May 1, 1996

Mr. William Rasin, Vice President
Technical/Regulatory Division
Nuclear Energy Institute
1776 "Eye" Street
Washington, D.C. 20006

SUBJECT: NRR STATUS REPORT ON GENERIC ACTIVITIES

Dear Mr. Rasin:

Enclosed is the April 1996 status report of generic activities under the cognizance of the Office of Nuclear Reactor Regulation. The report contains two attachments detailing generic activities:

(1) generic activities anticipated to require sufficient staff resources to warrant development of an action plan (Action Plans);

(2) generic communications and compliance activities (GCCAs) that are potential generic issues that are safety significant, require technical resolution, and possibly require generic communication or action. GCCAs do not rise to the level of complexity that an action plan is required.

The next report is scheduled to be issued July 1996. A copy of this report and subsequent revisions will be placed in the NRC's Public Document Room. This will be the last report formally transmitted to you; future revisions will be sent as distribution copies. The staff is in the process of making an electronic version accessible through the Internet. When this occurs, you will be notified under separate correspondence. If you have any questions concerning this report, please contact Stephen Koenick at (301) 415-2841 [SSK2@NRC.GOV] and Thomas Greene at (301) 415-1175 [TAG@NRC.GOV].

Sincerely,

Original signed by

Brian K. Grimes, Acting Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure: Status Report

cc w/enclosure:
See next page

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INTRODUCTION

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

The report includes two attachments: 1) action plan and 2) generic communications under development and other generic compliance activities. Generic communications and compliance activities (GCCAs) are potential generic issues that are safety significant, require technical resolution, and possibly require generic communication or action.

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

Attachment 2, "Generic Communications and Compliance Activities," consists of three monthly status reports. 1) open GCCAs, 2) GCCAs added since the previous report, and 3) GCCAs closed since the previous report. The generic communications listed in the attachment include bulletins, generic letters, and information notices. Compliance activities listed in the attachment do not rise to the level of complexity that require an action plan, and a generic communication is not currently scheduled. For each GCCA, there is a short description of the issue, scheduled completion date, and name of cognizant staff.

Attachment 1

NRR ACTION PLANS

TABLE OF CONTENTS

BOILING WATER REACTOR INTERNALS

Last Update: 03/27/96
Lead NRR Division: DE
Supporting Division: DSSA

MILESTONES	DATE (T/C)
PART I: REVIEW OF GENERIC INSPECTION AND EVALUATION CRITERIA	
1. Issue summary NUREG	03/96C
2. Review BWRVIP Re-inspection and Evaluation Criteria	08/96T
3. Review of generic repair technology, criteria and guidance	08/96T
4. Review generic mitigation guidelines and criteria	08/97T
5. Review of generic NDE technologies developed for examinations of BWR internal components and attachments	08/97T
6. Other Internals reviews (mitigation measures, inspections and repairs)	06/97T

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

<u>Historical Background</u>: Significant cracking of the core shr[illegible] was first observed at Brunswick, Unit 1 nuclear power plant in September 1993. The NR[illegible] [illegible]fied licensees of Brunswick's discovery of significant circumferential cracking of the core shroud welds. In 1994, core shroud cracking continued to be the most significant of reported internals cracking. In July 1994, the NRC issued Generic Letter 94-03 which requires licensees to inspect their shrouds and provide an analysis justifying continued operation until inspections can be completed.

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

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

In late 1994, extensive cracking was discovered in the top guide and core plate rings of a foreign reactor. The design is similar to General Electric (GE) reactors in the U.S., however, there have been no observations of such cracking in U.S. plants. GE concluded that it was reasonable to expect that the ring cracking could occur in GE BWRs with operating time greater than 13 years.

In the special industry review group's report, that was issued in January 1995, ring cracking was evaluated. The NRC concluded that the BWRVIP's assessment was acceptable and that top guide ring and core plate ring cracking is not a short term safety issue.

Proposed Actions: The staff will continue to assess the scopes that have yet to be submitted by licensees concerning inspections or re-inspections of their core shrouds. The staff will also continue to assess core shroud inspection results and any appropriate core shroud repair designs on a case-by-case basis. The staff will issue separate safety evaluations regarding the acceptability of core shroud inspection results and core shroud repair designs. The staff has been interacting with the BWRVIP and individual licensees. In an effort to lower the number of industry and staff resources that will be needed in the future, it is important for the staff to continue interacting with the industry on a generic basis in order to encourage them to continue their proactive efforts to resolve IGSCC of BWR internals. The BWRVIP has submitted four generic documents, supporting plant-specific submittals, for staff review. The staff is ensuring that the generic reviews are incorporating recent operating experience on all BWR internals.

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

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

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

In October 1995, industry's special review group issued a report which the NRC staff's preliminary review indicates was not comprehensive. The NRC staff is preparing requests for additional information. In addition, the industry group is planning to submit reports on reinspection of repaired and non-repaired core shrouds by February 1996. It is important to have these reports prior to the spring 1996 outages in order to have agreed upon generic inspection criteria. In an effort to lower the number of industry and staff resources that will be needed in the future, it is important for the staff to continue interacting with the industry on a generic basis in order to encourage them to continue their proactive efforts to resolve IGSCC of BWR internals. The NRC is also reviewing new information submitted by GE on the safety significance of and recommended inspections for top guide and core plate ring cracking.

NRR Technical Contacts:	David Terao, EMCB, 415-3317
	Merrilee Banic, EMCB, 415-2771
	Kerri Kavanagh, SRXB, 415-3743
	Frank Grubelich, EMEB 415-2784
NRR Lead PM:	C. E. Carpenter, EMCB, 415-2169

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

Action Plan dated April 1995

REACTOR PRESSURE VESSEL ACTION PLAN

Last Update: 03/28/96
Lead NRR Division: DE

MILESTONES	DATE (T/C)
1. ISSUE SUPPLEMENT TO GL 92-01	5/95 (C)
2. COORDINATION WITH RESEARCH	7/97 (T)
3. NRC/INDUSTRY WORKSHOP ON RPV ISSUES	7/95 (C)
4. REVIEW OF GL 92-01 SUPPLEMENT 1, 1ST ROUND	10/95 (C)
5. NUREG 1511 RPV STATUS REPORT SUPPLEMENT 1	4/96 (T)
6. REVIEW OF GL 92-01 SUPPLEMENT 1, 2ND ROUND	12/96 (T)
7. NUREG 1511 RPV STATUS REPORT SUPPLEMENT 2	6/97 (T)
8. ISSUE OF RVID REVISION 1	6/96 (T)
9. ISSUE OF RVID REVISION 2	6/97 (T)
10. REVIEW AND EVALUATE THE PALISADES ANNEAL PLAN	6/96 (T)
11. OBSERVE INDUSTRY ANNEALING DEMONSTRATION	12/96 (T)
12. REVIEW PALISADES ANNEAL	8/98 (T)

Description: Appendix G to 10 CFR 50 and 10 CFR 50.61 establish requirements to prevent fracture of the reactor pressure vessel (RPV). These rules require licensees to project the amount of embrittlement of RPV materials. As a result of the review of responses to Generic Letter (GL) 92-01, the review of Palisades PTS issue, and recent inspections conducted at Combustion Engineering, several issues related to RPV evaluations have been identified. These issues can be summarized as follows:

(1) It appears that licensees may not have been aware of or considered all relevant information and data in previous assessments of their RPVs,

(2) The variability in copper and nickel chemical composition may be independent of weld heat number and is greater than previously recognized by the staff,

(3) The Palisades reactor vessel will be the first commercial nuclear vessel annealed in the U.S. to improve its fracture toughness.

Historical Background: In March 1992, the NRC issued Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)." As a result of the information provided by the licensees in response to GL 92-01, Revision 1, the staff issued NUREG-1511, "Reactor Pressure Vessel Status Report," and the Reactor Vessel Integrity Database (RVID). NUREG-1511 provides a summary of the critical issues and regulatory requirements involved in RPV structural integrity and the status of each RPV with respect to the regulatory requirements. The RVID contains all the data that was submitted by licensees to demonstrate compliance with the regulatory requirements. Since licensees provide data during the life of the plant to demonstrate their compliance with regulatory requirements, NUREG-1511 and the RVID will require periodic upgrading.

In April 1995, the staff completed its evaluation of the Palisades plant compliance with the pressurized thermal shock (PTS) rule, 10 CFR 50.61. The staff concluded that the Palisades RPV could be operated in compliance with the requirements of the PTS rule through the plant's 14th refueling outage, which was scheduled for late 1999. To extend the life of the Palisades RPV beyond 1999, the licensee for Palisades has begun to plan for annealing of the Palisades RPV. The staff will review the licensee's annealing plan prior to its implementation. The Palisades anneal is scheduled for the 1998 refueling outage. Prior to this anneal the industry will be performing demonstration anneals at the Marble Hill and Midland-2 sites.

As a result of information received during the Palisades PTS review, a meeting with Combustion Engineering and two inspections at the Combustion Engineering offices in Windsor, Connecticut, the staff determined that licensees may not have been aware of or considered all relevant information and data in previous RPV assessments. Based on the above finding, the staff concluded that the most effective way to resolve this issue was through a supplement to GL 92-01 requiring the licensees to collect all data relevant to their RPVs, and if there are data that they had not previously considered, to perform a reassessment of their RPV.

As a result of the data supplied in response to GL 92-01 and the Palisades PTS review, the Office of Nuclear Reactor Regulation requested in a letter dated August 11, 1995 that the Office of Nuclear Regulatory Research evaluate whether changes to the PTS rule or Regulatory Guide 1.99 are necessary.

Proposed Actions: Specific actions included in the generic action plans are: (1) issue Supplement 1 to GL 92-01, (2) coordination with RES on RPV integrity issues, (3) hold an NRC/industry workshop on RPV issues, (4) review first and second round of responses to GL 92-01 Supplement 1, (5) issue supplement 1 to NUREG-1511 in 1996 and issue supplement 2 to NUREG-1511 in 1997, (6) issue revision 1 of the RVID in 1996 and issue revision 2 of the RVID in 1997, (7) observe industry annealing demonstrations, (8) review and evaluate the Palisades annealing plan, and (9) review the Palisades anneal.

Originating Document: Memorandum from Jack R. Strosı.   to Ashok C. Thadani, NRR, August 9, 1995.

Regulatory Assessment: This plan would allow for resolution of the issues discussed above in about two years. The staff anticipates that it will take the industry and the NRC this long to collect and assess all the relevant data. The staff assessed the impact of increased variability in chemistry on the RT_{PTS} value of PWR reactor vessels in a memorandum from J.R. Strosnider to A.C. Thadani dated May 5, 1995. The staff's assessment indicates that there is no immediate cause for concern and that there is adequate time to perform a more rigorous assessment of the issue. Based on the staff's generic assessment of the impact of increased variability, the staff has concluded that this is an acceptable schedule.

Current Status: GL 92-01, Supplement 1 has been issued. NRC/Industry workshop has been completed. A request for research on RPV integrity issues has been issued. The Reactor Vessel Integrity Database (RVID) has been issued (NRC Administrative Letter 95-03) to all licensees and to all individuals requesting a copy. The staff has completed the review of licensees' initial responses to Supplement 1 to GL 92-01. The licensee for Kewaunee in a letter from Clark R. Steinhardt dated August 21, 1995 provided the only notable response. They provided three methods of analysis of their surveillance data that indicate the Kewaunee reactor vessel will be below the PTS screening criteria at the expiration of its license. The licensee for Ginna in a letter from dated October 11, 1995 has also submitted a revised PTS evaluation. The Kewaunee PTS evaluation is being reviewed by the staff. The staff has completed the review of the Ginna PTS evaluation, which is documented in a March 22, 1996 letter to the licensee. Based on the currently available chemistry and surveillance data, the Ginna reactor vessel is projected to be below the PTS screening criteria at the expiration of its license.

Consumers Power Company has submitted a number of sections of their Thermal Annealing Report. These section are currently under staff review. The staff issued a request for additional information on Section 3 "Fracture Toughness Recovery and Reembrittlement Assurance Program"and the licensee responded. Below is a summary of the docketed information regarding the review of the Palisades Annealing:

October 12, 1995 - Section 3, "Fracture Toughness Recovery and Reembrittlement Assurance Program"
November 16, 1995 - Request for Additional Information (RAI) regarding Section 3
December 1, 1995 - Section 1.6, "Proposed Annealing Equipment" and Section 1.9, "ALARA Considerations"
December 12, 1995 - Section 1.1, "General Considerations" and Section 1.2 "Description of Reactor Vessel"
December 18, 1995 - Response to November 14, 1995, RAI regarding Section 3
January 12, 1996 - Section 1.3 "Equipment, Components, and Structures affected by Thermal Annealing" and Section 1.5 "Annealing Method, Instrumentation and Procedures"
February 2, 1996 - Section 1.8, "Proposed Annealing Conditions", Section 2.2, "Inspection Program" and Section 2.3, "Testing Program"
February 5, 1996 - Section 1.4, "Thermal Annealing Operating Conditions" and Section 2.1, "Monitoring the Annealing Process"
The licensee's schedule for submittal of the remaining sections (Section 1.7, "Thermal and Stress Analyses" and Section 1.10, Summary of Thermal Annealing Operating Plan") of the Thermal Annealing Report (excluding results from the Marble Hill ADP) has slipped from March 22, 1996 to the end of March because the license has requested its contractor to perform additional analyses to resolve licensee questions.

NRR Technical Contact:	Barry J. Elliot, EMCB, 415-2709	
NRR Lead PM:	Daniel G. McDonald, PD1-'	-1408
	Marsha K. Gamberoni, PD3	15-3024

References:

Memorandum to Ashok C. Thadani from Jack R. Strosnider, "Plan for Addressing Generic Reactor Pressure Vessel Issues," August 9, 1995.

NUREG-1511, "Reactor Vessel Status Report," December 1994.

Generic Letter 92-01, Revision 1, (and Supplement 1) March 6, 1992 and May 19, 1995.

Memorandum to Ashok C. Thadani from Jack R. Strosnider, "Assessment of Impact of Increased Variability in Chemistry of the RT_{PTS} Value of PWR Reactor Vessels," May 5, 1995.

NRC Administrative Letter 95-03, August 4, 1995

MOTOR-OPERATED VALVES ACTION PLAN

Last Update: 3/29/96
Lead NRR Division: DE

MILESTONES	DATE (T/C)
Regulatory Improvements: (1) Staff is working with ASME to improve the inservice testing requirements in the ASME Code and (2) Staff is working with OM to develop guidelines for periodic verification of MOV design-basis capability to replace stroke-time testing.	1/96-7/96 (T)
Supp 7 to GL 89-10 issued for 30-day public comment	7/95 (C)
Resolve public comments	
Issue Supp 7 in Federal Register	1/96 (C)
New Generic Letter on MOV Periodic Verification: Staff preparing generic letter to provide recommendations on the periodic verification of MOV design-basis capability.	
Issue for public comment	2/96 (C)
Final issuance	6/96 (T)
MOV Inspection Module: the staff will prepare an inspection module for inspecting MOV programs over the long-term and provide appropriate training for inspectors.	10/96 (T)
Review of EPRI MOV Performance Prediction Program: NRR and RES are currently reviewing a topical report submitted by NEI on the EPRI MOV Performance Prediction Program.	
SER	2/96 (C)
SER SUPPLEMENT	6/96 (T)

Description: Appendices A and B to 10 CFR Part 50 and 10CFR50.55(a) require nuclear power plant licensees to establish programs to ensure that structures, systems, and components important to the safe operation of the plant are designed, installed, tested, operated, and maintained in a manner that provides assurance of their ability to perform their safety functions. GL 89-10 and its supplements, asked licensees to help ensure the capability of MOVs in safety-related systems by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and looking for trends in MOV problems. EMEB has programmatic oversight responsibility of regional inspection activities conducted to verify that licensee MOV programs are being implemented. EMEB provides support to the regions, either by staff or contractor expertise, for the conduct of inspections in this area and closure of licensee actions pursuant to GL 89-10.

Historical Background: In 1985, the Davis-Besse nuclear power plant experienced a total loss of feedwater when, following a loss of main feedwater, safety-related MOVs in the auxiliary feedwater system could not be reopened after their inadvertent closure. As a result of this and other information, the NRC staff issued Bulletin 85-03 (November 15, 1985) requesting that licensees verify the design-basis capability of safety-related MOVs used in high pressure systems. The information from the implementation of Bulletin 85-03, additional operating events, and NRC-sponsored research indicated the need to expand the scope of Bulletin 85-03 to all safety-related systems.

In Generic Letter (GL) 89-10 (June 28, 1989) and its supplements, the NRC staff asked licensees to help ensure the capability of MOVs in safety-related systems by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and implementing necessary corrective action, and looking for trends in MOV problems. The NRC staff requested that licensees complete the verification of the design-basis capability of MOVs included in the scope of GL 89-10 within three refueling outages or five years from the date of issuance of the generic letter, whichever was later. The NRC staff has issued seven supplements to GL 89-10 that provide additional guidance and information on GL 89-10 program scope, design-basis reviews, switch settings, testing, periodic verification, trending, and schedule extensions.

