (C)
Brief CRGR on draft Regulatory Guides DG-1114 and DG-1115 and draft Generic Letter draft	GENERIC LETTER: 04/29/02 (C) DG-1114, DG-1115: 03/11/02 (C)
Issue draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115 and draft Generic Letter for public comment draft	GENERIC LETTER: 05/09/02 (C) DG-1111: 12/31/01 (C) DG-1113: 01/31/02 (C) DG-1114: 03/28/02 (C) DG-1115: 03/28/02 (C)
Public meeting on draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115 and draft Generic Letter	RI: 07/11/02 (C) RII: 07/16/02 (C) RIII: 08/06/02 (C) RIV: 07/18/02 (C)
Resolve public comments on draft Regulatory Guides DG-1111, DG-1113, DG-1114, and DG-1115	12/30/03*
Office review and concurrence of final Regulatory Guides and Generic Letter	DG-1111, DG-1113: 01/31/03 (C) DG1114, DG-1115, and GENERIC LETTER 2003-XX: 03/24/03 (C)

MILESTONE	DATE (T/C)
Brief ACRS on final Regulatory Guides and Generic Letter	04/10/03 (C)
Brief CRGR on final Regulatory Guides and Generic Letter	04/22/03 (C)
Commission Information Paper on Generic Letter	06/03 (C)
Issue final Regulatory Guides and Generic Letter	06/03 (C)
Review 60 days responses to Generic Letter	12/30/03 (C)
Develop replacement technical specification for Attachment B to Regulatory Guide 1.196	1/24/05 (C)*
Develop response to TSTF letter of 3/8/04	1/24/05 (C)
Transmit staff proposal for revision to TSTF-448	1/24/05 (C)
Review 180 days responses to Generic Letter	90 days after receipt**
Develop & transmit RAIs on 180 days responses to licensees	()***
Conduct survey of 3-4 plants who filed 180 days responses to Generic Letter	****
Establish Revised Technical Specification on Control Room Envelope Testing	12/31/05
Determine need for plant inspections	12/30/05
Develop Temporary Instruction	12/30/05
Conduct Plant Inspections	TBD
Assess overall response to Generic Letter	TBD

*The date of 1/31/05 is just to develop a replacement technical specification. It is not a date when the revision would be issued to Regulatory Guide 1.196.

**Beginning March 31, 2005.

***Beginning 10/1/05 4-5 prepared per month until back log is retired.

****Surveys to begin 3/15/06 and to be completed by 9/15/06.

Description: In its review of license amendment submittals over the past several years, the staff has identified numerous problems associated with the assessment of control room habitability. These problems have included the overall integrity of the control room envelope and the manner in which licensees have demonstrated the ability of their control room designs to meet GDC-19. Licensees have failed to:

1. Assess the impact of proposed changes to plant design, operation, and performance on control room habitability,

2. Identify the limiting accident,

3. Appropriately credit the performance of control room isolation and emergency ventilation systems in a manner consistent with system design and operation, and

4. Substantiate assumptions regarding control room unfiltered inleakage.

In response to Item 4 above, several utilities performed testing of their control room envelope (CRE) to determine unfiltered inleakage using methods from ASTM E741-93, "Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." As of May 2003, about

30 percent of the operating plants' control rooms had been tested. At that time, all of the control rooms except one measured unfiltered inleakage which exceeded the design basis analysis assumptions. In several cases, the measured inleakage exceeded the design basis value by over an order of magnitude. In most of the cases to date, licensees have been able to ultimately demonstrate compliance to GDC-19 through corrective action and retesting or by re-analysis. The nearly 100 percent failure rate of such a large fraction of the operating plant control rooms created a large uncertainty in the ability of the remaining untested facilities to meet control room habitability requirements. These control room habitability issues adversely affected the timely review of many license amendment requests. Licensee and staff expended significant resources to resolve differences regarding licensing and design basis issues and weaknesses in analysis assumptions, inputs and methods. While the capability of untested control rooms to meet their design basis was in question, the staff has reasonable assurance that continued operation was safe since compensatory measures; e.g., use of self-contained breathing apparatus and potassium iodide were established by licensees.

Background: General Design Criterion (GDC-19), "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, establishes criteria for a control room. It requires that a control room be provided which allows operators to take actions under normal conditions to operate the reactor safely and to maintain the reactor in a safe condition under accident conditions. GDC-19 also requires that equipment be provided at locations outside the control room with the design capability for hot shutdown of the reactor, including the necessary instrumentation and controls that both maintain the reactor in a safe condition during hot shutdown and possess the capability for the cold shutdown of the reactor through the use of suitable procedures. GDC-19 also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures more than 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Applicants to build or license a new plant under Part 50 after January 10, 1997, applicants for design certification under Part 52 after January 10, 1997, applicants to build a new plant under Part 52 who don't reference a standard design certification, or current licensees who want to use an alternative source term as allowed by 50.67, are required by GDC-19 to use as the control room dose criterion 0.05 Sv (5 rem) total effective dose equivalent (TEDE). In March 1998, the staff briefed the Office of Nuclear Reactor Regulation Executive Team (ET) on its concerns related to the infiltration testing results and other aspects of control room habitability. The ET directed the staff to work with the Nuclear Energy Institute (NEI) to resolve the issues. Pursuant to this direction, the staff co-hosted, with NEI and the Nuclear Heating Ventilation and Air Conditioning Users Group (NHUG), a workshop on control room habitability in July 1998. Following that workshop, NEI agreed to form a task force to address control room habitability. In August 1999, NEI submitted for staff review and comment a draft of a proposed NEI document intended to address this issue. This document, NEI 99-03, entitled, "Control Room Habitability Assessment Guidance," did not adequately address the staff's concerns. In response to the staff concerns, NEI agreed in December 1999 to restructure NEI 99-03. During the period January through June 2000, the NEI task force met with the NRC staff in a series of public meetings to resolve outstanding issues and to discuss the content of NEI 99-03. A revision to NEI 99-03 revision was sent to the staff on October 13, 2000. The staff reviewed the October 13, 2000, revision and determined that, while there was much agreement on positions taken in the document, areas remained where the staff and industry were in disagreement. The staff determined, and NEI agreed, that the staff should reflect its position in formal regulatory guidance with outstanding issues resolved through the public comment process.

In June 2001 NEI issued Revision 0 of NEI 99-03, "Control Room Habitability Assessment Guidance." This version was substantially the same as the October 13, 2000, draft reviewed by the NRC staff.

The NRC staff pursued a solution to the control room habitability issues with the NEI task force. The staff indicated its willingness to incorporate up-to-date information into its assessment of radiological analyses, consider possible changes in the radiological dose acceptance criteria and possible reductions in the conservatisms in control room habitability analyses.

NEI did not commit to making this industry initiative binding on individual utilities. The staff believed that a voluntary approach would not adequately resolve its concerns and that some generic approach would be needed. A Generic Letter would request licensees to take action to evaluate, in light of the ASTM E741 testing results to date, how licensees meet the requirements of GDC-19 with respect to unfiltered inleakage to their control room envelopes.