In June 1990, the NRC staff issued NUREG-1352, "Action Plans for Motor-Operated Valves and Check Valves," describing actions to organize the activities aimed at resolving the concerns about the performance of MOVs and check valves. These actions included evaluating the current regulatory requirements and guidance for MOVs, preparing guidance for and coordinating NRC inspections, completing NRC MOV research programs and implementing the research results, and providing the nuclear industry with information on MOVs.

Proposed Actions: Specific activities included in the generic action plan to improve MOV performance are:

(1) Regulatory Improvements - The staff is working with ASME to improve the inservice testing requirements in the ASME Code and the staff is working with OM to develop guidelines for periodic verification of MOV design-basis capability to replace stroke-time testing. Recently, ASME issued Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in LWR Power Plants OM - C[illegible] 1995 Edition; Subsection ISTC." The staff is evaluating methods to endorse the code case as an alternative to MOV stroke-time testing.

(2) EPRI MOV Performance Prediction Program - On February 5, 1996, the staff forwarded to NEI the Safety Evaluation on the topical report submitted by NEI on the EPRI MOV Performance Prediction Program. On March 15, 1996, the staff issued a non-proprietary version of the SE. The staff is reviewing the hand-calculation models for two unique gate valve designs with a supplement to the SE planned for June 1996.

(3) MOV Periodic Verification Generic Letter - The staff is preparing a generic letter to provide recommendations on the periodic verification of MOV design-basis capability. On February 20, 1996, the staff published the proposed generic letter in the Federal Register for a 60-day public comment period.

(4) MOV Inspection Module - The staff plans to prepare an inspection module for inspecting MOV programs over the long-term and provide appropriate training for inspectors.

Originating Document: NRC Bulletin 85-03 issued November 15, 1985.

Regulatory Assessment: While it is important for the licensee to take steps to ensure that MOVs will operate reliably under design-basis conditions, the probability of any individual MOV failure is small and safety systems are robust enough to provide reasonable assurance of public health and safety.

Current Status: Supplement 7 to GL 89-10 was issued on January 24, 1996. Coordination with industry and support to NRC regional staff, efforts on codes and standards, and MOV research and analysis are ongoing activities. The staff is developing a generic letter which will provide guidance to licensees on periodic verification program. The staff briefed CRGR on the proposed periodic verification GL on January 31, 1996. On February 20, 1996, the proposed GL on MOV periodic verification was published in the Federal Register for a 60-day public comment period. On February 5, 1996, the staff forwarded the SE on the EPRI MOV Performance Prediction Program Topical Report to NEI. On March 15, 1996, the staff issued a non-proprietary version of the SE. The staff is reviewing the remaining EPRI models for two unique gate valve designs and plans to issue a supplement to the SE addressing these two models later in 1996. The staff has been alerting licensees, NEI and EPRI to the staff's findings from the EPRI program review, and has been communicating staff views with industry regarding periodic verification. In addition, the staff has been factoring the overall findings from the EPRI program into staff activities.

NRR Technical Contact: Thomas G. Scarbrough, EMEB, 415-2794
NRR Lead PM: Allen G. Hansen, DRPW, 415-1390

References:

Bulletin 85-03, November 15, 1985

Generic Letter 89-10, June 28, 1989, and 7 supplements

NUREG-1352, "Action Plans for Motor-Operated Valves and Check Valves," June 1990

SRP REVISION ACTION PLAN

Last Update: 3/31/96
Lead NRR Division: DISP

MILESTONES	DATE (T/C)
1. Identify recommended changes	09/94C
2. Code and standard comparisons	12/95C
3. Prepare draft revisions of current sections	10/95C
4. Develop new sections	12/95C
5. Maintenance of program data	Ongoing

Description: The Standard Review Plan (SRP) Revision Action Plan deals with the development of draft revisions for all sections in NUREG-0800 (except Chapter 7) and the development of new SRP sections to cover review areas that are supported by established staff positions or are fully addressed in the evolutionary reactor design reviews. The draft revisions will incorporate recommended changes identified in the review of generic regulatory documents and NRR staff safety evaluation reports for evolutionary light water reactor designs. The objective of the tasks outlined in the action plan is to complete the preparation of draft revisions by December 1995, with contractor assistance, while minimizing the impact on NRR technical branches.

Historical Background: The Standard Review Plan Update and Development Program (SRP-UDP) was established in 1991 to update the Standard Review Plan, NUREG-0800, (SRP) for use in reviewing future reactor design applications. The revised SRP incorporates changes in the regulation of the nuclear power industry that have occurred since the 1981 revision of the SRP. In SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," the staff discussed, in part, the revision effort for the SRP. In that paper, the staff committed to produce supplements to the 1981 SRP in parallel with the conduct of future reactor design reviews. In a memorandum of November 18, 1991, the EDO requested that the Chairman approve a commercial contract to provide technical assistance in re[illegible]ng the SRP. The Chairman provided a response dated December 13, 1991, stating his conce[illegible]t the SRP had been allowed to become "outmoded." In this regard, the Chairman stated, "The staff should ensure that when this project is completed in FY 1997, adequate agency resources and procedures are in place to review and revise the SRP as needed at least annually."

Proposed Actions: Specific tasks included in the Action Plan are: 1) Identify established staff positions and new regulatory requirements from a review of generic regulatory documents issued since the last SRP revision and from a review of NRR staff safety evaluation reports for evolutionary LWR designs; 2) Prepare a side-by-side comparison of the SRP-cited version of codes and standards vs the current version of the standard; 3) Prepare draft revisions of the current SRP sections to incorporate the changes recommended; 4) Prepare new draft SRP sections that are supported by established staff positions or are fully addressed in the evolutionary design reviews; 5) Automate the SRP to make future revisions and accessibility easier to accomplish; and 6) Maintain the program data base to reflect new staff positions and requirements.

Originating Document: Memorandum of November 18, 1991, from James M. Taylor to The Chairman, Subject, Commercial Contract for Technical Assistance to Support the Standard Review Plan Update and Development Program; and memorandum of December 13, 1991, from Ivan Selin to James M. Taylor, same subject.

Regulatory Assessment: NRR has established the SRP Update and Development Program (SRP-UDP) to update the SRP for use in the review of future reactor applications to reflect existing agency requirements and guidance and to add new review criteria to accommodate future designs.

Current Status: One contract is currently in place to support SRP-UDP activities, JCN L-2013 with Pacific Northwest Laboratory (PNL). JCN J-2055 with Idaho National Engineering Laboratory (INEL) expired on December 31, 1995. The work approach and detailed procedures have been completed for the development of SRP draft revision packages and for new SRP section development. Draft revision work for current SRP sections and new SRP sections has been completed. PNL has completed code and standard comparison work which involves the preparation of side by side comparisons between the cited version of codes and standards and the latest version, to allow SRP reviewers to use the more current version and to support SRP updates of the citations. Code and standard NUREG/CRs have been published and a notice of availability and request for comment has been placed in the Federal Register. Delivery of draft revisions to technical branches for review and concurrence is complete. Review by technical branches is being completed on a resource available basis consistent with a priority 3 effort. An automated version of the current SRP has been installed on the NRC LAN. We will solicit public comment in Spring 1996 on the manner in which existing requirements and staff positions have been reflected in the revised SRP. A memorandum of February 16, 1996, from F. Miraglia to E. Jordan informed the CRGR of our intent to issue the draft revised SRP as a "work-in-progress" to solicit public comment without prior CRGR review. Delivery of the camera-ready document to the Publications Branch is scheduled for mid-April 1996.

NRR Technical Contact: A. Masciantonio, PIPB, 415-1290

References:

SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions"

Memorandum of November 18, 1991, from James M. Taylor to The Chairman, Subject, Commercial Contract for Technical Assistance to Support the Standard Review Plan Update and Development Program

Memorandum of December 13, 1991, from Ivan Selin to James M. Taylor, same subject

Memorandum of May 17, 1994, from Frank P. Gillespie to W[illegible]iam T. Russell, Subject, Action Plan for the Development of Draft SRP Revisions in the SRP-U[illegible] Available in Central Files)

UPDATE OF SRP CHAPTER 7 TO INCORPORATE DIGITAL INSTRUMENTATION AND CONTROLS (I&C) GUIDANCE

Last Update: 4/1/96
Lead NRR Division: DRCH

MILESTONES	DATE (T/C)
1. Develop Update of SRP Chapter 7	10/96T
2. ACRS Subcommittee Briefings	3/96T, 5/96T, 7/96T 10/96T
3. Incorporate new Regulatory Guides (provided by RES) in SRP Chapter 7 Update	8/96T
4. Incorporate results from National Academy of Sciences study	10/96T
5. Draft SRP to Chairman	9/30/96T
6. Publish Draft SRP Chapter 7 for Public Comment	12/96T
7. Incorporate Public Comments	3/97T
8. Final ACRS/CRGR Review of SRP Chapter 7	4/97T
9. Final SRP to Chairman	3/31/97T
10. Publish Final SRP Chapter 7	5/97T

Description: This task action plan is used to track and manage the final phase of codifying the digital I&C regulatory approach and criteria by updating the existing Standard Review Plan (SRP) Chapter 7.

Historical Background: By a staff requirements memorandum (SRM) dated November 30, 1995, from the Chairman, Shirley Ann Jackson, to the Executi[illegible] [illegible]ector of Operations, James M. Taylor, the Chairman requested that the staff develop an action [illegible] [illegible]n the area of digital instrumentation and controls. The action plan is for the expeditious development of a Standard Review Plan (SRP) to ensure that safety margins are addressed and that NRC regulatory requirements are available and ready for use when reviewing licensee proposed installation of digital instrumentation and control systems in nuclear power plants. The staff has an ongoing effort for updating Chapter 7 of the SRP that deals with instrumentation and control systems to accomplish the requested action and this task action plan was initiated to track and manage the final phase of that effort in response to the SRM.

Proposed Actions: Specific actions included in this task action plan are: (1) to develop the update of SRP Chapter 7, (2) to periodically brief the ACRS as sections of the SRP update are completed, (3) to incorporate new regulatory guides on digital I&C that will be provided by the Office of Nuclear Regulatory Research (RES), (4) to incorporate results from the National Academy of Sciences study of digital I&C at nuclear plants, (5) to publish the draft SRP Chapter 7 for public comments, (6) to incorporate the public comments, (7) to have final ACRS and CRGR review of the SRP Chapter 7 update, and (8) to publish the final revised SRP Chapter 7.

Originating Document: The memorandum from the EDO to Chairman Jackson dated January 3, 1996, "Improvements Associated with Managing the Utilization of Probabilistic Risk assessment (PRA) and Digital Instrumentation and Control Technology."

Regulatory Assessment: The approach and criteria that form the current regulatory framework for review and acceptance of digital I&C systems in nuclear power plants is being codified in the update to SRP Chapter 7. This framework has been communicated to the industry and public in safety evaluations for digital modifications to operating plants and design certification of the advanced reactor designs, and in Generic Letter 95-02, "Use of NUMARC/EPRI Report TR-102348, 'Guideline on Licensing Digital Upgrades,' in Determining the Acceptability of Performing Analog-to-Digital Replacements Under 10 CFR 50.59 dated" dated April 26, 1995. This action plan tracks and manages the codification of the existing framework by updating SRP Chapter 7. Consequently, this is not an urgent regulatory action, and continued plant operation is justified.

Current Status: The staff and its contractor, Lawrence Livermore National Laboratories (LLNL), are currently revising the seven existing sections of SRP Chapter 7 and developing two new sections and several new branch technical positions (BTPs) to incorporate criteria and guidance related to digital I&C systems. In parallel, the Office of Nuclear Regulatory Research (RES) is developing several regulatory guides that endorse national standards related to digital I&C.

Final staff comments for three draft documents related to SRP Chapter 7 update, SRP Sections 7.0 and 7.1 on Introduction to Chapter 7, General Review Criteria respectively and draft Branch Technical Position (BTP HICB-14) on Guidance for Software Reviews are being incorporated. These draft documents were transmitted to the ACRS on February 20, 1996. These documents were discussed with the ACRS Control Systems Subcommittee on March 6, 1996 and the full committee on March 7, 1996. The remaining Sections of SRP Chapter 7 and new BTPs are under development and will be transmitted to ACRS as they are available.

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NEW SOURCE TERM FOR OPERATING REACTORS

Last Update: 3/28/96
Lead NRR Division: DRPM
Supporting Division: DSSA

MILESTONES	DATE (T/C)
1. NEI Letter	07/94C
2. Commission Memo	09/94C
3. NEI Response	09/94C
4. NEI/NRC Meeting	10/94C
5. Publication of NUREG-1465	02/95C
6. NEI/NRC Meetings	06/95C 10/95C 01/96C
7. Submittal of Generic Framework Document (from NEI)	11/95C
8. First Pilot Plant Submittal (Brown's Ferry)	12/95C
9. Draft Commission Paper	04/96T
10. Pilot Plant Submittals	12/96T

Description: More than a decade of research has led to an enhanced understanding of the timing, magnitude and chemical form of fission product releases following nuclear accidents. The results of this work has been summarized in NUREG-1465 and in a number of related research reports. Application of this new knowledge to operating reactors could result in cost savings without sacrificing real safety margin. In addition, safety enhancements may also be achieved.

Historical Background: In 1962, the U. S. Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors." Since then licensees and the NRC have used the accident source term presented in TID-14844 in the evaluation of the dose consequences of design basis accidents (DBA).

After examining years of additional research and operating reactor experience, NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," in February 1995. The NUREG describes the accident source term as a series of five release phases. The first three phases (coolant, gap, and early in-vessel) are applicable to DBA evaluations, and all five phases are applicable to severe accident evaluations. The DBA source term from the NUREG is comparable to the TID source term; however, it includes a more realistic description of release timing and composition. Since the NUREG source term results in lower calculated DBA dose consequences, NRC decided not to require current plants to revise their DBA analyses using the new source term. However, many licensees want to use the new source term to perform DBA dose evaluations in support of plant, technical specification, and procedure modifications.

NRC and NEI met several times to discuss the industry's plans to use the new source term. To make efficient use of NRC's review resources, NRC encouraged the industry to approach the issue on a generic basis. The Nuclear Energy Institute (NEI) unveiled its plans for the use of the new source term at operating plants at the Regulatory Information Conference in May 1995. NEI, Polestar (EPRI's consultant), and pilot plant (Grand Gulf, Millstone, Beaver Valley, Browns Ferry, Perry, and Indian Point) representatives met with NRC staff on June 1 and October 12, 1995, to discuss more detailed plans.

Proposed Actions: The staff plans to review the framework document and draft a Commission paper in April that describes a generic implementation approach. The staff would review each pilot plant application and prepare a generic letter under a line-item improvement process addressing the use of each feature of the NUREG-1465 source term. Subsequent applications by utilities, employing the methodology and addressing the issues identified in the generic letter, would be approved without the need for additional, lengthy, detailed review. The staff anticipates that review of these applications will require less resources than the pilot plant reviews.

Originating Document: EPRI Technical Report TR-105909, "Generic Framework Document for Application of Revised Accident Source Term to Operating Plants," transmitted by letter dated November 15, 1995.

Regulatory Assessment: There will be no mandatory backfit of the new source term for operating reactors. The design-basis accident analyses for current reactors based on the TID-14844 source term are still valid. Therefore, non-urgent regulatory action and continued facility operation are justified.

Current Status: NEI submitted its generic framework document in November 1995 for NRC review and approval, and TVA submitted its pilot plant application for Brown's Ferry in December 1995. The staff met with NEI on January 23, 1996 to discuss its proposed actions. A meeting was held on February 7, 1996, in order to have Brown's Ferry discuss its pilot plant submittal and demonstrate how a utility would use the framework document to implement the new source term. The staff intends to complete its review of the framework document before the end of April 1996 and to issue a Commission paper describing how it intends to conduct its generic review of pilot plant submittals. Remaining pilot plant submittals are expected before the end of 1996.

NRR Technical Contact:	R. Emch, PERB, 415-1068
	A. Huffert, PERB, 415-1081
NRR Lead PM:	J. H. Wilson, PDST, 415-1108

References:
NUREG-1465, "Accident Source Term for Light Water Nuclear Power Plants," February, 1995.

July 27, 1994, letter to A. Marion, NEI, from D. Crutchfield, NRC, "Application of New Source Term to Operating Reactors".

September 6, 1994, letter to the Commission from NRC staff, "Use of NUREG-1465 Source Term at Operating Reactors".

Summaries of public meetings:
- dated November 10, 1994 for public meeting with NEI held on October 6, 1994;
- dated July 26, 1994 for public meeting with NEI held on June 1, 1995;
- dated November 17, 1995 for public meeting with NEI held on October 12, 1995.
- dated February 1, 1996 for public meeting with NEI held on January 23, 1996.

July 21, 1995, letter to the Commission from NRC staff, "Use of NUREG-1465 Source Term at Operating Reactors".