During staff interactions with the NEI issue task force, many issues were discussed. The staff believed that additional regulatory guidance was necessary in order that control room habitability issues were addressed in a complete and thorough manner. In addition, it was the staff's opinion that the regulatory information associated with control room habitability needed to be updated to reflect current knowledge. In meetings with the NEI Task Force on Control Room Habitability, changes to design basis accident radiological analysis assumptions were discussed. For those facilities whose licensing basis is based upon the TID-14844 source term, the staff and industry believed that it was necessary to consolidate existing information and to reflect current knowledge into one regulatory guide. For those licensees that implement an alternative source term as allowed by 10 CFR 50.67, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provided guidance for performing control room radiological analyses. These regulatory guides provided the industry and public more realistic assumptions for performing radiological analyses.

The staff also believed that creating regulatory guidance on meteorology for control room habitability assessments was necessary and appropriate. It had been almost 20 years since the staff updated its information on control room habitability. Various staff and industry studies had been conducted in those 20 years. These studies had identified issues which were addressed to only a limited extent in the previous guidance on control room habitability. A regulatory guide on control room habitability would assist licensees in their determination of the present state of the integrity of their control room envelope. Along with the control room habitability regulatory guide, an additional regulatory guide on control room envelope integrity testing would provide guidance to the industry on how plants may determine control room envelope integrity and continually demonstrate that integrity. Such regulatory guidance would utilize the information obtained from the testing that had already been conducted on 30 percent of the control room envelopes.

Therefore, control room habitability would be addressed through a Generic Letter and new Regulatory Guides on:

(1) Control room habitability,
(2) Control room envelope integrity testing,
(3) Meteorology for control room habitability assessments, and
(4) Design basis accident radiological analyses.

Additionally, to support licensees that begin testing the integrity of the control room envelope by measuring unfiltered inleakage, the staff proposed to the Technical Specifications Task Force (TSTF) changes to standard technical specifications on control room emergency ventilation systems. The staff discussed these changes with the NEI control room habitability task force. The staff considered resolution of this issue supportive of the NRR pillars of maintaining safety, increasing public confidence (both by restoring control room integrity to the level assumed in the facility's licensing basis), increasing effectiveness and efficiency of key NRC processes (via a generic approach to resolution rather than the current plant-by-plant approach), and reducing unnecessary regulatory burden and increasing realism (due to possible relaxation in certain analysis assumptions and acceptance criteria, based on current information).

Four draft regulatory guides, numbered DG-1111, DG-1113, DG-1114 and DG-1115, were issued for public comment. Proposed Generic Letter 2002- XX, "Control Room Envelope Habitability," (ADAMS accession number ML021430317) was published on May 9, 2002, at 67 FR 31385. The staff completed review and disposition of comments received during the public comment period and completed making necessary revisions to the draft guides and generic letter. Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, formerly DG-1111 was issued in June 2003. Regulatory Guide 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors, formerly DG-1113, Regulatory Guide 1.196, Control Room Habitability at Light-Water Nuclear Power Plants, formerly DG-1114; and Regulatory Guide 1.197, Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors, formerly DG-115, were issued in May 2003. Generic Letter 2003-01 was issued on June 12, 2003. The staff's proposed changes to technical specifications for control room emergency ventilation systems were included in Appendix B of Regulatory Guide 1.196.

During the finalization of the Regulatory Guides and the Generic Letter, NEI provided Revision 1 to NEI 99-03, "Control Room Habitability Assessment Guidance," March 11, 2003. The Generic Letter and Regulatory Guides referred to Revision 0 of NEI 99-03. Staff assessed the impact of Revision 1 and determined that revisions to the Generic Letter and Regulatory Guides were not necessary.

On December 30, 2002, NEI provided the Industry/ TSTF Standard Technical Specification Change Traveler TSTF-448, "Control Room Habitability," to the NRC for consideration. On July 1, 2003, the staff transmitted to NEI comments on Rev. 0 of TSTF-448. The staff held a meeting with the TSTF/NEI on July 11, 2003, to discuss its comments on Rev. 0. On August 19, 2003, Technical Specifications Task Force (TSTF) transmitted to the staff Rev. 1 to TSTF-448. On December 16, 2003, the staff provided comments and a request for additional information on Rev. 1. In a March 8, 2004, letter, the TSTF responded to the staff 's comments. In that letter the TSTF also identified what they considered to be beneficial revisions to TSTF-448. The TSTF indicated that if the staff agreed with these proposed revisions a formal revision to TSTF-448 would be provided.

Proposed Actions: This proposed action plan provides for staff activities involving the assessment of licensee's verification and confirmation that their facility meets GDC 19. Licensee have and are responding to Generic Letter 2003-01. In their responses they have been requested to confirm their licensing basis including the inleakage characteristics of the CRE when their control room ventilation systems are functioning in response to a radiological or hazardous chemical challenge. In addition, licensees have been requested to confirm that a fire will not prevent the control room operators from controlling the reactor from either the control room or the alternate shutdown panel. Licensees have also been requested to address the adequacy of their technical specifications to demonstrate control room habitability. Licensees were also requested to address whether they currently utilize compensatory actions such as KI or self-contained breathing apparatus in order to meet GDC 19 and, if they do, when such compensatory actions will be retired. Licensees were asked to identify if they thought that their facility was licensed such that it was not required to meet GDC 19 or its equivalent as presented in the draft GDCs or in the draft Principle Design Criteria. The staff will review the licensee 's response to the Generic Letter and, if necessary, develop requests for additional information as part of that assessment. In conjunction with the Davis Besse Lessons Learned, the staff will perform a survey of 3-4 plants to assess the manner in which licensees have responded to the Generic Letter. These surveys will be conducted at the plant sites. It is intended that the survey will give the staff a sense of licensee's verification and confirmation processes in developing their responses. It would provide the staff data on whether licensees have responded to the Generic Letter in the manner in which the staff expected. The survey would be utilized to establish the framework and the protocol for plant inspections and the TI should they be necessary. During the survey, the staff would be confirming the licensee's response. It is anticipated that these surveys will provide data as to whether plant inspections should be conducted. If it is determined that inspections are necessary, a temporary instruction will need to be developed. This TI will be utilized to conduct the inspections. The number of inspections to be conducted will be a function of the finding but it is anticipated that a minimum of 4-6 plants will be inspected. From these inspections, an overall assessment of licensee's responses to the Generic Letter may be made.

Originating Document: None.

Regulatory Assessment: The staff believes that the potential deficiencies in the control room habitability designs, operations, and analyses represent safety issues that warrant resolution. It is important to recognize that the objective of control room habitability requirements, such as those in GDC-19, is not to minimize operator exposure for the purposes of ALARA (which is controlled under 10 CFR Part 20), but to provide a habitable environment in which to take action to operate the reactor safely under normal conditions and to maintain it in a safe condition under accident conditions. The dose criterion of 5 rem whole body was selected as it was believed that operations personnel would not be distracted from necessary plant operations and would not unnecessarily evacuate the controls area due to concerns for their personal safety. Protection against smoke and other toxic gases is also necessary since these hazards could cause, in some cases, immediate physical impairment or incapacitation of control room operators. While toxic gases are considered in control room habitability analyses in accordance with the guidance in Regulatory Guide 1.78, the potentially toxic byproducts of fires and their impacts on control room habitability were not considered a problem in the past because of the presumed integrity of the control room envelope. In the past, a fire outside the control room was considered to have no impact upon the operators because smoke and toxic fire gases were never presumed to enter the control room envelope. If a fire occurred in the control room, the operators had the remote shutdown areas for controlling the reactor. Testing of the control room envelope's integrity has demonstrated that the perceived integrity does not exist. Consequently, some portions of the smoke issue may be covered under this action plan while other aspects may not. The staff considered the risk impacts of control room habitability and made a preliminary determination that control room habitability has not been addressed in current PRAs because:

(1) It has been assumed that the design basis was being met, and
(2) Quantification of the risk associated with failure to meet the design basis for control room habitability is not addressed by current metrics, methods, and risk experience data.

Current Status: As of December 31, 2003, approximately 65 percent of the plants provided 60 days responses to the Generic Letter. The remainder provided a complete response. As of June 24, 2005, the following plants have not provided final responses to the Generic Letter: Davis Besse, Ginna, Grand Gulf, Hatch, Indian Point Unit 3, Oyster Creek, Palisades, Perry, Pilgrim, VC Summer, and Susquehanna. Approximately 88 percent of the reactors have now provided completed responses. Approximately 5 percent of the reactors do not intend to test their CREs. Licensees that have performed integrated tracer gas leakage testing of their control room envelopes continue to inform the NRC staff of their findings in their response to the Generic Letter.

The staff has identified issues associated with the proposed technical specification in Appendix B of Regulatory Guide 1.196. The staff is proposing to revise the technical specification in the Appendix. The staff's proposed revision was reflected in a September 30, 2004, amendment issued to the Farley Plant and in a January 24, 2005, letter to the Technical Specification Task Force (TSTF). The January letter was the staff's response to the TSTF's changes to TSTF-448, Control Room Habitability, proposed in the TSTF's March 8, 2004, letter to the staff. On May 26, 2005, the staff met with the TSTF and the NEI Task Force on Control Room Habitability to discuss the basis for the proposed technical specification in the January 24 letter. During the meeting the TSTF provided their comments regarding the proposed technical specification. As a result of the meeting, the TSTF will propose a revision to TSTF-448. On August 18, 2005, the TSTF issued Revision 2 to TSTF-448. The staff is reviewing the Revision. A meeting will be scheduled with the TSTF to discuss Revision 2.

The staff has identified those licensees which have not tested their control room envelope, whose response to the Generic Letter is dependent upon approval of an Alternative Source Term amendment or approval of TSTF-448 or who have not committed to responding to the Generic Letter. Telephone calls have been made to such licensees and letters have been sent to some. These letters have been in the form of a request for additional information (RAI) for those who have not tested and a request for the submittal of the information requested by the Generic Letter for those who have not provided such information.

The staff needs to review the complete responses to the Generic Letter to determine whether licensees have actually confirmed and verified their licensing basis, that they have adequately determined the inleakage characteristic of the CRE and that reactor control may be maintained from the control room or the alternate shutdown panel in the event of a fire, radiological or hazardous chemical challenge. The staff also must determine when those facilities which are currently using compensatory actions such as self-contained breathing apparatus or KI and when they intend to retire those actions.

Contacts:

NRR Lead PM:	D. Jaffe, PD4-1/DLPM/NRR, 415-1439
NRR Technical Contacts:	R. L. Dennig SPSB/DSSA/NRR, 415-1156 J. J. Hayes, SPSB/DSSA/NRR, 415-3167 W. M. Blumberg SPSB/DSSA/NRR, 415-1083 H. Walker SPSB/DSSA/NRR, 415-2827 M. Kotzalas, SPSB/DSSA/NRR, 415-2737

References:

USNRC, Title 10 Code of Federal Regulations Part 50, Appendix A.
USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, 1980.
USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800.
L. Soffer, et al, "Accident Source terms for Light Water Nuclear Power Plants," NUREG-1465, 1995.
Murphy, K. G. and Campe, K. W., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19, " published in proceedings of 13th AEC Air Cleaning Conference.
Driscoll, J. W., "Control Room Habitability Survey of Licensed Commercial Nuclear Power Generating Stations," NUREG/CONTROL ROOM-4960, 1988.
DiNunno, et al, "Calculation of Distance Factors for Power and Test Reactor Sites," Technical Information Document TID-14844, USAEC, 1962.
USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," 2000.
American Society for Testing and Materials ASTM E741, "Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," 1993.

FIRE PROTECTION PROGRAM

TAC Nos. M40015, MA1139, MA1752, MA4507, MA4720, MA4721, MA9013, MA9014, MA9015, MA9016, MB1311, MB1313, MB2921, MB3203, MB6729, MB6932, MB7726, MC0514, MC2582, MC2584, MC2589,MC3184,MC3628, MC3796, MC4011, MC4395, MC5630, MC5660, MC6921

Last Update: 09/30/05
Lead NRR Division: DSSA
Supporting Divisions: DIPM, DRIP, and DRAA (RES)
Supporting Offices: OE, OGC

MILESTONES		DATE (T/C)
ITEM I: **RISK-INFORMED PERFORMANCE-BASED RULE (NFPA 805)**		
1.	Publish final rule	06/04 (C)
2.	Review of NEI implementation guide (NEI 04-02, Rev. F)	07/04 (C)
3.	Complete draft Regulatory Guide (RG) (DG-1139)	08/04 (C)
4.	Issue DG-1139 for public comment	09/04 (C)
5.	Public meeting on implementation, DG-1139, & inspection concepts	10/04 (C)
6.	Comment on draft fire PRA requantification report (NUREG/CR-6850)	11/04 (C)
7.	Receive DG-1139 public comments	12/04 (C)
8.	Commission grants enforcement discretion extension	01/05 (C)
9.	Develop inspection template task force guidance	01/05 (C)
10.	Issue draft pilot plant observation guidance in public mtg notice	01/05 (C)
11.	Public meeting on DG-1139, pilot plant reviews, NEI 04-02	02/05 (C)
12.	Received first licensee letter of intent to transition (Seven Duke NPPs)	02/05 (C)
13.	RIC Fire Protection Session - Implementation Challenges	03/05 (C)
14.	Incorporate public comments into final regulatory guide	04/05 (C)
15.	Form task force to develop draft inspection input	04/05 (C)
16.	Task force workshops to develop draft inspection template	04/05 (C)
17.	Request fee waiver for first pilot plant (Oconee NPP)	04/05 (C)
18.	Obtain DSSA approval on final RG for ACRS/CRGR Review	05/05 (C)
19.	Send final regulatory guide to ACRS, CRGR, & OGC for review	05/05 (C)
20.	ACRS meeting on final regulatory guide	05/05 (C)
21.	CRGR meeting on final regulatory guide	05/05 (C)
22.	Receive second letter of intent to transition to NFPA 805 (Five Progress Energy NPP’s)	06/05 (C)