December 22, 1995, pilot plant submittal, letter to Document Control Desk from Tennessee Valley Authority, "Brown's Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specifications (TS) No. 356 and Cost Beneficial Licensing Action (CBLA) 08 - Increase in Allowable Main Steam Isolation Valve (MSIV) Leakage Rate and Request for Exemption from 10 CFR 50, Appendix J... and 10 CFR 100, Appendix A...".

ENDANGERED SPECIES ACTION PLAN

Last Update: 4/1/96
Lead NRR Division: DRPM

	MILESTONE	DATE
1.	Development of action plan.	06/95C
2.	Develop list of currently listed protected species in the vicinity of each nuclear power plant site	11/95C
3.	Identify individual licensee programs and activities being conducted to further the conservation of protected species.	04/96T
4.	Determine priority for sites warranting follow-up actions.	06/96T
5.	Completion of site-specific follow-up actions.	11/96T
6.	Development and implementation of process for maintaining status and compliance with the ESA at each site.	03/97T

Description: Develop a list of currently listed protected species in the vicinity of each nuclear power plant site, identify individual licensee programs and activities being conducted to further the conservation of protected species, and conduct, as necessary, informal or formal consultation with either the National Marine Fisheries Service or the Fish and Wildlife Service is warranted for any specific site. specific

Historical Background: In 1973, Congress passed the Endangered Species Act for the protection of endangered or threatened species. In responding to a Commission memorandum of July 30, 1991, concerning efforts of the Commission, applicants, and licensees for protection of endangered species in the vicinity of nuclear power facilities, it was identified that the NRC may not have completed all the necessary activities required by the Endangered Species Act for some of the facilities that have identified endangered species. This action plan will determine the additional actions, if any, that need to be taken at individual sites so that the NRC can meet its obligations under the act.

Proposed Actions: Conduct evaluations of plant-specific lists of endangered species and existing licensee commitments to further the conservation of the protected species and determine if informal or formal consultation with either the National Marine Fisheries Service or the Fish and Wildlife Service is warranted.

Originating Document: Commission Memorandum of July 30, 1991

Regulatory Assessment: Continued facility operation is appropriate because this action plan does not involve a health and safety issue.

Current Status: A list of currently listed protected species in the vicinity of each nuclear power plant site was developed. Identification of licensee programs and activities is continuing. The staff has sent out letters to the regional offices of the Fish and Wildlife Service (FWS) and of the National Marine Fisheries Service (NMFS). Responses have been received from some but not all of the FWS and NMFS regional offices. This is causing a delay in the completion of milestones 3 and 4.

Contacts: NRR Technical Contacts: Mike Masnik, ONDD, 415-1191,
Jim Wilson, PDST, 415-1108
NRR Lead PM: Steve Reynolds, PDLR, 415-1115

References: Commission Memorandum of July 30, 1991

EFFECT OF HURRICANE ANDREW ON TURKEY POINT

Last Update: 3/28/96
Lead NRR Division: DRPM
Supporting Division: DISP

MILESTONES	DATE (T/C)
1. Evaluate the Adequacy of Licensee Offsite Communications for Natural Disasters Within the Plant Design Basis.	1/97T
Collect information on licensee communication capabilities and vulnerabilities via region inspection.	7/96T
Analyze inspection findings and report on results.	10/96T
Established schedule for issuance of generic correspondence (if necessary).	11/96T
2. Evaluate the Adequacy of NRC Guidance for Reviewing Licensee Preparation and Response to Natural Disasters and Industry Preplanned Support.	12/96T
The action will provide guidance for inspectors to address any vulnerabilities that may develop from the review of Individual Plant Examination of External Events (IPEEE) . Completion of this action is currently scheduled for 1996. PIPB	

Description: This action plan was developed to address the actions necessary to resolve the issues identified in the "Report on the Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20-30, 1992." Two of the issues are still being considered. They are:

1) Whether there is a need for generic guidance to licensees to ensure that their offsite communication circuits can reliably survive or recover from the impact of a severe natural event such as a hurricane. These circuits are required to provide reliable notification to offsite authorities of emergency conditions at the licensee's power reactor facility.

2) Whether there is a need for generic guidance to inspectors to review licensees' preparation for and response to natural disasters, including industry preplanned support.

Historical Background: On August 24, 1992, Category 4 Hurricane Andrew hit south Florida and caused extensive onsite and offsite damage at Turkey Point. An NRC/industry team was organized to review the damage that the hurricane caused the nuclear units and the utility actions to prepare for the storm and recover from it, and to compile lessons that might benefit other nuclear reactor facilities. Results of the team review are presented in the report, "Report on the Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station From August 20-30, 1992," issued in March 1993. This report was distributed to all power reactor licensees by the Institute of Nuclear Power Operations on June 10, 1993.

The EDO requested a review of the NRC/industry report to determine the actions necessary for resolving the issues identified in the report. An action plan was established on July 22, 1993, to perform this function. Annual written status reports are provided until all items are closed. The October, 1995 report contained two open items, listed above.

Proposed Actions: For item 1) above, a Temporary Instruction has been developed for inspectors to review licensee offsite communication circuits during emergency preparedness inspections scheduled at power reactor facilities between February and July of this year. Data collected from those inspections, as well as past inspections, will be evaluated to determine if guidance to licensees, in the form of generic communication, is necessary to provide either survivability or rapid recoverability of these circuits from a severe natural event.

Regulatory Assessment: Justification for non-urgent regulatory action: A qualitative safety assessment of the technical issues being addressed for item 1) demonstrates that the significance of the issue is at a level that will allow both continued facility operation and treatment of the issue as a non-urgent regulatory action.

Current Status: For item 1) a temporary instruction (TI 2515/131), issued 1/18/96, incorporating Regional comments, has been written to provide Regional inspectors guidance for collecting information on offsite notification circuits. The TI has been performed at two plants since February 1, 1996..

For item 2) the Inspection Program Branch (PIPB) has concluded that from an emergency preparedness standpoint, sufficient guidance exists for reviewing licensee preparations in response to a hurricane or other external events. The staff issued IN 93-53, Supplement 1, on April 29, 1994, in which the staff expanded the scope of lessons learned to other external events and discussed existing regulatory guidance for various external events. The action to provide guidance for inspectors to address any vulnerabilities that may develop from the review of individual plant examination of externally initiated events (IPEEE) (GL 88-20, Supplement 4) has been incorporated into the Probabilistic Risk Assessment Implementation (Activity 1.3 (b)). Completion of this action is currently scheduled for February 1997. On that basis, Milestone 2 is considered closed as the status will now be reported under the Probabilistic Risk Assessment Implementation Plan.

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NRR Lead PM:	R. Croteau, DRPE, 415-14[illegible]

ESRP REVISION ACTION PLAN

Last Update: 03/25/96
Lead NRR Division: DRPM

MILESTONES		DATE (T/C)
1.	Reflect Potential Impacts and Integrated Impacts in Options for Resolution	08/96T
2.	Prepare Final Draft of ESRP Sections for Public Comment	08/97T
3.	Disposition Public Comments	01/98T
4.	Publish Final NUREG-1555	08/98T
5.	Maintenance of program data	Ongoing

Brief Description: The Environmental Standard Review Plan (ESRP) Revision Action Plan deals with the revision to NUREG-0555 to reflect changes in the statutory and regulatory arena and to incorporate emerging environmental protection issues (e.g., SAMDA and environmental justice) since originally published in 1979. The ESRP will take the form of the SRP (including acceptance criteria) and follows the same update criteria outlined under the SRP-UDP project (with the exception of maintaining the MDB at this time). The objective of the tasks outlined in the action plan is to complete the identification of potential impacts by April 1996, the integrated impacts by June 1996, and the options for resolution by August 1996. After submittal of the draft by February 1996 for staff and CRGR review, if necessary, the sections will be published for public comment in August 1997. Disposition of public comments and staff review of the update (NUREG-1555) leads to a publication date of August 1998.

Regulatory Assessment: NRR has established the ESRP Update Program for use in the review of future reactor site approval applications, to fill the voids for operating reactors and license renewal applications, to reflect current NRC requirements and guidance, and to consider other statutory and regulatory requirements (e.g., the National Environmental Policy Act, Presidential Executive Orders).

Current Status: Two contracts are currently in place to support the ESRP Program, JCN J-2028 with Pacific Northwest National Laboratory (PNL) for overall coordination and most of the ESRP sections and JCN J-2039 with Lawrence Livermore National Laboratory (LLNL) for the seismology and geology sections. The work approach and detailed procedures rely heavily on the framework established for the SRP-UDP. The project team was established in 1994; resources were diverted twice to work off higher priority activities (i.e, the Watts Bar Environmental Statement Update and the RADTRAD project). Work is expected to resume in the second quarter FY-96.

NRR Technical Contact: B. Zalcman, PDST, 415-3467

10 CFR 50.59 ACTION PLAN

Last Update: 4/15/96
Lead NRR Division: DRPM
Supporting Divisions: all

MILESTONES	DATE (T/C)
1. Action plan approval/copy to Commission	04/15/96(C)
2. Identify work group members	04/25/96(T)
3. Brief D/NRR on issues	06/01/96(T)
4. Conduct workshop	06/14/96(T)
5. Brief D/NRR on proposed positions	07/15/96(T)
6. Draft position papers	08/15/96(T)
7. Obtain regional comments	09/15/96(T)
8. Obtain public comments	12/96(T)
9. ACRS Review	12/96(T)
10. Issue inspection guidance/Assess results /prepare recommendations	01/97(T)
11. Commission Paper	02/97(T)
12. Followon Actions	TBD

Description: This action plan defines measures to improve licensee implementation and NRC staff oversight of the 10 CFR 50.59 process.

Historical Background: 10 CFR 50.59 was promulgated 62 to describe the circumstances under which licensees may make changes to their facility (or to make changes to procedures, or to conduct tests and experiments) without prior NRC approval when the change does not involve the Technical Specifications or an unreviewed safety question. Licensees are required to submit periodically information related to changes made pursuant to 50.59. The NRC has programs for monitoring licensee processes for implementing 50.59. In a memorandum dated October 27, 1995, Chairman Jackson raised a number of questions concerning 50.59 implementation and NRC oversight, and proposed a systematic reconsideration and reevaluation of the process.

The December 15, 1995, memorandum from the EDO responded to the specific questions and stated that within 120 days from the date of the memorandum, the staff would review previously issued guidance on implementation of the 50.59 process to define areas where the guidance needs to be amended and to develop an action plan to identify actions to be undertaken to improve both the licensee's implementation and the NRC staff's oversight of the 50.59. The staff has completed its review of existing guidance and has identified certain issues for further examination, which this action plan addresses.

The staff plans to make the results of its review of guidance, the action plan, and its interim inspection guidance publicly available.

Planned Actions:

The staff's approach to development of regulatory guidance would proceed in phases. Over the next several months, the staff will attempt to provide specific positions (guidance) to accomplish the objectives listed below, and will evaluate the feasibility of implementing such guidance within the existing regulatory framework. At the end of the first phase, estimated to take six to eight months, the staff would take stock of its progress and make recommendations on issuing guidance, undertaking rulemaking or other actions.

Specifically, the objectives of this effort are to develop guidance that would:

- o define the elements of safety evaluation review or screening processes within the context of various licensee design or change control processes, to provide greater assurance that effects on safety of changes, whether to equipment, procedures, or methods of system operation, are appropriately evaluated.
- o define more specifically the scope of applicability of 50.59 (that is, to identify those changes, tests, or experiments) that need to be evaluated to determine if NRC approval is needed). This would include a more comprehensive description of change, and guidance for broader consideration of "as described."
- o establish the process for resolving nonconforming conditions such that differences from the FSAR are reconciled (from both safety and regulatory viewpoints) in a time frame commensurate with their safety significance.
- o improve USQ determinations in the following respects:
  - address the extent to which short and long term compensating actions may be considered as part of change under 50.59 so that it can be determined that the probability has not increased or margins of safety as defined in the basis for any technical specification has not been reduced. Also address when consideration of compensating actions should be reviewed as part of the basis for approving a proposed license amendment.
  - clarify the extent to which PRA techniques may be useful in evaluating the effects on safety of a change, and in addressing the "probability may be increased" criterion for unreviewed safety questions.
  - clarify what is meant by "margin of safety" in relation to numerical parameters, analysis methods, calculated results of safety analyses, and licensing limits such that changes that might affect the basis for staff's safety conclusions with respect to Technical Specifications are more consistently identified.

Public comments on the position paper(s) will be obtained. The ACRS will be requested to provide its comments on these positions. Actions, milestones and schedules for further phases of this effort will be developed after the results of the first phase are assessed.

In the area of staff oversight, the staff plans to conduct a roundtable discussion with regional staff, resident inspectors and NRR staff who have participated in 50.59 inspection efforts to share experiences and to discuss such topics as the mix of programmatic and implementation reviews, sampling and team composition. Appropriate changes to inspection procedures will be made.

Other related efforts are being tracked under other programs.

Originating Document: December 15, 1995 memorandum from the EDO to Chairman Jackson, Subject: Response to Questions on Facility Changes Pursuant to 10 CFR 50.59

Regulatory Assessment: The action plan was developed to identify actions to improve implementation of the 50.59 process. A number of improvements have been implemented in the last few months, such as directing inspectors conducting all routine inspections to specifically address FSAR compliance, and reviewing spent fuel pool/core offload procedures and practices at all facilities. As stated in the December 15, 1995, memorandum, "The staff concludes that there is currently no indication that implementation of 10 CFR 50.59, as it is carried out today, has led to decreased safety, based on inspection experience. While improvements can be made to achieve a higher degree of uniformity of review, the current process as it is being implemented provides reasonable assurance that plant safety has not been decreased." The above conclusion is confirmed by the additional analysis of inspection experience presented in the staff review document. Therefore, non-urgent regulatory action and continued facility operation are justified.

Current Status: The action plan was issued on April 15, 1996.

NRR Technical Contact: E. McKenna, PECB, 415-2189

References: October 27, 1995 memorandum from Chairman Jackson to EDO
November 30, 1995 memorandum from Chairman Jackson to EDO
December 15, 1995 memorandum from EDO to Chairman Jackson
December 28, 1995 memorandum from EDO to Chairman Jackson

GENERAL ELECTRIC EXTENDED POWER UPRATE ACTION PLAN (A STRATEGY FOR COMPLETION OF BOTH THE GENERIC AND PLANT SPECIFIC REVIEWS FOR EXTENDED POWER UPRATE SUBMITTALS FOR BWRs)

Last Update: 03/31/96
Lead NRR Division: DRPW
Supporting Division: DSSA

MILESTONES	DATE (T/C)
Milestone 1: GE Topical ELTR1 submitted.	3/95 C
Milestone 2: Issue Staff Position Paper on ELTR1	
Actions:	
- Meeting with GE/NSP.	4/95 C
- Identify differences between LTR1 and ELTR1.	8/95 C
- Issue RAIs as appropriate.	9/95 C
- Incorporate information on foreign experience obtained from SRXB.	10/95 C
- Develop power uprate database for all U.S. plants.	10/95 C
- Issue Staff Position Paper.	2/96 C
-	
Milestone 3: Receive ELTR2. (GE plans to submit ELTR2 in two parts: the first part in March 1995 and the second part in June 1996.)	
Actions:	
- Open TAC No. and issue work orders to technical branches to review ELTR2.	3/96 C
Milestone 4: Issue SE on GE ELTR2.	
Actions:	
- Meeting with GE/Industry.	2/96 C
- Issue RAIs as appropriate.	8/96T (1st set) 10/96T (2nd set)
- Input to the SE from technical branches.	2/97 T
- Issue SE.	4/97 T
- ACRS presentation.	4/97 T
Milestone 5: Receive Lead Plant Application.	6/96 T
Actions:	
- Issue Secy Information paper.	7/96 T
Milestone 6: Issue SE for Lead Plant.	
Actions:	
- Meeting with Monticello.	6/96 T
- RAIs input from tech branches.	11/96 T
- Issue RAIs as appropriate.	11/96 T
- Input to the SE from tech branches.	6/97 T
- Issue SE.	6/97 T

MILESTONES		DATE (T/C)
Milestone 7:	Develop a Standard Review Procedure. Incorporate lessons learned from Lead Plant activity.	6/97 T

Description: This action plan describes the strategy for completing both the generic and plant-specific reviews for extended power uprate submittals for boiling water reactors (BWRs). General Electric Company (GE) submitted a licensing topical report (ELTR1), which outlines the methodology for implementation of an extended power uprate program. ELTR1 encompasses power uprates of up to 120 percent of the original licensed thermal power. Individual plant submittals for uprates will likely contain requests for an optimum power level specific for that plant which is something less than the full 120 percent.

The technical branches will review the applicable portions of the ELTR2, GE topical report containing generic analyses and the lead plant application, and provide input into both safety evaluation reports. Review criteria from the reviews performed on ELTR1, generic analyses, and the lead plant submittal will be developed and assembled into a review procedure for individual PMs to use for subsequent plant-specific reviews. If an area in an individual plant submittal is outside the bounds of the previously established criteria, the applicable technical branch will perform a review of that specific area and provide input into the safety evaluation.

Historical Background: The generic BWR power uprate program was created to provide a consistent means for individual licensees to recover additional generating capacity beyond their current licensed limit. In 1990, GE submitted licensing topical reports to initiate this program by proposing to increase the rated thermal power levels of the BWR/4, BWR/5, and BWR/6 product lines by approximately 5 percent. Since 1990, the staff has reviewed and approved at least 9 such power uprate requests under this generic BWR power uprate program. As a follow-on to this program, GE submitted ELTR1 in March 1995 to propose "extended" power uprates of up to 120 percent of the original licensed thermal power.

Proposed Actions: Specific actions included in the generic action plan are: (1) review ELTR1 and issue a staff position paper, (2) review ELTR2 and issue a [illegible]y evaluation report, (3) review the lead plant application and issue a safety evaluation report, (4) develop a standard review procedure based on ELTR1, ELTR2, and the lead plant review.

Originating Document: GE Licensing Topical Report (NEDC-32424), "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," dated February 1995.

Regulatory Assessment: Not applicable. (A safety assessment is not needed for this action plan because a justification for continued operation of a plant is not required.) This program is an industry initiative that is strictly voluntary.

Current Status: The staff position paper on the BWR Extended Power Uprate Program was issued on February 8, 1996 (completion of Milestone 2). The staff also met with GE and the interested BWR utilities on February 14, 1996 to discuss the contents of the staff position paper and the overall status of the power uprate program. On March 26, 1996, GE submitted ELTR2, the generic bounding analyses supporting the extended power uprate program. The lead plant application from Monticello is now expected in June 1996.

NRR Lead PM: T. J. Kim, DRPW, 415-1392

DRY CASK STORAGE ACTION PLAN

Last Update: 03/29/96
Lead NRR Division: DRPW

MILESTONES	DATE (T/C)
1. Develop action plan	07/95C
2. Near-term technical issues	
a. Heavy Loads/Cranes	
- develop working group plan	11/95C
- complete actions	12/96T
b. Cask Trunnions[1]	
- develop staff position	09/95C
- modify standards/guidance	No changes required (C)
c. Hydrostatic Testing[1]	12/95C
d. Seismic Requirements for Pads	
- issue Information Notice	06/95C
3. Long-term technical issues	
a. Cask weeping[1]	
- meet with NEI	08/95C
- determine NRC actions to resolve	As Necessary
b. Cask loading/unloading procedures	
- contact NEI about industry efforts	08/95C
- resolve high priority issues	09/95C
- form working group	10/95C
- complete working group determination on further issues	04/96T
c. Off Loading after fuel pool is decommissioned[1]	
- develop guidance and modifications to inspection procedures	As required in response to submittals
d. Failed Fuel Storage[1]	
- review proposed solutions	Reviewing first submittal, ECD 06/96T
e. Safeguards Concerns[1]	
- complete analysis of designs	12/95C

[1] NMSS has the lead for this issue.

MILESTONES	DATE (T/C)
4. Procedural issues	
a. Change processes	
- issue SRP and 50.59 guidance	03/96C
- training for staff	06/96T
b. Reporting Requirements[1]	
- develop position, communicate to licensees	09/95C
c. Inspection of site activities	
- issue revised procedures	02/96C
- develop resource estimates and inspection schedule	02/96C
d. Vendor Inspections[1]	
- issue revised procedures	02/96C
- develop resource estimates and inspection schedule	10/95C
e. Cask and SAR differences[1]	
- contact vendors	09/95C
5. Communications	
a. Interface meetings	Ongoing
b. Staff training[1]	10/95C
c. Industry workshop[2]	07/95C

Description: The Plan was developed to identify and resolve major issues and problems in the area of dry cask storage of spent reactor fuel in independent spent fuel storage installations (ISFSIs). Specific issues encompassed by the plan include heavy load control, procedures for cask loading and unloading, failed fuel storage, change processes, inspection activities, and communications (internal and external). Issues have been divided into the following categories: near-term technical, long-term technical, communications, and process issues.

Historical Background: Since 1986, several U.S. nuclear power plant licensees have installed independent spent fuel storage installations (ISFSIs), that is, licensee-owned dry cask storage facilities. Other licensees are also planning such installations In recent years, licensees have encountered a number of problems during the fabrication, ' 'lation and licensing of some of these ISFSIs and there has been an inconsistent level of p. mance by involved licensees and cask fabricators with respect to the use of dry cask storage of spent reactor fuel. Because of the anticipated increased industry effort in this area, the staff needed to fully understand the problems that occurred and take appropriate measures to reduce such problems in the future. Therefore, NMSS and NRR reviewed the lessons learned from past experience with ISFSIs, both our experience and the experience of other headquarters and regional offices, and developed a plan to resolve major issues and problems.

Proposed Actions: Actions included in the plan are: (1) review each general issue and identify the specific problems to be addressed, (2) develop corrective actions for each problem, and (3) implement the corrective actions.

Originating Document: Memorandum from Carl J. Paperiello and William T. Russell to James M. Taylor, July 28, 1995, "Dry Cask Storage Action Plan".

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[2] An additional workshop has been tentatively scheduled for May 1996.

BWR SUCTION STRAINER CLOGGING ISSUE

Last Update: 4/1/96
Lead NRR Division: DSSA

MILESTONES		DATE (T/C)
1.	Barsebäck Event	07/92C
2.	BWROG Survey Results	10/92C
3.	Perry Event	03/93
4.	IN 93-34 Supp 1	05/93C
5.	Bulletin 93-02	05/93C
6.	Preliminary Scientific Engineering Associates (SEA) Study	01/94C
7.	OECD/NEA Workshop	01/94C
8.	NRCB 93-02 Supplement 1	02/94C
9.	Response to NRCB 93-02 Supp 1	04/94C
10.	User Need letter to RES for filtering experiments	05/94C
11.	Review of NRCB 93-02 Supp 1 complete	08/94C
12.	Alden Laboratories starts preparing experimental program	08/94C
13.	SEA report out for public comment	08/94C
14.	Draft Consensus of CSNI Working Group	04/95C
15.	Public comment period ends for SEA report. Input from BWROG on proposed resolution.	11/94C
16.	Alden commences experimental program	10/94C
17.	Final SEA report issued	12/95C
18.	Final test report from Alden	09/95C
19.	Establish technical position in Draft Bulletin and Draft Reg. Guide 1.82, Rev. 2.	03/95C
20.	CRGR Brief on Draft Bulletin	06/95C
21.	Draft Bulletin on resolution of issue out for public comment	07/95C
22.	Issue Urgent Bulletin 95-02 on Limerick Event	10/95C
23.	Complete preliminary review of Licensee responses to Bulletin 95-02/Complete resolution of public comments on draft bulletin. Incorporate appropriate changes into final Bulletin.	3/96C
24.	Brief CRGR	3/96C
25.	Brief ACRS	2/96C
26.	Issue final Bulletin and Reg. Guide	4/96T

Regulatory Assessment: The plan addresses dry storage of fuel that is several years old. Technical issues have been addressed on a site-specific basis for existing facilities. The action plan will improve guidance, enhance communications with industry and the public, and aid future applicants.

Current Status: The following action plan issues have been completed: cask trunnions, cask weeping, hydrostatic testing, safeguards concerns, Part 72 reporting requirements, inspection of site activities, and vendor inspections. The inspection procedures for dry cask activities (site and vendor) were issued in February, 1996. These procedures included resource estimates for inspection activities. The balance of the technical issues are on schedule. The draft SRP has been issued for comment. Public comments are due by June 18, 1996. The staff has not identified any cask-specific 50.59 issues that require further clarification. A related 50.59 issue involving heavy load control in general will be the subject of a bulletin that the staff has drafted. The staff is currently resolving CRGR comments on the draft bulletin. The staff is evaluating a request from the Civil Engineering and Geosciencas Branch to incorporate additional guidance on seismic issues into Inspection Procedure 60851. All of the communications issues are ongoing efforts with no specific criteria for closure. However, there have been significant improvements in these areas. The Regions, NMSS, and NRR hold regular interface calls to discuss dry cask issues, training has been given to many of the affected staff, and NRC has established open communications with the newly-formed Nuclear Energy Institute Dry Cask Storage Issue Task Force. Based on these improvements, the staff will review these issues for closure in the coming months.

NRR Contact: Andrew Kugler, DRPW, 415-2828
NMSS Contact: Patricia Eng, SFPO, 415-8577

References:

Memorandum from Robert M. Bernero and William T. Russell to James M. Taylor, March 15, 1995, "Realignment of Reactor Decommissioning Program"

Memorandum from Carl J. Paperiello and William T. Russell to James M. Taylor, July 28, 1995, "Dry Cask Storage Action Plan"

Memorandum from Carl J. Paperiello and William T. Russell to James M. Taylor, January 25, 1996, "Update to the Dry Cask Storage Action Plan"

Description: Two operating reactor events have led to the re-examination of the issue of the potential for blockage of BWR ECCS strainers by debris generated during a LOCA.

Historical Background: On July 28, 1992, an event occurred at Barsebäck Unit 2, a Swedish BWR, which involved the plugging of two ECCS suction strainers. The strainers were plugged by mineral wool insulation that had been dislodged by steam from a pilot-operated relief valve that spuriously opened while the reactor was at 3,100 kPa [435 psig]. Two of the five strainers on the suction side of the containment spray pumps were in service and became partially plugged with mineral wool. Following an indication of high differential pressure across both suction strainers 70 minutes into the event, the operators shut down the containment spray pumps and backflushed the strainers. The Barsebäck event demonstrated that the potential exists for a pipe break to generate insulation debris and transport a sufficient amount of the debris to the suppression pool to clog the ECCS strainers. Following this event, the staff issued NRC Information Notice 92-71 informing U.S. licensees of this event.

On January 16 and April 14, 1993, two events involving the clogging of ECCS strainers also occurred at the Perry Nuclear Power Plant, a domestic BWR. The first Perry event involved clogging of the suction strainers for the residual heat removal (RHR) pumps by debris in the suppression pool. The second Perry event involved the deposition of filter fibers on these strainers. The debris consisted of glass fibers from temporary drywell cooling unit filters that had been inadvertently dropped into the suppression pool, and corrosion products that had been filtered from the pool by the glass fibers which accumulated on the surface of the strainer. The Perry events demonstrated the deleterious effects on strainer pressure drop caused by the filtering of suppression pool particulates (corrosion products or "sludge") by fibrous glass materials entrained on the ECCS strainer surfaces. Following these two events, the staff issued NRC Information notice 93-34 and its supplement, and NRC Bulletin 93-02, which requested licensees to remove all temporary sources of fibrous material from their containments.

The staff then performed calculations to assess the vulnerability of each domestic BWR. The results of these calculations showed that the potential existed for the ECCS pumps to lose net positive suction head (NPSH) margin due to clogging of the suction strainers by LOCA-generated debris. This led the staff to conduct a detailed study of a reference BWR 4 plant with a Mark I containment. The results of the staff study are contained in NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA [illegible]enerated Debris," which was published in November 1995. The study results reaffirm[illegible] results of the earlier staff calculations.

Members of the NRC staff also attended an Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) workshop on the Barsebäck incident held in Stockholm, Sweden, on January 26 and 27, 1994. Representatives from other countries at this conference discussed actions taken or planned which would prevent or mitigate the consequences of BWR strainer blockage. Based on the preliminary results of the staff's study, described above, as reinforced by information learned at the OECD/NEA workshop, the staff issued NRC Bulletin 93-02, Supplement 1, which requested licensees to implement interim measures to ensure ECCS reliability until a generic resolution for this issue could be achieved. In addition, an action plan for this issue was developed for taking generic action to ensure that the ECCS in all BWRs are capable of performing their safety functions.

Proposed Actions: Specific actions included in the generic action plan are: (1) issuance of NRC bulletins 93-02 and its supplement to request licensees to take appropriate interim actions to ensure reliability of the ECCS so that the staff and industry have sufficient time to develop a permanent resolution, and (2) to develop for issuance a final bulletin which will request licensees to implement appropriate programs and hardware modifications to ensure that their ECCS can perform its safety function.

Originating Documents: NRC Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," dated September 30, 1992, and NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," published in November 1995.

Regulatory Assessment: Continued operation is allowed while a final resolution is developed because BWR licensees have adequately responded to NRCB 93-02 and its supplement. These bulletins requested licensees to take interim actions to ensure their ability to mitigate a LOCA/ECCS strainer clogging event. Measures have been requested on a related issue in NRCB 95-02 as of October 17, 1995 which will have an impact on the LOCA debris issue. The bulletin requested licensees to implement a suppression pool cleaning program and to strengthen their foreign material exclusion (FME) practices. The effect of the actions requested in the bulletin will be to minimize the amount of debris in the suppression pool which could potentially clog the ECCS strainers.

Current Status: Draft Bulletin and Regulatory Guide (RG) have undergone a 60-day public comment period. The staff has dispositioned the public comments on the draft Bulletin and RG. The proposed resolution in the draft bulletin consists of three options. The first option is to install a large capacity passive strainer design with sufficient capacity to handle a bounding scenario. The second option is to install a self-cleaning strainer design and implement a program to clean the suppression pool every outage. The third option is to install a backflush system. RES contractor analytical work is completed and a confirmatory experimental phase is ongoing. Public comments have been received and dispositioned on the contractor (SEA) report (NUREG/CR-6224), and the final report was published in November 1995. The staff issued an urgent bulletin on October 17, 1995 (NRCB 95-02). The staff will track the bulletin and its responses through an MPA number.

Contacts:	NRR Technical Contact:	R. Elliott, SCSB, 415-1397
	RES Contact:	A. Serkiz, EIB, 415-3942
	NRR Lead PM:	D. Lynch, DRPW, 415-3023

References:

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

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

3. NRC Information Notice 95-47, Revision 1, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," dated November 30, 1995.

4. NRC Information Notice 93-34 and Supplement 1, "Potential for Loss of Emergency Core Cooling Function due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1995, and May 6, 1995.

5. NRC Bulletin 93-02 and Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated May 11, 1993, and February 18, 1994.

6. NRC Information Notice 92-85, "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," dated December 23, 1992.

7. NRC Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," dated September 30, 1992.

8. NRC Information Notice 88-28, "Potential for Loss of Post LOCA Recirculation Capability Due to Insulation Debris Blockage" dated May 19, 1988.

remaining plants and hold a public workshop. Based on feedback from the workshop, the staff will finalize the inspection procedure, and the approach and schedule for evaluating A/M implementation for the remaining plants.

Originating Document: SECY-88-147, Integration Plan for Closure of Severe Accident Issues, May 25, 1988.

Regulatory Assessment: Accident management programs are being implemented by licensees as part of an initiative to further reduce severe accident risk below its current, and acceptable, level. Consequently, this is a non-urgent regulatory action and continued facility operation is justified.

Current Status: Severe accident management guideline documents have been submitted by each of the PWR owners groups, and reviewed by the staff. The BWROG has submitted two major accident management products: an overview document on February 3, 1995, and an emergency procedure and severe accident guidelines (EPG/SAG) document on April 6, 1995. The BWROG response to staff comments on the overview document was received on March 6, 1996. The BWROG plans to submit remaining documents to NRC for information in April , including: (1) revised draft EPG/SAG and associated draft technical basis document, including hydrogen control measures for Mark III containments, and (2) a "strawman" position paper on operator responsibility for SAMG and how this would be tested in operator exams. A follow-up meeting to discuss specific staff concerns regarding the BWROG products is tentatively planned for June 1996.

Licensee target dates for completing A/M implementation have been submitted to NRC, and a draft TI for use in the pilot inspections has been completed. Comments on the draft TI have been received from the NRC Region offices. The staff met with industry on February 22, 1996 and ACRS on March 1, 1996 to discuss plans for inspecting utility implementation of the formal industry position on severe accident management and major elements of the draft TI. The staff will visit approximately 2 to 4 sites in 1996 for the purpose of obtaining an early understanding of how the various elements of the formal industry position are being implemented. The information and perspectives obtained through these visits as well as comments from the Region offices will be used to update the draft TI. The draft TI will be made available to NEI and the public after the information-gathering visits.