MILESTONES		DATE (T/C)
23.	NRC response to Duke Power's letter of intent	06/05 (C)
24.	Complete HQ & Regional draft inspection input review	07/05 (C)
25.	Request fee wavier for second pilot plant (Shearon-Harris NPP)	07/05 (C)
26.	Review final fire PRA report (NUREG/CR-6850)	08/05 (C)
27.	Conduct Transition Pilot Program Public Meeting	08/05 (C)
28.	Incorporate ACRS, CRGR, & OGC comments into final regulatory guide	01/06 (T)
29.	NRC response to Progress Energy's letter of intent	09/05 (C)
30.	Draft Transition Pilot Plant Plan	10/05 (T)
31.	Conduct first pilot observation visit	11/05 (T)
32.	Provide input to IIPB on risk-informed fire protection inspection procedures for trial use	12/05 (T)
33.	Final Transition Pilot Program Plan	02/06 (T)
34.	Obtain final approval on regulatory guide and send for publication	03/06 (T)
35.	Conduct second pilot observation visit	03/06 (T)
36.	Complete risk-informed fire protection inspector training	12/06 (T)
37.	Receive Oconee LAR to transition to NFPA 805	06/07 (T)
38.	Receive Shearon-Harris NPP LAR to transition to NFPA 805	11/07 (T)
39.	Receive McGuire NPP LAR to transition to NFPA 805	06/08 (T)
40.	Receive Crystal River 3 NPP LAR to transition to NFPA 805	02/09 (T)
41.	Receive Catawba NPP LAR to transition to NFPA 805	06/09 (T)
42.	Receive Robinson NPP LAR to transition to NFPA 805	05/10 (T)
43.	Receive Brunswick NPP LAR to transition to NFPA 805	08/11 (T)
ITEM II: **POST-FIRE SAFE-SHUTDOWN CIRCUIT ANALYSIS RESOLUTION**		
1.	Suspend selected circuit inspections	11/00 (C)
2.	Perform and analyze circuits test data	01/01 (C)
3.	Issue Regulatory Issue Summary (RIS) 2004-03, Revision 0 on circuit analysis	03/04 (C)
4.	Revise inspection procedures for Regions' review	06/04 (C)

MILESTONES		DATE (T/C)
5.	Conduct NRC inspector training	07/04 (C)
6.	Conduct public meeting to discuss NEI 00-01	07/04 (C)
7.	Develop primarily circuit screening tool	08/04 (C)
8.	Conduct public meeting in Atlanta	10/04 (C)
9.	Conduct public meeting on NEI 00-01 comments	11/04 (C)
10.	Issue revised inspection procedures	12/04 (C)
11.	Issue revised MC 0305 Section 6	12/23 (C)
12.	Resume circuit inspections	01/05 (C)
13.	Issue RIS 2004-03, Revision 1 on circuit analysis	12/04 (C)
14.	Issue draft RIS to provide clarification of regulatory requirement issues associated with post-fire safe-shutdown circuit analysis and protection - for 60-day public comment period	05/05 (C)
15.	Receive and address comments on circuit issues/compliance	09/05 (C)
16.	Issue final RIS on circuit issues/compliance	12/05 (T)
17.	Prepare draft GL for management concurrence	06/05 (C)
18.	Issue GL for public comment	10/05 (C)
19.	Conduct public meeting on regulatory requirements RIS	08/05 (C)
20.	Conduct public meeting on GL	01/06 (T)
21.	Issue final GL on circuit issues terminology	06/06
ITEM III: **OPERATOR MANUAL ACTIONS RULEMAKING**		
1.	Prepare proposal for rulemaking plan	03/03 (C)
2.	Commission approval of rulemaking plan	09/03 (C)
3.	Publish interim criteria for enforcement discretion for public comment	11/03 (C)
4.	Conduct public meeting for comments on draft interim criteria	11/03 (C)
5.	ACRS fire protection Subcommittee briefing on technical basis for interim criteria	04/04 (C)
6.	Conduct public meeting on requirement for detection/suppression	06/04 (C)
7.	Submit draft interim criteria for enforcement discretion to Office of Enforcement (OE)	07/04 (C)
8.	Re-issue modified interim criteria for enforcement discretion to OE	09/04 (C)
9.	Brief the ACRS fire protection Subcommittee and Full Committee	10-11/04 (C)

MILESTONES	DATE (T/C)
10. Issue interim enforcement discretion (NRC decided to rely on current enforcement discretion on circuits for manual actions)	12/04 (C)
11. Submit proposed rule to Commission	12/04 (C)
12. Issue proposed rule for public comment with draft Regulatory Guide (RG)	02/05 (C)
13. Conduct public meeting to further solicit comments	04/05 (C)
14. Conduct public meeting to discuss staff recommendation to withdraw the rulemaking and to convey closure plan	09/05 (C)
15. Submit policy paper to Commission recommending withdrawal of the proposed rule.	12/05 (T)
16. Assuming withdrawal of the rule, issue Regulatory Issue Summary.	04/06 (T)
17. Assuming withdrawal of the rule, provide input to IIPB to revise IP71111.05T.	04/06 (T)
18. Provide input to OE to revise Enforcement Manual section 8.1.7.1. (This has a nexus with fire-induced circuit discretion and has overall considerations with our NFPA-805 enforcement discretion policy.)	12/05 (T)
19. Assuming withdrawal of the rule, revise internal staff guidance NUREG-0800, SRP section 9.5.1 to include acceptance criteria.	TBD
ITEM IV: **EMERGING FIRE PROTECTION ISSUES RESOLUTION**	
1. Hemyc and MT, fire barrier performance qualification	
○ Review 1-hour Hemyc fire performance test report	04/05 (C)
○ Meet with industry/licensees on Hemyc	04/05 (C)
○ Issue generic communication on Hemyc	04/05 (C)
○ Issue draft Generic Letter on Hemyc and MT	07/05 (C)
○ Review 3-hour MT, fire performance test report	05/05 (C)
○ Conduct public meeting on Generic Letter	09/05 (C)
○ Conduct CRGR meeting on Generic Letter	11/05 (T)
○ Conduct ACRS meeting on Generic Letter	12/05 (T)
○ Issue final generic communication on Hemyc and MT	03/06 (T)
2. Epoxy coatings	
○ Review NEI white paper	01/05 (C)
○ Develop response and have NRC Regional Offices review	04/05 (C)
○ Issue response to NEI white paper	08/05 (C)

MILESTONES		DATE (T/C)
3.	Manual lockout of automatic carbon dioxide fire suppression systems	
	○ Review NEI white paper	01/05 (C)
	○ Develop response and have NRC Regional Offices review	04/05 (C)
	○ Issue response to NEI white paper	07/05 (C)
ITEM V: **REGULATORY TOOLS DEVELOPMENT**		
1.	Quantitative Fire Hazard Analysis Tools - NUREG-1805	
	○ Research and write text and spreadsheets	01/01 (C)
	○ Publish draft NUREG-1805	06/03 (C)
	○ Present NUREG-1805 at EPRI meeting in Maryland	09/03 (C)
	○ Present NUREG-1805 at EPRI meeting in Florida	09/04 (C)
	○ Publish softbound advanced copy of NUREG-1805	11/04 (C)
	○ Conduct public meeting on applications of NUREG-1805	11/04 (C)
	○ Publish hardbound NUREG-1805 with final spreadsheets	04/05 (C)

Description: Today the fire protection programs (FPPs) at U.S. nuclear power plants have the primary goals of minimizing both the probability of occurrence, and consequences of fire. To meet these goals, the FPPs are designed to provide reasonable assurance that a fire will not prevent the performance of necessary safe shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The primary FPP objectives for operating reactors are to:

- Prevent fire from starting,
- Detect, rapidly control, and promptly extinguish those fires that do occur, and
- Protect structures, systems, and components important to safety so that a fire that is not promptly extinguished will not prevent the safe shutdown of the plant.