References:

1. Memorandum from F. Rowsome to W. Minners, "A New Generic Safety Issue: Accident Management," April 16, 1985

2. SECY-88-147, Integration Plan for Closure of Severe Accident Issues

3. SECY-95-079, Implementation Plan for Probabilistic Risk Assessment

4. SECY-89-012, Staff Plans for A/M Regulatory and Research Programs

5. Generic Letter 88-20, Supplement 2, April 4, 1990

6. Letter from W. Rasin to W. Russell, November 21, 1994

7. Letter from W. Russell to W. Rasin, January 9, 1995

NRR Technical Contact: R. Palla, SCSB, 415-1095
NRR Lead PM: Ramin Assa, DRPW, 415-1391

ACCIDENT MANAGEMENT IMPLEMENTATION

Last Update: 3/28/96
Lead NRR Division: DSSA

MILESTONES	DATE (T/C)
1. Review BWROG Severe Accident Management Guidance (SAMG) documents	07/96T
2. Review severe accident training materials and BWROG prioritization methodologies	06/95C
3. Develop TI for pilot inspections	
Initial draft (for internal use)	11/95C
Site visits of "in-progress" activities	11/96T
Revised draft (to NEI and public)	12/96T
Final TI	03/97T
4. Complete pilot inspections and follow-up	12/97T
5. Revise inspection procedures (IP) and hold public workshop	
Draft IP	03/98T
Public meeting/workshop	05/98T
Final IP	07/98T
6. Review remaining plants	TBD

Description: This action plan is intended to guide staff efforts to assess the quality of utility implementation of accident management (A/M), and the manner in which insights from the IPE program have been incorporated into the licensees A/M program. Specific review areas will include: development and implementation of plant-specific severe accident management guidelines (SAMG), integration of SAMG with emergency operating procedures and emergency plans, and incorporation of severe accident information into training programs.

Historical Background: The issue of A/M and the potential reduction in risk which could result from developing procedures and training operators to manage accidents beyond the design basis was first identified in 1985 [1]. A/M was evaluated as Generic Issue 116 and subsumed by A/M-related research activities in late 1989. Completion of A/M is a major remaining element of the Integration Plan for Closure of Severe Accident Issues [2]. The development of generic and plant-specific risk insights to support staff inspections utility A/M programs is also identified in the Implementation Plan for Probabilistic Risk Assessment [3]. NRC's goals and objectives regarding A/M were established at the inception of this program [4]. Generic A/M strategies were issued in 1990 for utility consideration in the IPE process [5]. The staff has continued to work with industry to define the scope and content of utility A/M programs and these efforts have culminated in industry-developed A/M guidance for utility implementation. Industry has committed to implement an accident management program at each NPP [6]. NRC has accepted the industry commitment and developed tentative plans for staff inspection of utility implementation [7].

Proposed Actions: Specific actions included in the A/M action plan are: (1) complete the review of BWROG SAMG documents, (2) conduct site visits in 1996 to observe how the elements of the formal industry position are being implemented, (3) complete the draft Temporary Instruction (TI) using the information and perspectives obtained through the site visits, (4) complete pilot inspections and follow-up, and (5) develop an inspection procedure for use at

FIRE PROTECTION TASK ACTION PLAN

Last Update: 03/27/96
Lead NRR Division: DSSA

MILESTONES		DATE (T/C)
1.	Semiannual Commission status reports	Last: 09/20/95C Next: 04/96T
2.	Recommendations for action (Part I)	01/97T
3.	Recommendations for future study (Part II)	05/97T
4.	Confirmation issues (Part III)	05/97T
5.	Other issues (Part IV)	08/95C

Description: The Fire Protection Task Action Plan (FP-TAP) is used to track and manage implementation of the recommendations made in the "Report on the Reassessment of the NRC Fire Protection Program," of February 27, 1993.

Historical Background: In February 1993, the Office of Nuclear Reactor Regulation (NRR) completed a reassessment of the reactor fire protection review and inspection programs in response to programmatic concerns raised during the review of Thermo-Lag fire barriers. The results of the reassessment were documented in the "Report on the Reassessment of the NRC Fire Protection Program," of February 27, 1993. The staff prepared the FP-TAP to implement the recommendations made as a result of the reassessment report.

Proposed Actions: The FP-TAP tracks the implementation of a wide range of technical and programmatic fire protection issues. It includes recommendations for action (Part I), recommendations for further study (Part II), confirmation (Part III), and lessons learned (Part IV). The staff is implementing the recommendations, priority order, as resources allow. The staff focus is now on implementing its plan for future direction of the NRC fire protection program with emphasis on the fire protection functional inspection (FPFI) programand centralizing the management, by NRR, of the FPFI program and all other reactor fire protection work. The principal objective of these efforts is to ensure that the NRC has a strong, broad-based and coherent fire protection program which is commensurate with the safety significance of the subject.

Originating Document: "Report on the Reassessment of the NRC Fire Protection Program," February 27, 1993.

Regulatory Assessment: Each operating reactor has an NRC-approved fire protection plan that, if properly implemented and maintained, satisfies 10 CFR 50.48, "Fire protection," and General Design Criterion 3, "Fire protection." Therefore, each plant has an adequate level of fire safety and the individual action plan items are receiving appropriate priority.

Current Status:
The Plant Systems Branch (SPLB) continued to work with Probabilistic Risk Assessment (PRA) Branch staff and Brookhaven National Laboratory (BNL), its technical assistance contractor, to evaluate the risk associated with the post-fire safe-shutdown methodology that imposes a

self-induced station blackout. SPLB has reviewed BNL final draft report on the use of self-induced station blackout by licensees. BNL and the staff developed a probabilistic risk assessment (PRA) model for assessing the risk significance of the self-induced station blackout methodology. The staff will apply the PRA model to two plant-specific cases. The staff is also working on resolving an issue recommended for further study, fire barrier reliability, under Generic Safety Issue (GSI) 149, "Adequacy of Fire Barriers." The staff and BNL have performed scoping analyses, using fault trees and event trees, to assess the effectiveness of a degraded fire barrier in mitigating the consequences of a fully developed fire in a plant area that is important to post-fire safe shutdown. The staff and BNL discussed the preliminary results of these two studies and future plans with the Advisory Committee on Reactor Safeguards (ACRS) on February 29, 1996. By letter of March 15, 1996, the ACRS gave its comments and recommendations to SPLB. The staff is preparing a response to the ACRS letter.

The staff continued to work with Scientech, its technical assistance contractor, to establish a task order for the development of the FPFI program.

The staff prepared the semiannual report to the Commission on the status of the FP-TAP that is due April 1996.

Several tasks are on hold until an expected increase of fire protection resources is implemented. The tasks that need to be rescheduled include (1) a fire protection training program, (2) two recommendations for further study, shutdown operability requirements, and (3) several remaining confirmation issues.

Contact: D. Oudinot, DSSA, 301-415-3731

References:

"Report on the Reassessment of the NRC Fire Protection Program," of February 27, 1993.

SECY-95-034, "Status of Recommendations Resulting From the Reassessment of the NRC Fire Protection Program," February 13, 1995.

Memorandum of September 20, 1995, from J. M. Taylor, EDO, to the Commission, "Semiannual Report on the Status of the Thermo-Lag Action Plan and otection Task Action Plan."

PRA IMPLEMENTATION ACTION PLAN

Last Update: 3/29/96
Lead NRR Division: DSSA

MILESTONES		DATE(T/C)
1.	ACRS Meeting	07/94C 12/96T
2.	Commission Briefing	08/94C
3.	Publish PRA Policy Statement for 60-day comment period	12/94C
4.	ACRS Subcommittee Meeting	09/94C 07/96T
5.	Conduct Public Workshop on PRA Implementation Plan	12/94C
6.	Publish final PRA policy statement	08/95C
7.	Semi-annual Update to Commission	04/95C 04/96T 10/96T
8.	Detailed Implementation	NA
1.1(a)	Develop draft Standard Review Plans for risk-informed regulation for ACRS review	11/96T
1.1(b)	Publish draft Standard Review Plans for Public comment	12/96T
1.1(c)	Final draft Standard Review plans for ACRS review	9/97T
1.1(d)	Publish final Standard Review Plans	12/97T
1.2	Pilot Applications to Specific Regulatory Initiatives: (a) MOVs (b) IST (c) ISI (d) Graded QA (e) Maintenance Rule (f) Technical Specifications (g) Other applications to be identified later	 (a) 2/96C (b) 9/96T (c) 6/97T (d) 12/96T (e) 09/95C (f) 09/96T
1.3(a)	Develop Inspection Guidance to Use IPEs and Plant-Specific PRAs	12/96T
1.3(b)	Develop training course for inspectors	12/96T
1.3(c)	Support regional inspection activities	Ongoing
1.4	Operator Licensing - Revise Examiner's Handbook to Reflect Revised Knowledge & Abilities Based on Risk Insights	06/96T

MILESTONES		DATE(T/C)
1.5	Event Assessment - (a) Conduct event assessment of reactor events (b) Assess desirability of risk assessment on non-power reactors	(a) Ongoing (b) TBD
1.6	Review Adequacy of Licensee Analysis in IPEs/IPEEEs	6/97T
1.7 Apply Guidance to Assess Effectiveness of SBO and ATWS Rules		09/97T
1.8(a)	Staff review of PRAs for design certification applications	Ongoing
1.8(b)	Develop SRP for Review of PRAs for Evolutionary Reactor Designs	12/99T
1.8(c)	Develop Guidance for Use of Risk in Simplification of Emergency Planning Requirements	12/96T
1.9	Accident Management - Develop Risk Insights to Review and Inspect Industry Accident Management Programs	TBD

Description: This action plan is intended to describe the process for the staff to use PRA method and technology in the agency's effort toward risk-informed regulatory approach. The plan encompasses methods development, pilot applications, and staff training. The plan will be used to ensure timely and integrated agency-wide effort that is consistent with the PRA Policy Statement.

Historical Background: The NRC has been making use of PRA technology to varying degrees in its regulatory activities since WASH-1400. Prior to 1991, this had been an ad hoc application, depending on the availability of expertise in various techr[illegible] roups. Since 1991, there have been a number of high-level studies within NRC that have focu[illegible] on the status of PRA use and its role in the regulatory process. Collectively, the findings and recommendations from these studies support the view that there is a need for increased emphasis on PRA technology applications. For the full value of our investment in risk assessment methodology to be achieved, it is important that consistent high-level agency guidance be provided on the appropriate use of PRA. To this end, in November 1993, the Office Directors of NRR, AEOD, NMSS, and RES proposed to take the initiative in providing guidance on coordination and expectations for PRA efforts. Specifically, they proposed to develop an integrated plan for the staff's risk assessment and risk management practices. In August 1994, the staff submitted SECY-94-219, "Proposed Agency-Wide Implementation Plan For Probabilistic Risk Assessment," for the Commission's information. On March 30, 1995, The staff submitted SECY-95-079, "Status Update of the Agency-Wide Implementation Plan for PRA," and briefed the Commission on the subject on April 5, 1995. On May 18, 1995, the staff forwarded SECY-95-126, "Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," for Commission vote. On June 8, 1995, the staff briefed the ACRS on the PRA policy statement. The final PRA policy statement was published in the *Federal Register* on August 16, 1995.

Proposed actions: The PRA Implementation Plan includes activities for NRR, RES, AEOD, and NMSS staff to increase the use of PRA methods in all regulatory matters. NRR focuses on the PRA applications in reactor regulations, the development of standard review plans, the pilot programs to use PRA technology in specific regulatory initiatives, events assessment, and working with regions on risk-informed inspections. RES focuses on the IPE/IPEEE reviews, PRA method and quality, and the development of PRA regulatory guides for the industry. AEOD focuses on risk-informed trends and patterns analysis, reliability data for PRA applications, and staff training. NMSS focuses on using PRA in high and low level waste issues. The detailed actions are described in the PRA Implementation Plan.

Originating Document: Memorandum dated November 2, 1993, T. Murley et al. to J. Taylor, "Agency Directions For Current and Future Uses of Probabilistic Risk Assessment".

Regulatory Assessment: This action plan is meant to improve the regulatory process by developing state-of-the-art PRA tools that will expand the use of PRA technologies in making regulatory decisions. The plan is not intended to correct safety problems at licensed facilities. Therefore, continued facility operation is justified.

Current Status: On November 17, 1995, a memorandum was forwarded to senior NRR management providing additional guidance on implementing the Commission's PRA Policy Statement and managing tasks contained in the PRA Implementation Plan. As a result of this memorandum, several additional Action Plans are expected to be developed for individual line items in the PRA Implementation Plan. In addition, more detailed information concerning PRA Implementation Plan activities will be collected so that more accurate and timely status of all NRR PRA Implementation Plan activities can be maintained in the "living" PRA Implementation Plan. On November 27, 1995, the staff forwarded SECY-95-280, "Framework For Applying Probabilistic Risk Analysis In Reactor Regulation," to provide a general structure to ensure consistent and appropriate application of PRA methods and outlined a process for developing guidance and standards.

On November 20, the staff briefed Chairman Jackson on the activities regarding risk-informed technical specifications. On November 30, 1995, Chairman Jackson issued a memorandum requesting the staff to develop action plans and timetables to provide better focus and accelerate NRC's risk-informed regulatory effort. The staff briefed Chairman Jackson concerning PRA Implementation Plan Pilot Applications and Guidance Development on December 7, 1995. On January 3, 1996, the EDO forwarded a memorandum to Cha'-man Jackson responding to the November 30, 1995 SRM. This memorandum described ... aff action plan which included the PRA pilot programs and the accelerated milestones for the . .elopment of regulatory guidance documents for utilizing PRA in reactor related activities. Several teams consisting of NRR and RES staff members have been established to develop the broad scope and application specific Regulatory Guides and Standard Review Plans. Work is currently ongoing.

On February 27 and 28, 1996, the staff met with the ACRS PRA subcommittee to discuss technical issues related to risk-informed regulation. This was followed by a meeting with the ACRS full committee on March 8, 1996. On March 26, 1996, the EDO forwarded a memorandum to the Commission updating the progress and status of the PRA Implementation Plan. The corresponding Commission briefing is scheduled for April 4, 1996.

NRR Technical Contacts: Tony Hsia, SPSB, 415-1075

References:

SECY-94-219, "Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment"

SECY-95-079, "Status Update of The Agency-Wide Implementation Plan for Probabilistic Risk Assessment"

SECY-95-126, "Final Policy Statement on The Use of Probabilistic Risk Assessment Methods In Nuclear Regulatory Activities"

SECY-95-280, "Framework For Applying Probabilistic Risk Analysis In Reactor Regulation"

Memorandum from James M. Taylor to Chairman Jackson, "IMPROVEMENTS ASSOCIATED WITH MANAGING THE UTILIZATION OF PROBABILISTIC RISK ASSESSMENT (PRA) AND DIGITAL INSTRUMENTATION AND CONTROL TECHNOLOGY," January 3, 1996.

Memorandum from James M. Taylor to the Commission, "Status Update of the Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA) (From March 30, 1995 to February 29, 1996)," March 26, 1996.

PRA IMPLEMENTATION ACTION PLAN 1.2(d)
Graded Quality Assurance Action Plan

Last Update: 3/28/96
Lead NRR Division: DRCH
Support Division: DSSA

MILESTONES	DATE (T/C)
1. Issued SECY 95-059	03/95C
2. Begin interactions with volunteer licensees - Palo Verde letter dated 4/6/95 - Grand Gulf meeting 5/4/95 - South Texas meetings on 4/19/95 and 5/8/95	05/95C
3. NRC Steering Group meetings to guide working level staff activities - Meetings on: 8/25/95, 10/10/95, 10/25/95	As Needed
4. Staff interactions with Palo Verde - Site visit on 5/23/95 on ranking and QA controls - NRC letter dated 7/24/95 on proposed QA controls - Site visit on 8/29-30/95 on risk ranking - Site visit on 9/6-7/95 on procurement QA controls - NRC letter conveying trip reports issued on 12/4/95	Ongoing through 12/30/96
5. Staff interactions with South Texas - Meeting on 7/17/95 on project status - Site meeting on 10/3-4/95 on risk ranking and QA controls - Meeting on 12/7-8/95 to discuss risk ranking and QA controls - South Texas Submittal of QA Plan for implementation of graded QA, 4/96 est. - South Texas begins implementation of grading specific QA elements, 7/96 est.	Ongoing through 12/30/96
6. Staff interactions with Grand Gulf - Site meeting on 7/11-14/95 to observe expert panel - Meeting at hdqt. on 10/24/95 on QA controls - Meeting at RIV on 11/16/95 on graded QA effort - Site meeting on 11/17 to observe expert panel - GGNS system and component ranking criteria under staff evaluation	Ongoing through 12/30/96
7. Revision 3 of Draft Evaluation Guide for Volunteer Plants issued for staff comment	07/95C
8. Revision 4 of Draft Evaluation Guide for Volunteer Plants Issued for Steering Group Review	10/95C
9. Issue letter to 3 volunteer plants outlining program objectives and review expectations. Distribute staff evaluation guide to licensees.	1/96C
10. Evaluation Guide Issued for use by staff in evaluating volunteer plants - Meeting scheduled with volunteer plants to receive feedback on staff evaluation guide	1/96C 4/96T

11. Regulatory Guide and SRP development milestones per PRA Action Plan - Draft SRP and RG for cognizant office review and comment - Draft SRP and RG for inter-office review and concurrence - Draft SRP and RG for ACRS/CRGR review - Draft SRP and RG for public comment - Draft SRP and RG public comment period ends - Final draft SRP and RG for ACRS/CRGR review - Final draft SRP and RG for inter-office concurrence - Publish final SRP and RG	 7/31/96T 9/30/96T 10/31/96T 12/31/96T 3/3/97T 9/1/97T 12/1/97T 12/31/97T
12. ACRS Briefings - Expert Panel and deterministic considerations - graded QA	 2/27-28/96C 4/11 /96T
13. Disseminate lessons learned to date at regional counterpart meetings	5/96T
14. Issue Lessons Learned NUREG report regarding Graded QA Programs at volunteer plants	11/96T
15. (This item has been superseded by the SRP/RG Development action plan, see item 11 above)	
16. Public Workshop on Graded QA	2/97T
17. (This item has been superseded by the SRP/RG Development action plan, see item 11 above)	
18. Issue Staff Inspection Guidance (Reactive IP)	5/97T
19. Conduct NRC Staff Training	5/97T
20. (This item has been superseded by the SRP/RG Development action plan, see item 11 above)	
21. Issue SECY Update (close-out of action plan)	12/97T

Description: Prepare staff evaluation guidance and regulatory guidance for industry implementation for the grading of quality assurance (QA) practices commensurate with the safety significance of the plant equipment. The development of this guidance will be based on staff reviews of regulatory requirements, proposed changes to existing practices, staff development of a draft regulatory guide with input from a national laboratory, and assessment of the actual programs developed by the three volunteer utilities implementing graded quality assurance programs.

Historical Background: The NRC's regulations (10 CFR Part 50, Appendices A & B) require QA programs that are commensurate (or consistent) with the importance to safety of the functions to be performed. However, the QA implementation practices that have evolved have often not been graded. In the development of implementation guidance for the maintenance rule, a methodology to determine the risk significance of plant equipment was proposed by the industry (NUMARC 93-01). During a public meeting on December 16, 1993 the staff suggested that the industry could build on the experience gained from the maintenance rule to develop implementation methodologies for graded QA. The staff had numerous interactions with the Nuclear Energy Institute (NEI) during calendar year 1994 as the graded QA concepts were discussed and the initial industry guidelines were developed and commented on. In early 1995, three licensees (Grand Gulf, South Texas, and

Palo Verde) volunteered to work with the staff. The staff has reviewed the licensee developmental graded QA efforts.

Proposed Actions: The goal of the action plan is to utilize the lessons learned from the 3 volunteer licensees to modify staff-developed draft guidance to formulate regulatory guidance on acceptable methods for implementing graded QA. The staff will develop a regulatory guide based in part on input from Brookhaven National Laboratory, a standard review plan revision for Chapter 17, and a reactive inspection procedure (IP) for graded QA. An inter-office team has been established to prepare the regulatory guidance documents and test their implementation during the evaluation of volunteer plant activities.

Originating Document: Letter from J. Sniezek, NRC to J. Colvin (NUMARC) dated January 6, 1994, describing the establishment of NRC steering group for the graded QA initiative.

Regulatory Assessment: Existing regulations provide the necessary flexibility for the development and implementation of graded quality assurance programs. The staff will issue a NUREG report regarding the lessons learned from the volunteer plant implementations. Additional regulatory guidance will be issued to either disseminate staff guidance or endorse an industry approach. Planned guidance for the staff will involve an evaluation guide for application to the volunteer plants, the lessons learned report, training sessions and public workshops, Standard Review Plan revision, and inspection guidance in the form of a reactive IP. The staff is evaluating the appropriate mechanism for inspections of the risk significance determination aspects of graded QA programs.

The safety benefits to be gained from a graded QA program could be significant since both NRC reviews and inspections and the industry's quality controls resources would be focused on the more safety significant plant equipment and activities. Secondarily, cost savings to the industry could be realized by avoiding the dilution of resources expended on less safety significant issues. The time frame to complete this action plan is directly related to the overall PRA implementation plan schedules.

Current Status: A draft evaluation guide for NRC staff use has been prepared for application to the volunteer plants implementing graded quality assurance programs. The staff will utilize the guide for the review of the volunteer plant graded QA programs. e guide and the staff's proposed interaction framework has been transmitted in a letter to ree volunteer licensees. The letter seeks licensee comments. Outlines of a draft regulatory guide and SRP for both risk ranking and grading of QA controls have been prepared and circulated for review for the inter-office team. A meeting is planned with the three volunteer licensees on April 11, 1996 to receive their feedback on the staff developed evaluation guide. In addition, a presentation on graded QA will be made to the full ACRS on April 11th.

NRR Contact: S. Black 415-1017, R. Gramm 415-1010
RES Contact: R. Woods 415-6622

References:

1) Letter from J. Sniezek (NRC) to J. Colvin (NEI) dated 1/6/94
2) Regulatory Guide 1.160
3) NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"
4) SECY-95-059, "Development of Graded Quality Assurance Methodology", 3/10/95
5) Letter from B. Holian (NRC) to W. Stewart (APSCo) dated 7/24/95
6) Letter from C. Thomas (NRC) to W. Stewart (APSCo) dated 12/4/95
7) Memorandum from S. Black to W. Beckner and W. Bateman dated 1/24/96, Draft Staff Evaluation Guidance

ENVIRONMENTAL QUALIFICATION TASK ACTION PLAN

Last Update: 03/27/96
Lead NRR Division: DSSA

MILESTONES	DATE (T/C)
1. Inform Commission	05/93C
2. Meet With Industry	Ongoing
3. Programmatic Review	TBD
4. Risk Assessment	4/96T
5. Data Collection and Analysis	Ongoing
6. Status Review	TBD
7. Technical Issues	10/98T
8. Options for Resolution	TBD
9. Implementation	TBD

Description: This action plan will evaluate environmental qualification (EQ) issues, including operating experience, testing methodology, and adequacy of current rule and guidance for operating reactors. It will resolve EQ issues for aging operating reactors and license renewal.

Historical Background: A review of environmental qualification requirements for license renewal and failures of qualified cables during research tests led to the development of the EQ Task Action Plan (TAP), which was issued in July 1993. The EQ TAP was developed to address: (1) staff concerns regarding the differences in EQ requirements for older and newer plants; (2) concerns raised by some research tests which indicate that qualification of some electric cables may have been non-conservative; and (3) concerns that programmatic problems identified in the staff Fire Protection Reassessment Report might also exist in the NRC EQ Program.

Proposed Actions: The EQ TAP includes meetings with industry, a program review of EQ, data collection and analysis, a risk assessment, and research on aging and condition monitoring. Annual Commission papers are written to update the status of the EQ TAP. The staff will develop options for resolving EQ concerns, which may include issuing a generic letter, changing the rule, or documenting the acceptability of the current EQ rule and standards. The basis for the appropriate regulatory action will be documented.

Originating Document: June 28, 1993, memorandum from Samuel J. Chilk to James M. Taylor (SECY 93-049); May 27, 1993, letter to the Commission from J. Taylor on Environmental Qualification of Electric Equipment.

Regulatory Assessment: Depending on the application, failure of these cables during or following design-basis events could affect the performance of safety functions in nuclear power plants. There is no immediate safety issue because of the degree of conservatism already included in the EQ qualification test margins.

Current Status: The programmatic review is nearing completion. The second draft of the report that summarizes the results of the program review was completed in January 1996 and is undergoing management review. Data collection and analysis activities are continuing. The staff review of past and ongoing EQ-related work, including literature from qualification tests and research has been completed. The Literature Review Report will be published as a NUREG/CR report in April 1996. SPLB staff is preparing a position paper based on the preliminary risk scoping study and other past PRA work for EQ to complete Task 4, Risk Assessment. This paper will provide recommendations regarding further work in PRA for EQ.

BNL has developed cable testing and cable acquisition programs and has identified some sources of naturally aged cable for the program. The cable test plan includes testing of new, naturally aged, and artificially aged cables and evaluation of condition monitoring techniques that could give insights into methods for determining how cable is actually aging and performing in plants. The plan includes LOCA testing of some cables under design-basis event conditions. These plans were released for public comment in February 1996. RES and BNL with NRR assistance continue to pursue the acquisition of naturally aged cable samples from PGE/Trojan and EPRI. BNL, with assistance from RES and NRR, performed an audit of the Wyle test lab QA program the week of March 25. Testing is scheduled to start within a few months.

As activities of the program review and data collection proceed or are completed, the staff will make changes to the research program as necessary. Following completion of the program review and data collection effort, staff activities will focus on research in the areas of accelerated aging, condition monitoring techniques, and accident testing. Research activities will extend over the next few years.

Contacts:	NRR Technical Contact:	G. Hubbard, SPLB, 415-2870
	RES Contact:	S. Aggarwal, EMEB, 415-5849
	NRR Lead PM:	L. Olshan, DRPE, 415-3018

References:

Letter to the Commission from J. Taylor on Environmental Qualification of Electric Equipment dated May 27, 1993 (Accession No. 9308180153).

Staff requirements memorandum (SECY 93-049) dated June 28, 1993 (Accession No. 9409010107).

Task Action Plan for Environmental Qualification and upc July 1, 1993, April 8, 1994, November 16, 1994, and June 27, 1995 (Accession Nos. 9308120145, 9404260206, 950110431, 9507110203, respectively).

RES Program Plan for Environmental Qualification, July 7, 1994 (Accession No. 9407250066).

GENERIC SPENT FUEL STORAGE POOL
PART A: OPERATING FACILITIES

Last Update: 3/27/96
Lead NRR Division: DSSA

	MILESTONES	DATE (T/C)
1.	Identify significant SFP concerns.	12/94C
2.	Review existing NRC guidance and requirements.	08/94C
3.	Report significant SFP problems to NRR management.	12/94C
4.	Develop a SFP inspection plan.	1/95C
5.	Conduct inspections of selected plants.	06/95C
6.	Evaluate and report results of inspections.	09/95C
7.	Assess risk/significance of individual concerns.	4/96T
8.	Assess monitoring of potential off-site releases.	4/96T
9.	Assess radioactive material storage practices.	4/96T
10.	Propose course of action.	5/96T
11.	Take selected actions.	TBD

Description: The action plan is intended to encompass Spent Fuel Pool (SFP) issues identified through a 1994 special inspection at Dresden 1, the staff's review of loss of SFP cooling concerns at Susquehanna Steam Electric Station (SSES), and other SFP concerns identified as part of this plan. Specific review areas identified through implementation of this action plan include plant design features and administrative controls that affect the probability of spent fuel pool boiling, adverse environmental effects on essential equipment due to boiling, significant loss of spent fuel pool coolant inventory, adverse radiological conditions, unplanned spent fuel pool reactivity changes, undetected spent fuel pool events, and adverse ef' ts of control system actuations.

Historical Background: In November 1992, two engineers, who formerly worked under contract for the Pennsylvania Power and Light Company (PP&L), filed a report contending that the design of the Susquehanna station failed to meet regulatory requirements with respect to sustained loss of the cooling function to the SFP that mechanistically results from a loss-of-coolant accident (LOCA) or a loss of offsite power (LOOP). The licensee (PP&L) and the engineers each made a series of additional submittals to the NRC and participated in public meetings with the NRC staff to describe their respective positions on a number of technical and licensing issues. In order to inform the nuclear power industry of the issues, the agency issued Information Notice (IN) 93-83 on October 7, 1993. The staff evaluated these issues as they related to Susquehanna using a probabilistic safety assessment, a deterministic engineering assessment, and a licensing basis analysis. The staff issued their final safety evaluation report on June 19, 1995. This closed the Susquehanna action plan (TAC No. M85337).

A generic action plan was developed and adopted on October 13, 1994, with two parts. Part A (TAC No. M88094) encompasses the staff's review of generic issues relating to the SFP at operating reactor facilities. Part B (TAC Nos. M40004, M90441, and M93805) includes applicable issues from the Part A review and concerns from the Dresden 1 special inspection particular to permanently shutdown facilities with stored, irradiated fuel to establish evaluation criteria for spent fuel pools at permanently shutdown facilities. Part B was included after the special inspection at Dresden 1 determined that problems in implementing the facility's decommissioning plan combined with certain SFP design features created the potential for a substantial loss of SFP water inventory. Dresden 1, which is permanently shutdown, experienced containment flooding due to freeze damage to the service water system on January 25, 1994, and the licensee for Dresden 1 reported a similar threat to SFP integrity. This licensee report resulted in the special inspection.

The principal concerns included in Part A of the generic action plan involve the potential for a sustained loss of SFP cooling capability, which was identified through the report filed with the NRC relating to Susquehanna, and the potential for a substantial loss of SFP coolant inventory, which was given renewed emphasis following the Dresden 1 special inspection. Postulated adverse conditions that may develop following a LOCA or a sustained loss of power to SFP cooling system components could prevent restoration of SFP decay heat removal. The heat and water vapor added to the building atmosphere by subsequent SFP boiling could cause failure of accident mitigation or other safety equipment and an associated increase in the consequences of the initiating event. Incomplete administrative controls combined with certain design features, particularly at the oldest facilities, may create the potential for a substantial loss of SFP coolant inventory and the associated consequences, which include high local radiation levels due to loss of shielding, unmonitored release of radiologically contaminated coolant, and inadequate cooling of stored fuel.

Proposed Actions: Specific actions included in Part A of the generic action plan are: (1) determination of the safety significance of identified concerns, (2) determination of the facilities where the concerns may be applicable, (3) evaluation of the adequacy of present SFP designs, (4) evaluation of the adequacy of current NRC guidance for SFP designs, and (5) evaluation of the need for generic actions to address significant issues at operating and permanently shutdown facilities. Based on findings from these review areas and their risk significance, the staff will develop criteria for specific spent fuel pool operations for potential use in formulating generic communications, revisions of regulatory guidance, and other appropriate regulatory actions.

Originating Documents: (1) Letter from D.A. Lochbaum and D.C. Prevatte to T. Martin, NRC, November 27, 1992, "Susquehanna Steam Electric Station Docket No. 50-387, License No. NPF-14, 10 CFR 21 Report of Substantial Safety Hazard;" (2) Inspection Report No. 50-010/94001.

Regulatory Assessment: The postulated events do not p[illegible] undue risk to the public based on the availability of common design features that help prot[illegible]ored irradiated fuel, protect essential reactor safety systems, and prevent development of adverse radiological conditions. These design features include the provision of diverse means of cooling, the strong structural design of the spent fuel pool, the absence of drainage paths from the pool, the anti-syphon protection on piping within the spent fuel pool, the availability of multiple sources of make-up water, spent fuel pool instrumentation with control room annunciation, the maintenance of a substantial shutdown reactivity margin in the pool, radiation shielding provided by coolant inventory, and spent fuel pool water purification systems. Additionally, the relatively slow evolution of these events in the spent fuel pool resulting from the initial large cooling water inventory creates significant opportunity for operator recovery prior to experiencing adverse conditions or consequences. Therefore, continued facility operation is justified.

Current Status: The identification of concerns for evaluation, and review of existing guidance have been completed. On-site safety assessments of spent fuel storage have been completed at Brunswick, Monticello, Comanche Peak, and Ginna. The assessment team concluded that the potential for a sustained loss of spent fuel pool cooling or a significant loss of spent fuel pool coolant inventory at the sites visited was remote, based on certain design features and operational controls. The team found that other concerns within the scope of the action plan review were much less significant in terms of risk at the plants visited. Individual assessment reports have been completed for Brunswick, Monticello, Comanche Peak, and Ginna.

An FSAR-based review to identify facilities whose design is not well represented by any of the facilities reviewed through on-site assessments has been completed by DSSA staff. Based on this FSAR review of 16 sites in addition to the sites visited, DSSA has determined that the significant spent fuel pool issues are best resolved through a site-specific evaluation because of the small number of facilities affected by each particular concern and variations in design and operation of the spent fuel pool and associated systems. To accomplish this task, the FSAR-based review has been expanded to encompass development of a data-base specifying the current licensing basis for the SFP cooling system, selected design basis parameters, and current operating procedures relevant to SFP cooling for all facilities. Projects initiated this expanded review on January 16, 1996. Project Managers have assumed the data collection function, which is being performed under TAC M94480, for this task with completion expected by April 9, 1996. In order to ensure a more consistent licensing basis determination, SPLB has been devoting substantial resources to a plant-by-plant licensing basis review to forward to Project Managers prior to on-site visits. To accommodate this effort, completion dates for Milestones 7,8,9, and 10 have been extended by one month. This extension does not impact our ability to meet commitments to Chairman Jackson.

The staff briefed Chairman Jackson regarding SFP issues on February 1, 1996. Following the briefing, the staff committed to provide results of the plant-specific review effort to the Chairman by (May 8, 1996), and the staff committed to prepare a course of action for resolution of significant issues by (June 28, 1996), with a Commission Briefing to follow in July.

Approximately 26 total issues in the major review areas have been identified through this plan. Additional issues associated with the Millstone 1 SFP (adequacy of SFP cooling during refueling with a full core off-load) have been included in the plan. Each issue is being tracked for resolution and will be addressed on the basis of a qualitative safety assessment. An issue relating to spent fuel pool criticality control (Boraflex degradation) is being pursued through issuance of an information notice and a planned generic letter.