The FPP objectives at plants that have permanently ceased operations are to:

- Reasonably prevent fires from occurring,
- Rapidly detect, control and extinguish those fires that do occur that could result in a radiological hazard, and
- Ensure that the risk of fire-induced radiological hazards to the public, environment and plant personnel is minimized.

The challenges within the FPP stem from (1) the fact that we have prescriptive regulations that are subject to different interpretations and are not always able to be enforced in a clear and consistent way, and (2) the fact that licensees have varying degrees of specificity in their licensing basis and in some cases are substantially different, which can also lead to different interpretations of regulatory intent. The activities described below address these challenges with the objective of achieving the strategic plan.

ITEM I: RISK-INFORMED PERFORMANCE-BASED RULE IMPLEMENTATION (NFPA 805)

Historical Background: The goal of revising 10 CFR 50.48 is to allow licensees to adopt a risk-informed, performance-based approach to fire protection as described in the consensus standard NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants.” The revised rule provides a means to re-establish poorly defined fire protection licensing bases and enables licensees to manage their fire protection programs with minimal regulatory intervention. NEI developed NEI 04-02, “Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c).” and NEI 00-01, Rev. 1, “Guidance for Post-Fire Safe Shutdown Analysis”. The staff will endorsed this guidance in the final regulatory guide RG-XXXX, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants.”

Proposed Actions: See Milestone chart.

Originating Document: SECY Paper 00-0009, “Rulemaking Plan, Reactor Fire Protection Risk-Informed, Performance-Based Rulemaking”, dated January 2000.

Regulatory Assessment: See Historical Background.

Current Status: The staff has worked with NEI to address ACRS comments on the final regulatory guide and NEI Implementation Guidance and met with ACRS in early October to discuss changes. SPSB has raised issue with the risk acceptance criteria within the regulatory guide and SPLB is working with SPSB to resolve their concern. These concerns may delay publication of the regulatory guide until resolved. Draft Inspection Procedures have been developed and are being distributed to the Regions for formal comment. The Transition Pilot Program has held it’s kickoff meeting with the two pilot plants, Oconee and Shearon-Harris NPP and scheduled the first two observation visits with both Utilities for November and next March. RES has completed publication of the fire PRA methodology, but has delayed the public comment period for the verification and validation of fire model report until next summer.

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References:

Draft Final Regulatory Guide RG-X.XXX, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants”, dated September 2005.

NEI 04-02, Rev. 1, “Guidance for Implementing a Risk-informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)”, dated September 2005

NEI 00-01, Rev. 1, "Guidance for Post-Fire Safe Shutdown Analysis," dated January 2005.

NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities,” dated September 2005.

NUREG/BR-0312, “The Alternate Fire Protection Regulation”, dated September 2004.

Draft Regulatory Guide DG-1139, “Risk-Informed, Performance-Based Fire Protection For Existing Light-Water Nuclear Power Plants”, dated October 2004.

SECY Paper 02-132, “Proposed Rule: Revision of 10 CFR 50.48 to Permit Light-Water Reactors to

Voluntarily Adopt National Fire Protection Association (NFPA) Standard 805, “Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants,” 2001 Edition (NFPA 805) as an Alternative Set of Risk-Informed, Performance-Based Fire Protection Requirements”, dated July 2002.

SECY Paper 00-0009, “Rulemaking Plan, Reactor Fire Protection Risk-Informed, Performance-Based Rulemaking”, dated January 2000.
NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants,” 2001 Edition, National Fire Protection Association, Quincy, Massachusetts.

Staff Requirements Memorandum M040511A, “Affirmation of SECY-04-0050 - Final Rule: Revision of 10 CFR 50.48 to Allow Performance-based Approaches Using National Fire Protection Association (NFPA) Standard 805, "Performance-based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition”, dated May 2004.

ITEM II: POST-FIRE SAFE-SHUTDOWN CIRCUIT ANALYSIS RESOLUTION

Historical Background: The goal of the Post-Fire Safe-Shutdown Circuit Analysis program is to clarify regulatory expectations with respect to circuit analyses and provide guidance to licensees and NRC inspectors on a risk-informed approach to inspection of post-fire safe-shutdown spurious actuations resulting from failure of circuits. The guidance documents, developed in cooperation with industry, will be used to bring clarity to a long-standing unresolved issue.

Proposed Actions: See Milestone chart.

Originating Document: Information Notice 99-17, “Problems Associated with Post-Fire Safe-Shutdown Circuit Analysis”, dated June 1999.

Regulatory Assessment: See Historical Background.

Current Status: RIS 2004-03, Revision 1, “Risk-Informed Approach for Post-Fire Safe-Shutdown Circuit Inspection” (ML042440791) has been issued for use and the revised inspection procedures have been issued to reflect the RIS. NEI 00-01, “Guidance for Post-Fire Safe Shutdown Circuit Analysis,” Revision 1 (ML050310295) has been issued final. This guidance document will be endorsed for risk-informed fire protection programs in the NFPA 805 regulatory guide. The deterministic methods in NEI 00-01 will be endorsed, with qualifications, in Regulatory Issue Summary 2005-XX, “Clarification of Post-Fire Safe-Shutdown Circuit Regulatory Requirements.” This RIS has been issued for public comment, the comments have been received and the final RIS is in preparation. In addition, a generic letter (Generic Letter 2005-XX, “Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations”) was issued for public comment to confirm regulatory requirements for analyzing multiple spurious actuations. NEI 04-06, “Guidance for Self-Assessment of Circuit Failure Issues,” is still being reviewed by the staff for possible endorsement.

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References:

NRC Bulletin 75-04, “Cable Fire at Browns Ferry Nuclear Power Station”, dated March 1975.

NRC Bulletin 92-01, “Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free From Fire Damage”, dated June 1992.

Information Notice 84-09, “Lessons Learned From NRC Inspections of Fire Protection Safe Shutdown Systems (10 CFR 50, Appendix R)”, dated February 1984.

Information Notice 84-09r1, “Lessons Learned From NRC Inspections of Fire Protection Safe Shutdown Systems (10 CFR 50, Appendix R)”, dated March 1984.

Information Notice 99-17, “Problems Associated with Post-Fire Safe-Shutdown Circuit Analysis”, dated June 1999.

"Circuit Analysis-Failure Mode and Likelihood Analysis,” A Letter Report to USNRC, Sandia National Laboratory, Albuquerque, New Mexico, ADAMS Accession # ML010450362, dated May 8, 2000.

NUREG/CR-6776, "Cable Insulation Resistance Measurements Made During Cable Fire Tests,” Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC, dated June 2002.

NUREG/CR-6834, "Circuit Analysis - Failure Mode and Likelihood Analysis,” Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC, dated September 2003.

Draft NUREG-1778, “Knowledge Base for Post-Fire Safe-Shutdown Analysis”, dated January 2004.

NEI 00-01, Rev. 0, "Guidance for Post-Fire Safe Shutdown Analysis," dated May 2003.

NEI 04-06, Rev. G, “Guidance for Self-Assessment of Circuit Failure Issues”, dated March 2004.

Regulatory Issue Summary 04-03, “Risk-Informed Approach for Post-Fire Safe-Shutdown Associated Circuit Inspections”, dated March 2004.

Regulatory Issue Summary 04-03, Rev. 1, “Risk-Informed Approach for Post-Fire Safe-Shutdown Circuit Inspections”, dated December 2004.

Regulatory Issue Summary 2005-XX, Draft, “Clarification of Post-Fire Safe-Shutdown Circuit Regulatory Requirements”, dated May 2005.

Generic Letter 05-XX, Draft, “Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations”, dated June 2005.

ITEM III: OPERATOR MANUAL ACTIONS RULEMAKING

Historical Background: The goal of the Operator Manual Actions Rulemaking is to revise Appendix R, Section III.G.2 and add new Section III.P to allow ex-Control Room operator manual actions as a Section III.G.2 compliance option if they conform to criteria to demonstrate their acceptability and meet the requirement for fire detectors and automatic fire suppression systems. The revised rule will provide reasonable assurance that post-fire operator manual actions will maintain the ability to achieve safe shutdown. Guidance for evaluating the actions will be provided in a Reg. Guide so that they can be uniformly evaluated by licensees and inspectors. SECY 03-0100 provided the primary purposes for the rulemaking. Based on public comments on the proposed rule, the staff will recommend withdrawal of the proposed rule. The staff briefed NRC upper management and the Commission Technical Assistants in August 2005 to convey our recommendation. The staff held a public meeting in September 2005 with the public to convey the same recommendation.

For further information, see Attachment 2, “NRR Rulemaking” DRIP RM#616.

Proposed Actions: See Milestone chart.

Originating Document: SECY Paper 03-0100, “Rulemaking Plan on Post-Fire Operator Manual Actions”, dated June 2003.

Regulatory Assessment: See Historical Background.

Current Status: The staff is working on the policy paper recommending withdrawal of the proposed rule. This policy paper is planned to be completed by the end of 2005 for Commission consideration. Concurrent with the policy paper development, the staff is working on elements of the closure plan in the event the Commission accepts the recommendation. One closure plan element is a Regulatory Issue Summary informing licensees of the rule withdrawal, reiterating regulatory compliance expectations, reaffirming that an unapproved operator manual action is not one of the three means for compliance, and informing licensees of the end date for enforcement discretion. A draft version of the RIS is in development. Another closure plan element is a revision to IP 71111.05T to reflect rule withdrawal, however inspections will continue in accordance with our ROP. The closure plan also includes a revision to internal staff guidance projected to be made late 2006 following completion of a NUREG from RES containing acceptance criteria.

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David Diec, RPRP, 415-2834

RES Technical Contact: Jimi Yerokun, PRAB/DRAA, 415-6009

References:

Letter, NRC Chairman to Congressman Markey and Dingell, dated May 16, 2004.

SECY Paper 03-0100, “Rulemaking Plan on Post-Fire Operator Manual Actions”, dated June 2003.

SRM “Staff Requirements - SECY-04-0233 - Proposed Rulemaking - Post-Fire Operator Manual Actions (RIN 3150 AH-54)”, dated January 18, 2005.

Communication Plan for Fire Protection Operator Manual Actions, dated September 7, 2005. January 26, 2005.

Draft Regulatory Guide DG-1136, “Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire”, dated February 2005.

Summary of Meeting to Receive Stakeholder Feedback, dated May 16, 2005.

Public Comments on Proposed Rule, dated May 16, 2005.

ITEM IV: EMERGING FIRE PROTECTION ISSUES RESOLUTION

Historical Background: To facilitate the resolution of emerging issues, a protocol is established between the NEI/industry and the NRC. This process is intended to identify emerging fire protection generic issues, discuss priorities and schedules, and facilitate improved coordination without affecting NRC's oversight responsibility. Issues are tracked, prioritized, and given an action status by the responsible party. Stakeholders are kept informed through publicly issued meeting summaries. This process was modeled after the protocol applied in the resolution of steam generator issues.

Hemyc Fire Barrier System - RES performed confirmatory testing on multiple configurations of 1-hour Hemyc and 3-hour MT fire barrier systems to determine if the material can be rated as a fire barrier based on approved test methods. The test results revealed that Hemyc and MT fire barrier systems failed to provide the protective function intended for compliance with fire protection regulations, for the configurations tested using staff-approved test methods.

Epoxy Floor Coatings - Epoxy floor coatings are used in all nuclear power plants. There is a debate as to whether epoxy coating has been applied to floors in such a manner (numerous coats applied over the years, thickness greater than those tested, etc.) that it should be included in a plant' s combustible loading calculations.

CO_2 - The issue concerns the manual lockout of automatic carbon dioxide (CO_2) fire suppression systems. CO_2 fire suppression systems present a hazard to plant personnel if they accidently discharge. Consideration of this hazard must be weighed against the effectiveness of the system and the manual action, some licensees are utilizing to activate them.

Proposed Actions: See Milestone chart.

Originating Document: Not applicable.

Regulatory Assessment: See Historical Background.

Current Status: RES completed testing for both conduit and cable tray configurations of the Hemyc fire barrier system and conduit configurations for MT. (RES issued a test report and notified the stakeholders). NRR issued an IN, while a draft GL has been published for public comment and is expected to be issued by early 2006. A communication plan has been prepared to further detail the activities surrounding this issue (ML050670609).

In accordance with the NEI/NRC Issue Management Protocol, NEI agreed to offer industry evaluation of the combustibility of epoxy coatings in nuclear plants. NEI provided that input and the staff issued its response to the NEI white paper (ML052020025).

NEI issued a white paper on the CO_2 fire suppression systems topic in January of 2005. The staff provided comments on the NEI white paper in July 2005 (ML051740050).

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Robert Wolfgang, SPLB, 415-1624
Angie Lavretta, SPLB, 415-3285

RES Technical Contact: Mark Salley, PRAB/DRAA, 415-2840

References:

Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis", dated July 1998.

Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety", dated December 1998.

Information Notice 03-19, "Unanalyzed Condition of Reactor Coolant Pump Seal Leakoff Line During Postulated Fire Scenarios or Station Blackout", dated October 2003.

NRC Inspection Manual, Chapter 0609, Appendix F, "Determining Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Findings", dated February 27, 2001.