Contacts: S. Jones, 415-2833
J. Shea, 415-1428

<u>References</u>:

Letter from Lochbaum and Prevatte, November 1992

Task Action Plan for Spent Fuel Storage Pool Safety, October 13, 1994 (publicly available, Accession No. 9410190155)

SER for Susquehanna, June 19, 1995 (publicly available, Accession No. 9507070008)

Information Notice 95-54, December 1, 1995 (SFP cooling design basis at Millstone 1 and Cooper)

Information Notice 93-83 (and Supplement 1), October 7, 1993 and August 24, 1995.

Information Notice 94-38, May 27, 1994 (Dresden 1 Special Inspection Results)

Inspection Report No. 50-010/94001, April 14, 1994 (Dresden 1 Special Inspection)

GENERIC SPENT FUEL STORAGE POOL
PART B: PERMANENTLY SHUTDOWN FACILITIES

Last Update: 3/27/96
Lead NRR Division: DSSA

	MILESTONES	DATE (T/C)
1.	Identify significant SFP concerns applicable to permanently shutdown facilities.	11/95C
2.	Provide technical assistance to DRPM for rulemaking or other generic activity.	TBD

Description: This Part B effort will use the results of Part A activities to establish evaluation criteria for spent fuel pools (SFPs) at permanently shutdown plants to support rulemaking and other generic activities initiated by the Decommissioning and Non-Power Reactor Project Directorate (PDND).

Historical Background: A generic action plan was developed and adopted on October 13, 1994, with two parts. Part A (TAC No. M88094) encompasses the staff's review of generic issues relating to the SFPs at operating reactor facilities. Part B (TAC Nos. M40004, M90441, and M93805) includes applicable issues from the Part A review and concerns from the Dresden 1 special inspection particular to permanently shutdown facilities with stored, irradiated fuel to establish evaluation criteria for SFPs at permanently shutdown facilities. Part B was included after the special inspection at Dresden 1 determined that problems in implementing the facility's decommissioning plan combined with certain SFP design features created the potential for a substantial loss of SFP water inventory. Dresden 1, which is permanently shutdown, experienced containment flooding due to freeze damage to the service water system on January 25, 1994, and the licensee for Dresden 1 reported a similar threat to SFP integrity. This licensee report resulted in the special inspection.

The staff issued NRC Bulletin 94-01, "Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1," on April 14, 1994. This bulletin requested all holders of licenses for nuclear power reactors that are permanently shutdown with spent fuel in the spent fuel pool to take actions to ensure the quality of the SFP coolant, the ability to maintain an adequate coolant inventory for cooling and shielding, and the necessary support systems are not degraded. In order to evaluate the management controls and SFP activities at permanently shutdown reactors, the NRC staff initiated a series of special team inspections at permanently shutdown facilities with stored, irradiated fuel in the SFP. These inspections were completed at all of the subject facilities by the first quarter of 1995.

Proposed Actions: Specific actions included in Part B of the generic action plan are: (1) the determination of significant identified concerns from Part A applicable to permanently shutdown facilities and (2) the evaluation and implementation of additional requirements specifically applicable to permanently shut down facilities with stored, irradiated fuel.

Originating Documents: Inspection Report No. 50-010/94001 for Dresden Unit 1.

Regulatory Assessment: The postulated events involving a loss of cooling do not pose undue risk to the public, because of the low residual decay heat in the spent fuel at permanently shutdown reactors and the associated long period of time available for recovery. Concerns involving maintenance of the coolant quality and ability to control coolant inventory have been addressed through the special inspection activities. Therefore, continued facility operation is justified.

Current Status: The staff determined that all significant identified concerns from Part A applicable to permanently shutdown facilities were encompassed by the special inspection activities. The special inspections found no significant deficiencies other than at Dresden 1. In response to the Dresden 1 Special Inspection findings, PDND will proceed with issuance of their decommissioning action plan. The Division of Systems Safety and Analysis will provide technical support for that action plan and other existing action plans associated with rulemaking for decommissioning facilities. Staff resources will be tracked through TACs assigned to the associated action plans.

NRR Technical Contact:	S. Jones, SPLB, 415-2833
NRR Lead PM:	R. Dudley, PDND, 415-1116

References:

Task Action Plan for Spent Fuel Storage Pool Safety, October 13, 1994 (publicly available, Accession No. 9410190155)

Information Notice 94-38, May 27, 1994 (Dresden 1 Special Inspection Results)

NRC Bulletin 94-01, April 14, 1994.

Inspection Report No. 50-010/94001, April 14, 1994 (Dresden 1 Special Inspection)

CORE PERFORMANCE ACTION PLAN

Last Update: 03/29/96
Lead NRR Division: DSSA
Supporting Division: DISP

MILESTONES	DATE (T/P/C)
Task 1 - Inspection of Nuclear Fuel Vendors (DISP)	10/96T
SPC [PWR]	06/94C
ABB/CE [PWR]	11/94C
TWC (Teledyne-Wah Chang)	12/94C
SSM (Sandvik Specialty Metals)	12/94C
WESTINGHOUSE	07/95C
GE	10/95C
FRAMATOME/COGEMA (was B&W Fuels)	05/96T
ABB/CE [BWR]	08/96T
SPC [re-inspect]	10/96T
Task 2 - Inspection of Licensee Reload Analyses (DSSA)	12/96T
RI - GPU [TMI-1];	12/95C
RII - Duke [Oconee]; SSI?[Hatch?]	03/95C; 04/96T
RIII - ComEd [Zion]; ?[]	10/94C; 06/96T
RIV - NPPD?[Cooper]; WPPS?[WNP-2]	04/96T; 08/96T
*** - APS (original pilot audit)	04/93C
Task 3 - Core Performance Data Gathering/Evaluation (DSSA)	12/96T
Regions - Morning Reports & Event Notification	09/96T
Other - Data Acquisition and Collation	09/96T
PNL - Core Performance Evaluation Analysis	12/96T
Task 4 - Participation of Regions in Action Plan (DSSA)	09/96T
Identification of Vendor Issues	
Feedback from Licensee Inspections	
Counterparts Meetings (RI-RIV)	
Task 5 - Evaluate Inspection Guidance (DSSA/DISP)	09/96T
Evaluate Results of Vendor/Licensee Inspections	
Incorporate Feedback from Regions	
Draft Guidance for Residents	
Draft Inspection Criteria and Plan Outline	
Task 6 - Evaluate Lead Test Programs for Early Identification of Performance Issues (DSSA)	tbd

Description: The action plan is intended to assess safety through inspections of fuel vendors, evaluation of licensee's reload analyses, independent evaluation of core performance information, and by regional training and interaction activity.

Historical Background: The action plan addresses the review of fuel fabrication, core design, and reload analysis issues that were discussed during the March 29, 1994, briefing given to James M. Taylor, Executive Director for Operations. The briefing presented by the Reactor Systems Branch (SRXB), Division of Systems Safety and Analysis (DSSA), covered generic fuel and core performance issues and related evaluations of fuel failures. Representatives of the Vendor Inspection Branch (VIB), Division of Reactor Inspection and Licensee Performance (DRIL), participated in the briefing. As a result of this briefing, the Office of Nuclear Reactor Regulation (NRR) was requested to prepare an action plan for a proactive approach to improve core performance in operating reactors.

Proposed Actions: Specific actions included in the action plan are: (1) evaluate fuel vendors' performance through performance-based inspections that evaluate the reload core design, safety analysis, licensing process, fuel assembly mechanical design, and fuel fabrication activities; (2) evaluate the performance of licensees that perform core reload analysis functions; (3) identify, document, and categorize core performance problems and root cause evaluations that will be further evaluated during these inspections and provide input to SALP evaluations as well as regional enforcement actions, as appropriate; and (4) train and coordinate regional support staff participating in these activities as well as evaluating the results of these activities for use in formulating generic communications, revisions of regulatory guidance and guidance for regional inspectors,. and other appropriate regulatory actions. As a result of recent generic concerns related to failure of control rods to fully insert, the plan is being expanded to review vendor lead testing programs for new fuel designs.

DSSA — The action plan identifies one or more licensee inspections in each region that shall be performed, in coordination with the regional inspectors, to assess licensee performance in reload core analysis oversight and participation. The data acquired through licensee/vendor inspections will be integrated with information supplied by the regions and other sources and will be evaluated for generic core performance indicators and industry conformance to current regulatory requirements. The end product of the initial assessment will include guidance for resident inspectors and regional staff. These activities are schedul to be completed in FY96. The ongoing activities to capture and address early warning erging issues will continue into FY97, and the action plan will be updated to reflect the living plan.

DISP — The action plan currently identifies nine vendor inspections that shall be performed by multi-disciplined inspection teams lead by the Special Inspection Branch (PSIB) with contracted technical assistance. These inspections will be completed by October 1996.

Originating Document: Memorandum from Gary M. Holahan and R. Lee Spessard to Ashok C. Thadani, dated October 7, 1994, "Action Plan to Monitor, Review, and Improve Fuel and Core Components Operating Performance"

Regulatory Assessment: Core design is a fundamental component of plant safety because maintaining fuel integrity is the first principal safety barrier (i.e., fuel cladding, reactor coolant system boundary, or the containment) against serious radioactive releases. Likewise, the safety analyses must be properly performed in order to verify, in conjunction with startup tests and normal plant parameter monitoring, that the core reload design is adequate and provide assurance that the reactor can safely be operated. Quality assurance activities are important to ensure that proper interfaces are established and that shortcuts are not taken that could degrade safety or quality.

Current Status:

DSSA — The data acquired from the vendor inspections at SPC, ABB/CE, Westinghouse, and GE are being evaluated. The vendor inspection at Framatome (B&W), in March 1996, was supported by SRXB/DSSA staff and contract specialists in reload design. Interaction with the regions is ongoing to coordinate a license inspection schedule, and SRXB participated in the Region I inspector counterparts meeting in December 1995. DSSA is re-evaluating the action plan to better integrate and prioritize its activities. Options and recommendations will be provided for management review in April 1996.

DISP — The inspection of Framatome Cogema Fuels (formerly Babcock and Wilcox Fuel Company), located in Lynchburg, Virginia, began in March 1996; however, FCF production delays will result in delaying the end of the inspection until May 1996. The remaining planned inspections include ABB Combustion Engineering's supply of a transition core reload for WNP-2 as well as a follow-up inspection of Siemens Power Corporation issues.

NRR Technical Contacts: E. Kendrick, SRXB, 415-2891
S. Matthews, PSIB, 415-3191

* time spent on-site at vendor inspections (Task 1) is allocated to appropriate fuel vendor docket #

HIGH BURNUP FUEL ACTION PLAN

Last Update: 03/29/96
Lead NRR Division: DSSA
Supporting Office: RES

MILESTONES		DATE (T/C)
1.	Issue User Need Letter to RES	10/93C
2.	Contracts Issued by RES	03/94C
3.	Schedule and Coordinate Meetings with Foreign Experimenters and Regulatory Authorities	09/95C
4.	Issue Information Notice (IN 94-64) Announcing New RIA Data	08/94C
5.	Present High Burnup Data at Water Reactor Safety Meeting	10/94C
6.	Schedule/Coordinate Industry Meetings to Discuss Actions	10/94C
7.	Determine Need for Further Generic Communications	11/94C
8.	Issue Letter to Vendors	11/94C
9.	Issue IN 94-64, Suppl. 1, Providing Data and Vendor Letter	03/95C
10.	RES Update NUREG-0933 on Generic Issue* and Plan of Action	03/95C* 01/96C
11.	Review Industry (NEI) Response	09/95C
12.	Assess Effects on Design Basis Accidents of Reduced Failure Threshold for High Burnup Fuel	09/95C
13.	Committee on the Safety of Nuclear Installations <u>Specialists Meeting on the Transient Behavior of High Burnup Fuel</u>	09/95C
14.	CNRA (OECD) Committee on Nuclear Regulatory [illegible]ies and CSNI annual meetings.	11/95C
15.	Issue Letter to NEI Assessing Industry Actions (Vendor response to IN)	04/96T
16.	Water Reactor Safety Information Meeting on High Burnup	10/95C
17.	RES Briefs ACRS and Completes Response to NRR User Need Letters	04/96T 07/96T
18.	Complete Review of Available Fuel Transient Data Relevant to Design Basis Event; Define Acceptance Criteria; Establish Schedule for Final Assessment and State Need for Further Regulatory Action	08/96T

* RES has prioritized as Generic Issue #170.

<u>Description</u>: The action plan covers assessment of fuel performance for high burnup fuel and evaluation of the adequacy of SRP licensing acceptance criteria.

<u>Historical Background</u>: Recent experimental data on performance of high burnup (>50 GWd/MTU) under reactivity insertion conditions became available in mid-1993. The unexpectedly low energy deposition (30 cal/gm) to initiation of fuel failure in the first test rod (at 62 GWd/MTU) led to a re-

evaluation of the licensing basis assumptions in the SRP. As a result, the Office of Nuclear Reactor Regulation (NRR) was requested to prepare an action plan, in coordination with the Office of Nuclear Regulatory Research (RES).

Proposed Actions: After a preliminary safety assessment was performed, an action plan was developed, to include a user need letter to RES and the issuance of contracts to assess all aspects of the high burnup fuel issue. Concurrently, meetings would be scheduled with the non-domestic experimenters and regulatory authorities to discuss the experimental data and to assess potential consequences and regulatory actions. Meetings with industry would be scheduled to discuss their planned actions and to solicit cooperation with the safety evaluations. Based on a complete review of all available fuel transient data, relevant to design basis events, NRR/RES would define acceptance criteria, establish a schedule for final assessment, and state need for further regulatory action.

Originating Documents: Commission memorandum from James M. Taylor (EDO), "Reactivity Transients and High Burnup Fuel," dated September 13, 1994, including IN 94-64, 'Reactivity Insertion Transient and Accident Limits for High Burnup Fuel,' dated August 31, 1994. Commission Memorandum from James M. Taylor, "Reactivity Transients and Fuel Damage Criteria for High Burnup Fuel," dated November 9, 1994, including an NRR safety assessment and the joint NRR/RES action plan.

Regulatory Assessment: There is no immediate safety issue, because of the low to medium burnup in currently operating cores. Since the fuel failure threshold declines with increasing burnup, the licensing basis design acceptance criteria may need to be redefined as a function of burnup. The end product of the plan will determine the need for regulatory action and will establish and define the need for further action on extended burnup cycles and high burnup fuel issues.

Current Status: The industry (NEI) submittal, evaluating the safety significance of recent high burnup data, was reviewed by the staff, and initial feedback was provided at a meeting, in which the industry further discussed their submittal. Further analytical assessments were presented at the CSNI Specialists Meeting in September and at the October Water Reactor Safety Information Meeting, which gave a summary of the industry (including EPRI) position. The Siemens, Westinghouse, B&W, ABB/CE, and GE evaluations of potential impact on their topical reports are being reviewed. The preliminary review indicates that the industry responses provide, in general, sufficient justification to show no current safety issues a confirm that there is no present licensing concern. However, the industry responses were . wholly consistent in detailing their plans for resolution and closeout of the high burnup fuel issue. The staff has contacted the individual fuel vendors to discuss their planned actions and schedule meetings. The first meetings were held on 9/28/95 (Westinghouse) and 12/12/95 (General Electric). The Industry Task Force stated that NRC formal feedback on the submittals was needed before additional industry actions are defined. A staff letter response is in concurrence, based on the industry assessments, which outlines the staff's ongoing plans and requests continued industry support. This letter will be sent to NEI, as the industry coordinator. The staff has concluded that additional actions by industry, other than the fuel vendor assessments that have been received and the continued vendor meetings, will not be needed at this time. An ACRS subcommittee meeting is scheduled for 4/96 to discuss the status of this issue.

NRR Technical Contacts:	Laurence Phillips, NRR/DSSA/SRXB, 415-3232 Edward Kendrick, NRR/DSSA/SRXB, 415-2891
RES Contact:	Ralph Meyer, RES/RPSB, 415-6789

RRG TOPIC AREA 55: CYCLE SPECIFIC PARAMETER LIMITS IN TECH SPECS AND GENERIC LETTER 88-16 REVISION

Last Update: 3/25/96
Lead NRR Division: DSSA

MILESTONES		DATE (T/C)
1.	Complete draft guidance for GL 88-16 revision	8/94C
2.	Office concurrences on GL (NRR/OGC/RES/OC)	n/a
3.	Contractor report received on reload report content	6/94C
4.	Complete draft guidance on contents of reload package (Reg. Guide) and GL 83-11 revision	9/94C
5.	Office concurrences on GL 83-11 revision	9/95C
6.	CRGR concurrence on GL 83-11 revision	10/95C
7.	EDO concurrence on GL 83-11 revision	n/a
8.	Publish proposed GL 83-11 revision for public comment	10/25/95C
9.	Receive public comments on GL 83-11 revision	12/11/95C
10.	Office concurrence on GL 83-11 revision	5/96T
11.	CRGR concurrence on GL 83-11 revision	6/96T
12.	EDO concurrence on GL 83-11 revision	8/96T
13.	Publish GL 83-11 revision	9/96T

Brief Description: This item recommended actions to reduce schedule and resource requirements for the NRC's review of reactor core reloads and the reload analysis methodology.