NUREG-0800, Section 9.5.1, Rev. 4, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants Fire Protection Program", dated October 2003.

Test Plan for Heymc and MT, Fire Barrier Performance, Package.

Information Notice 05-07, "Results of Hemyc Electrical Raceway Fire Barrier System Full Scale Fire Testing," dated April 1, 2005.

Draft Generic Letter 2006-XX, "Impact of Potentially Degraded Hemyc and Mt Fire Barriers on Compliance with Approved Fire Protection Programs," dated July 2005.

NEI White Paper on Epoxy Coatings, June 28, 2004.

NRC Staff Review of the Westinghouse Owners Group (WOG) Request for Enforcement Discretion for Reactor Coolant Pump (RCP) Seal Performance Findings in Triennial Fire Protection Inspections, November 12, 2004.

ITEM V: REGULATORY TOOLS DEVELOPMENT

Historical Background: Advancements have been made with several regulatory tools that the Staff can utilize to better ensure fire protection safety. For example, the fire protection SDP was updated to simplify the process, without reducing safety, by screening out very low risk findings that do not warrant further NRC involvement.

NUREG-1805 will be a key component in providing a simplified risk-informed methodology for use by inspectors to assess potential fire hazards that could cause critical damage to safe shutdown components.

Other regulatory tools are being developed by the Office of Research (RES) but will no longer be tracked in this report.

Proposed Actions: See Milestone chart.

Originating Document: Not applicable.

Regulatory Assessment: See Historical Background.

Current Status: The simplified risk-informed methodology was developed and issued for public comment as draft NUREG–1805, "Fire Dynamics Tools (FDT^s) Quantitative Fire Hazard Analysis Methods for the

U.S. Regulatory Commission Fire Protection Inspection Program". Comments have been received and incorporated, as appropriate, and the final NUREG was issued in November 2004. Errata and revised spreadsheets were placed on the public website (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1805/final-report/index.html) in August 2005.

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References:

Dey, M., A. Hamins, and M. Steward, "International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications: Summary of 5th Meeting," NISTIR 6986, National Institute of Standards and Technology, Gaithersburg, Maryland, dated September 2003.

NUREG/CP-0170, "International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications: Summary of Planning Meeting," Held at University of Maryland College Park, Maryland, October 25-26, 1999, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC, dated March 2000.

NUREG/CP-0173, "International Collaborative Project to Evaluate Fire Models for Nuclear Power Plant Applications: Summary of 2nd Meeting," Held at Institute for Protection and Nuclear Safety, Fontenay-aux-Roses, France, June 19–20, 2000, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC, dated March 2001.

NUREG-1805 (Advanced Copy), "Fire Dynamics Tools - Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspections Program", dated November 2004.

Draft NUREG/CR-6850, Vol. 1 & 2, "Fire PRA Methodology for Nuclear Power Facilities", dated October 2004.

PHASED APPROACH TO PRA QUALITY

TAC No. MC1747

Last Update: 09/30/05
Lead Division: DSSA
Supporting Office: RES
CTL: N/A
GSI No.: N/A

TASK	MILESTONE	DATE (T/C)
1.1	Identify current risk-informed applications (e.g., 50.69)	03/31/04 (C)
1.2	Specify PRA quality needs for each risk-informed application	12/30/04 (C)
1.3	Phase 2 Guidance Document Schedule	12/31/04 (C)
1.4	Revise application-specific guidance to address PRA quality	06/30/06 (T)
	PRA quality (RG 1.200) pilots for internal events	02/28/05 (C)
	Implementation - quality for internal events PRA (Note 1)	12/30/06 (T)
	Standards development - ANS fire PRA	06/30/06 (T)
	NRC endorsement - ANS fire PRA standard	06/30/07 (T)
	Implementation - quality for fire PRAs (Note 1)	06/30/08 (T)
	Standards development - ANS low-power & shutdown PRA	In ballot
	NRC endorsement - ANS low-power & shutdown standard	06/30/06 (T)
	Implementation - quality for low-power & shutdown PRAs	06/30/07 (T)
1.5	Development of Prioritization Process for Staff Review	12/30/05 (T)
1.6	Phase 2 Implementation Schedule	Note 4
1.7	Develop Phase 3 guidance	12/31/08 (T)
2.1	Alternate methods & treatment of uncertainties, draft NUREG	12/30/04 (C)
2.2	Standards development - ANS external events PRA	(C)
	NRC endorsement - ANS external events standard	Note 3
	Implementation - quality for external events PRAs (Note 1)	Note 1

Note 1: It is assumed that the standards documents will lag behind the guidance documents for the applications. It is further assumed that a delay of one year between the completion of the quality guidance documents and that time at which each application is expected to conform to those documents is sufficient for the review of the associated PRA elements to be completed. Furthermore, this time delay allows for the staff infrastructure necessary to transition to Phase 2 to be developed.

Note 2: Primary lead organization is RES with input from NRR.

Note 3: Initial staff review of ANS standard was issued for public comment in August 2004. ANS is revising the standard to respond to those comments, but it has not been issued to date.

Note 4: The schedule is dependent on the schedule for task 1.3. Based on informal feedback, the original proposal date of one year may be unrealistic given the resources available to perform the task.

Description: The objective of the phased approach to stabilizing the PRA quality expectations and requirements is to achieve an appropriate level of PRA quality for NRC's risk-informed regulatory decisionmaking. The phased approach defines the needed PRA quality for current or anticipated applications and the process for achieving this quality, while allowing risk-informed decisions to be made using currently available methods until all the necessary guidance documents defining the PRA quality are developed and implemented.

It is expected that meeting the phased approach objective will result in the following:

a. Industry movement towards improved and more complete PRAs

b. Increased efficiencies in the staff's review of risk-informed applications

c. Clarification of expectations for 10 CFR 50.46 and 10 CFR 50.69 rulemakings

d. Continued near-term progress in enhancing safety through the use of available risk-informed methods while striving for increased effectiveness and efficiency in the longer term

An additional objective is to ensure that activities are coherently and properly integrated such that they complement one another and continue to meet the 1995 PRA Policy Statement.

There are three Phases defined. Each phase is characterized in terms of the available guidance documents relative to the risk-informed activities. What distinguishes the phases is the availability and implementation of technical guidance documents that address the use and quality of the PRA with scope and level of detail necessary to support an application.

Phase 1 corresponds to the current status of the use of PRA in regulatory decisionmaking. Guidance for using PRA in regulatory decisionmaking exists in the form of regulatory guides such as RG 1.174, 1.175, 1.176, 1.177, and 1.178. These guides address PRA quality in a general way, stating that the quality of the PRA must be "commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process." They do not, however, provide detailed guidance on what is technically adequate for the defined scope. The review of the base PRA used to support applications has been based on the reviewers' experience guided by previous staff reviews such as those performed on the Individual Plant Examinations (IPE) submittals, and on observations from peer reviews that were performed for the licensee. However, until recently there has been no formal guidance on PRA technical adequacy. The focus of the reviews has, in general, been on those aspects of the PRA that contribute to the evaluation of the change in the CDF and LERF associated with the application, with particular attention to those aspects of the licensee's PRA that have been identified as potential concerns in previous reviews.

Phase 2 corresponds to the situation where, for each general application type (such as risk-informed Inservice Inspection (ISI) applications, risk-informed technical specifications applications, and 10 CFR 50.69 applications), the baseline PRA that supports the application meets applicable consensus standards, such as the ASME PRA Standard as endorsed in RG 1.200. Furthermore, the PRA scope is such that all operational modes and initiating events that could change the regulatory decision substantially are included in the model quantitatively. Thus, for a specific application type to be considered Phase 2, guidance must be in place for (1) performing the PRA analyses needed to support the application, and (2) assessing whether the level of detail and technical adequacy of the PRA models for the significant modes of operation and initiating events (i.e., those whose inclusion could change the regulatory decision substantially) is sufficient to support the application.

In Phase 3, the regulatory framework is in place (i.e., guidance documents are available) for the operational modes and initiating events that could affect a decision for existing and planned risk-informed applications. Therefore, to transition to Phase 3, a licensee will need a PRA that is of sufficient scope (in terms of operational modes and initiating events) to address currently envisioned applications and will meet the requirements of the applicable industry consensus standards.

Background: The Commission, by publishing its Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (Ref. 1), reflected its belief that an overall policy on the use of probabilistic risk assessment (PRA) methods in nuclear regulatory activities should be established so that the many potential applications of PRA would be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. Furthermore, the Commission stated its belief that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach. With implementation of this policy statement, the Commission also recognized, and encouraged, continuation of industry initiatives to improve PRA methods, applications, and data collection to support increased use of PRA techniques in regulatory activities.

Since the PRA Policy Statement was issued, a number of risk-informed activities have been undertaken and a number of documents have been written by both the staff and industry that provide guidance on the use of PRA information in the risk-informed reactor regulatory activities, and on PRA quality.

- Reactor owners groups have been developing and applying a PRA peer review program for several years. In a letter dated April 24, 2000, the Nuclear Energy Institute (NEI) submitted NEI-00-02 (Ref. 2) to the NRC for review in the context of the staff's work to risk-inform the scope of special treatment requirements contained in 10 CFR Part 50 (discussed in SECY-99-256, Ref. 3).

  On August 16, 2002, NEI submitted draft industry guidance for self-assessments (Ref. 4) to address the use of industry peer review results in demonstrating conformance with the American Society of Mechanical Engineers (ASME) PRA standard. This additional guidance, which is intended to be incorporated into a revision of NEI-00-02 (per NEI, see Reference 4), contains:

  - Self-assessment guidance document
  - Appendix 1 (actions for industry self assessment)
  - Appendix 2 (industry peer review subtier criteria)

- PRA standards have been under development by the ASME and the American Nuclear Society (ANS). On April 5, 2002, ASME issued a standard for a full-power, internal events (excluding internal fire but including internal floods) Level 1 PRA and a limited Level 2 PRA, supplemented by addenda on December 5, 2003 (Ref. 5). In December 2003, ANS issued a standard for external events (Ref. 6), which addresses seismic, high wind, external flood, and other (e.g., aircraft crash, chemical release) hazards. In the future, ANS plans to issue standards for PRAs for evaluating internal fire risk and risk from low-power and shutdown modes of operation.

- RG 1.200 (Ref. 7), “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” issued for trial use. RG 1.200 is expected to provide the level of confidence that the technical adequacy of the PRA is sufficient to support the identified applications such that an in-depth technical review by NRC staff would not be needed to ensure its quality to support the applications. This regulatory guide (RG) will allow staff members to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application. Consequently, RG 1.200 will provide for a more focused and consistent review process.

On December 18, 2003, the Commission provided a staff requirements memorandum (SRM) (Ref. 8) regarding stabilizing PRA quality expectations and requirements. In the SRM, the Commission approved implementation of a phased approach to achieving an appropriate quality for PRAs for NRC’s risk-informed regulatory decisionmaking. This phased approach was described in an attachment to the SRM. The SRM also directed the staff to develop an action plan that would define a practical strategy for the implementation of the phased approach to PRA quality.

<u>Proposed Actions</u>: The milestones listed in the milestone table comprise the actions for this initiative. The only additional task is to develop a communication plan. The objectives of this plan are to, (1) explain the staff activities to stakeholders, (2) describe the staff's approach, and (3) provide a structure for communicating the messages to stakeholders. This communication plan is scheduled for development in the third quarter of the fiscal year 2005.

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

<u>Regulatory Assessment</u>: Not applicable.

<u>Current Status</u>: The December 30, 2004, milestone for Task 1.4, completion of the RG 1.200 pilots was delayed by two months due to the licensees; late submittals. However, this delay will not affect the plan completion date of December 31, 2008.

Work has begun on Task 1.5, with an initial meeting between DSSA and DLPM staff on December 17, 2004. The development of the prioritization process under Task 1.5 will be coordinated with ongoing activities within DLPM, scheduled to be completed for implementation in FY2006.

Task 1.6 provides for a phasing in of the expectations for submittals to allow licensees time to develop PRA models, and perform the necessary peer reviews or self assessments to demonstrate conformance with the appropriate standards, once those standards have been developed and endorsed by NRC. The schedule for this phasing in is dependent on the schedule in Task 1.3. Based on informal feedback received by the MSPI PRA quality task group, the original proposal of one year for a delay in full implementation by the licensees, may be unrealistic given the resources available to perform these tasks. This will be revisited as more experience is obtained.

A draft NUREG entitled “Guidance on Alternative Methods and the Treatment of Uncertainties in Risk-Informed Decisionmaking”, developed by RES, was issued for internal NRC staff review per the schedule (December 31, 2004). This has been reviewed by DSA staff and a revision to the document is planned for issuance for public comment in October 2005.

<u>Contacts</u>:

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RES Lead Contact: Mary T. Drouin, RES/PRAB, 415-6675

<u>References</u>:

1. USNRC, “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,“ *Federal Register*, Vol. 60, p. 42622 (60 FR 42622), August 16, 1995.

2. Nuclear Energy Institute, “Probabilistic Risk Assessment Peer Review Process Guidance,” NEI-00-02, Revision A3, March 20, 2000.

3. USNRC, SECY-99-256, “Rulemaking Plan for Risk-Informing Special Treatment Requirements,” October 29, 1999.

4. Letter from NEI, Anthony Pietrangelo, Director of Risk and Performance Based Regulation Nuclear Generation, to the USNRC, Ashok Thadani, Director of Office of Nuclear Regulatory Research, December 18, 2001.

5. American Society of Mechanical Engineers, “Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,” ASME RA-S-2002, April 5, 2002, and “Addenda to ASME RA-S-2002,” ASME RA-Sa-2003, December 5, 2003.

6. American Nuclear Society, “American National Standard External-Events PRA Methodology,” ANSI/ANS-58.21-2003

7. USNRC, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Regulatory Guide 1.200, for trial use, February 2004.

8. USNRC, COMNJD-03-0002 “Stabilizing the PRA Quality Expectations and Requirements.”