Historical Background: The objective of this task is to res, . to the Regulatory Review Group (RRG) Item #55. The RRG recommendations were to provide quicker review of core reload codes and to revise current Tech Specs to permit changes in accordance with approved core topical reports to take advantage of improved analyses without a license amendment by revising Generic Letter (GL) 88-16 (Core Operating Limits Report [COLR] Guidance. The task was subsequently revised to address the first recommendation only by preparing a supplement to GL 83-11 (Licensee Qualification for Performing Safety Analyses).

Proposed Actions: Prepare a supplement to GL 83-11 which presents criteria intended for licensees who wish to perform their own licensing analyses using previously approved methods. By complying with these criteria, the licensee would eliminate the need to submit a topical report qualifying its use of a previously approved methodology.

Originating Document: Regulatory Review Group Topic Area Item #55, Cycle Specific Parameter Limits in Tech Specs and Generic Letter 88-16 Revision.

Regulatory Assessment: This regulatory action has no safety impact on operating plants; it is intended to reduce resources required for methodology reviews.

Current Status: The proposed supplement to GL 83-11 was published for comment in the *Federal Register* on October 25, 1995. The comment period expired December 11, 1995. A final package has been developed and is currently on hold.

NRR Technical Contact: Larry Kopp, SRXB, 415-2879
NRR Lead PM: Steve Bloom, DRPW, 415-1313

References: Generic Letter 83-11 (February 8, 1983) and *Federal Register* Notice 60 FR 54712 (October 25, 1995).

THERMO-LAG ACTION PLAN

Last Update: 03/27/96
Lead NRR Division: DSSA

MILESTONES	DATE (T/C)
1. Semi-annual Commission status reports	Last: 9/20/95 Next: 04/96T
2. Resolve technical issues (Part I)	06/96T
3. Testing (Part II)	04/95C
4. Assess NRC fire prot. program (Part IV)	02/93C

Description: Evaluation and resolution of generic Thermo-Lag fire barrier issues regarding toxicity, construction and installation, fire endurance, ampacity derating, combustibility, seismic capabilities, and uniformity of materials. Includes special review team findings, public concerns, coordinating with Nuclear Energy Institute (NEI) and licensees, conducting fire endurance and ampacity derating tests, and assessing NRC reactor fire protection program. The staff has issued 16 generic communications regarding Thermo-Lag fire barriers.

Historical Background: In June 1991, the Office of Nuclear Reactor Regulation (NRR) established a special team to review the safety significance and generic applicability of technical issues regarding the use of Thermo-Lag fire barriers. In April 1992, the special review team issued its final report, which identified concerns about fire endurance, combustibility, and ampacity derating. Subsequently, the NRR staff prepared an action plan to address the issues associated with Thermo-Lag and the NRC fire protection program. The scope of the action plan includes coordination with industry and testing by the staff.

Proposed Actions: Specific actions include (1) the resolution of concerns and generic issues raised by the special review team and (2) resolution of plant-specific issues that emerge from the generic issues. In June 1994, the Commission approved a staff recommendation to resolve Thermo-Lag concerns by requiring compliance with existing NRC requi[illegible]nts and to permit plant-specific exemptions, where justified.

Originating Document: Final Report of the Special Review Team for the Review of Thermo-Lag Fire Barrier Performance, April 1992.

Regulatory Assessment: In response to Bulletin 92-01 and its supplement, licensees with Thermo-Lag fire barriers established NRC-approved measures, such as fire watches, to compensate for possibly inoperable fire barriers. The combination of compensatory measures and the defense-in-depth fire protection features provides an adequate level of fire protection until licensees implement permanent corrective actions.

Current Status: NRR staff briefed the Chairman on 02/08/96. At the request of the Chairman, the staff is considering options for ensuring that licensees complete corrective action programs in accordance with schedular commitments made in response to Generic Letter 92-08. The staff is considering, for example, periodic followup with individual licensees and confirmatory orders.The staff prepared the semiannual report to the Commission on the status of the Thermo-Lag Action Plan that is due April 1996.

Two major milestones remain: (1) mechanical properties test program and (2) plant-specific fire test curve feasibility study. NIST has completed all mechanical property tests. This included shear, flexural, compression and pure tension tests. NIST is currently documenting the results and will submit its test report before the end of April 1996. After the staff receives the report, it will assess the test results and determine whether or not additional staff or industry action is warranted. Because of the blizzard of 1996, a problem with test equipment operability, and the furlough of the National Institute of Standards and Technology (NIST), which is providing technical assistance, the originally proposed overall completion date of March 1996 has slipped to June 1996. The staff does not anticipate further delay.

NIST submitted its final draft report regarding the feasibility of developing fire curves for rating fire barriers on the basis of representative nuclear power plant fire hazards rather than the fire curves specified in existing fire test standards on November 9, 1995. The staff provided comments and technical direction to NIST by letter dated November 30, 1995, and during a meeting on December 7, 1995. NIST provided the results of its study to the Advisory Committee for Reactor Safeguards, Fire Protection Subcommittee, during the February 29, 1996, meeting. On the basis of its work, NIST has concluded that it would be possible to develop nuclear power plant specific fire curves. The overall completion date for the staff work on this issue is July 1996 (Yellow Ticket 0940144).

The staff planned to meet with NEI on March 20, 1996 to discuss Revision 2 of NEI Thermo-Lag Application Guide. However, NEI did not submit the revision in time for the staff to review it before the meeting. The meeting will be rescheduled.

The review, implementation, and inspection of plant-specific corrective actions is tracked as Multi-Plant Action L208 with plant-specific TAC numbers in WISP. These actions are not part of the Thermo-Lag Action Plan but appear on the Chairman's tracking list as item II.N.1. Responses to 2.206 petitions are also tracked by TAC numbers in WISP. Since the last status report, the staff drafted a final director's decision in response to eight petitions pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations in regard to the use of Thermo-Lag by licensees.

Contacts: D. Oudinot, SPLB, 301-415-3731
M. Gamberoni, DRPW, 301-415-3024

References:

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," June 24, 1992.

Bulletin 92-01, Supplement 1, "Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function," August 28, 1992.

Generic Letter 92-08, "Thermo-Lag 330-1 Fire Barriers," December 17, 1992.

Memorandum of September 20, 1995, from J. M. Taylor, EDO, to the Commission, "Semiannual Report on the Status of the Thermo-Lag Action Plan and Fire Protection Task Action Plan."

WOLF CREEK DRAINDOWN EVENT: ACTION PLAN

Last Update: 03/28/96
Lead NRR Division: DSSA

MILESTONES	DATE (T/C)
1. Draft Generic Letter	11/95(C)
2. Issue Supplement to IN 95-03	03/96(C)
3. Complete Draft TI/ Issue to the Regions for Comments	04/96(T)
4. Generic Letter to be Concurred by CRGR/ Letter Issued	04/96(T)
5. Receive Regional Comments on TI	06/96(T)
6. Complete Evaluation of the Responses to the Generic Letter	09/96(T)
7. Issue TI	09/96(T)
8. Complete Inspections (As necessary)	12/96(T)

Description: The objective of this action plan is to collect and evaluate information from the licensees regarding plant system configurations and vulnerabilities to draindown events. A 10 CFR 50.54(f) letter will be used to gather the information.

Historical Background: On September 17, 1994, the Wolf Creek plant experienced loss of reactor coolant system (RCS) inventory, while transitioning to a refueling shutdown. The event occurred when operators cycled a valve in the train A side of the RHR system cross-connect line following maintenance on the valve, while at the same time establishing a flow path from the RHR system, train B, to the refueling water storage tank for reborating train B. The failure of the reactor operating staff to adequately control two incompatible activities resulted in transferring 9200 gallons of hot RCS water to the RWST in 66 seconds.

The Wolf Creek event represents a LOCA with the potential to consequentially fail all the ECCS pumps and bypass the containment. Another important [illegible]e of this event is the short time available for corrective action. Based upon calculations [illegible] licensee and the staff, it is estimated that if the draindown had not been isolated within 3-5 minutes, net positive suction head would have been lost for all ECCS pumps, and core uncovery would follow in about 25-30 minutes. This event represents a PWR vulnerability which was not previously recognized.

Proposed Actions: Specific actions of this generic action plan are: (1) issue IN 95-03 issued January 12, 1995; and supplement to IN 95-03 which is being issued, (2) Request all PWR licensees, via an information gathering (10 CFR 50.54(f)) Generic Letter (GL), to provide information on draindown vulnerabilities and the measures they implemented to diminish the probability of a draindown.

Originating Document: AEOD/S95-01, "Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994".

Regulatory Assessment: The staff performed an evaluation of the probability for event initiation and of the conditional core damage probability. The value of this probability for core damage along with licensee awareness for this scenario makes the risk for continued PWR operation acceptably small.

Current Status: Information Notice IN 95-03 Supplement has been issued. The generic letter CRGR package is in concurrence in DRPM.

NRR Technical Contact:	Lambros Lois, SRXB, 415-3233
NRR Lead PM:	J. C. Stone, DRPW, 415-3063

References:

* AEOD/S95-01, "Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994"

* IN 95-03, issued January 18, 1995.

* Action Plan dated October 20, 1995

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Open Generic Communication and Compliance Activities
Sorted by Lead Technical Division and Branch

TAC	Type	Contact	LA Comp	Title	Description
** LTD =					
* LTB = Special Inspections Branch					
M92594	IN	JRTappert	4/30/96 T	IN: Fires in Emergency Diesel Generator Excitors	Fuse failures in EDG start-event that could remain undetected.
** LTD = Associate Director for Projects					
* LTB = Technical Specifications Branch					
M91404	GL	JWShapaker	5/17/96 T	GL: Administrative Controls Section	Line item improvement, guidance on revising the admin controls section of T.S.
M92544	GL	JWShapaker	6/28/96 T	GL: Design Features Technical Specifications	Guidance to revise the design features section of T.S. (line item improvement)
** LTD = Division of Engineering					
* LTB = Civil Engineering and Geosciences Branch					
M85236	LT	TAGreene	9/30/96 T	Problem of Grease Leakage in Prestressed Concrete Containment	Petroleum-based grease leaks could reduce concrete strength. 40 plants have greased unbonded tendons in their containment.
M92553	LT	RABenedict	9/1/96 T	Investigate Impact of Failure of SMRFs (During Northridge EQ) to NPP Steel Structures	Certain steel framing members failed in earthquake. Determine if same construction used in other plants.

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TAC	Type	Contact	LA Comp	Title	Description
M93707	GL	JWShapaker	6/28/96 T	GL: Plant Shutdown Criteria Following an Earthquake	Announce NRC approval of OBE exceedance criteria and associated plant shutdown guidelines proposed by EPRI as acceptable alternative to NRC interim guidelines for recommending plant shutdown following an earthquake.
M94293	GL	JLBirmingham	12/31/96 T	GL: NRC Preliminary Findings Related To The Use Of Reduced Seismic Criteria For Temporary Conditions.	Develop a GL to advise licensees that the use of reduced seismic criteria for temporary conditions may involve unreviewed safety questions and staff review may be needed.
M94861	IN	RABenedict	7/1/96 T	IN: Liner Plate Corrosion in Concrete Containment	Corroded liner might be weakened against post-accident leakage.
* LTB = Electrical Engineering Branch					
M91622	IN	JRTappert	4/25/96 T	IN: Inadequate Control of Molded-Case Circuit Breakers	Inappropriate pre-conditioning of breakers before surveillance.
* LTB = Materials and Chemical Engineering Branch					
M67462	LT	EJBenner	7/28/96 T	Augmented Reactor Vessel Inspection	Provide answers to questions as licensees implement 10 CFR 50.55(g)(6)(ii)(4) requiring augmented reactor vessel inspections.

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TAC	Type	Contact	LA Comp	Title	Description
M93024	LT	CVHodge	1/31/97 T	Evaluate Impact of RCP Support Column Tilt on Leak Before Break Analyses	To avoid interference with crossover leg, RCP support placed closer to vessel.
M93227	IN	EJBenner	3/8/96 L	IN: Fish Mouth Burst and Bowing of Previously-Plugged Steam Generator Tubes	Discusses recommendations made by Westinghouse in response to Haddam Neck event where previously-plugged steam generator tubes were found to have burst and bowed, potentially impacting other tubes.
M93643	IN	EJBenner	4/12/96 T	IN: Augmented Examination of Reactor Vessel	Discusses rule 10 CFR 50.55a(g)(6)(ii)(A) on augmented vessel exams
M94862	IN	EJBenner	9/1/96 T	IN: Steam Generator Tube Inspection Results	Discusses weakness in licensee's methods for identifying and sizing SG tube indications
* LTB = Mechanical Engineering Branch					
M93706	GL	JWShapaker	6/28/96 T	GL: Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves	Linked to a Task Action Plan
M93841	LT	EMMcKenna	4/30/96 T	Implications of Target Rock 2-Stage SRV Pilot Leakage	Evaluate safety inplications of leakage on valve operability and adequacy of leak detection.

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TAC	Type	Contact	LA Comp	Title	Description
M94371	IN	TJCarter	4/15/96 T	IN: Valve Stem Coupling of Gimpel Auxiliary Feedwater Turbine Trip Throttle Valves	Identifies mechanisms for linkage disengagement that impact turbine throttle valve operability.
** LTD - Division of Inspection and Support Programs					
* LTB - Special Inspections Branch					
M93979	IN	JRTappert	5/15/96 T	IN 92-68, Supp: Potentially Substandard Slip-On, Welding Neck, and Blind Flanges	Alerts licensees to potentially substandard flanges supplied by foreign vendors.
M94794	IN	ENFields	6/1/96 T	IN: Deficiencies in Material Dedication and Procurement Practices and Vendor Audits	Headquarters inspector in response to allegation found deficiencies in material dedication and procurement practices and vendor audits.
M95074	IN	DLSkeen	6/30/96 T	IN: Problems with Westinghouse DHP Circuit Breaker Levering-In Device	Worn and damaged levering-in device may prevent breaker from closing.
** LTD - Division of Reactor Controls and Human Factors					
* LTB - Human Factors Branch					
M92294	LT	NKHunemuller	12/31/96 T	Develop Regulatory Guide For Part 26 to Describe Acceptable Methods For FFD Programs to Address Fatigue	Develop guidance for the nuclear industry that will describe acceptable methods for licensees to address fatigue as a FFD issue in light of Commission Policy and 10 CFR 26 requirements.

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TAC	Type	Contact	LA Comp	Title	Description
* LTB = Instrumentation and Controls Branch					
M94468	IN	ENFields	4/15/96 T	IN: Improper Equipment Settings Due to the Use of Non-Temperature Compensated Test Equipment	Regional inspection found non-temperature compensated test euipment used to calibrate safety -related equipment.
* LTB = Operator Licensing Branch					
M93336	GL	JWShapaker	6/28/96 T	GL: Exemption For Applicants For the Senior Reactor Operator License Limited to Fuel Handling (LSRO)	Applicants for a LSRO may request an exemption from the requirements in 10 CFR 55.31(a)(5) since literal compliance is inappropriate.
M94840	GL	JWShapaker		GL: Changes in The Operator Licensing Program and Issuance of Rev. 8 of NU[illegible]-1021	
* LTB = Quality Assurance and Maintenance Branch					
M91542	IN	EYWang	4/30/96 T	IN: ANSYS and GTSTRUDL Computer Program Error Notifications	Part 21 notifications regarding ANSYS and GTSTRUDL computer program errors. Some of these errors cause erroneousd calculations resulting in wrong answers which may not be detected by the user.

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TAC	Type	Contact	LA Comp	Title	Description
** LTD - Division of Reactor Program Management					
* LTB - Emergency Preparedness and Radiation Protection Branch					
M91620	GL	JWShapaker	10/30/96 T	GL: Revision to Augmentation Staffing Levels For Nuclear Power Plant Emergencies	Ensuring adequate staffing for emergencies.
* LTB - Events Assessment and Generic Communications Branch					
M91544	GL	JWShapaker	5/31/96 T	GL: Defining Info in Monthly Operating Report Required by Tech Specs	Reducing reporting requirements to the minimum needed by the staff (part of RRG).
M94470	IN	EYWang	8/23/96 T	IN: Overwithdrawal of TIP Probe	Discusses the potential of personnel exposure as a result of overwithdrawn TIP outside its shield room.
M94480	LT	DLSkeen	5/1/96 T	PM Survey: Spent Fuel Pool Cooling	All NRR PM's to take survey of spent fuel pool licensing basis.
* LTB - Safeguards Branch					
M86951	LT	JRTappert	2/28/98 T	Protection of Safety Equipment Against Vehicle Bombs	Rule has been issued. A TI will be drafted to verify licensee implementation. TAC will remain open to support TI inspections.

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TAC	Type	Contact	LA Comp	Title	Description
** LTD = Division of Systems Safety and Analysis					
* LTB = Analytical Support Group					
M94615	IN	JLBirmingham	5/17/96 T	IN: Users of Decay Heat Standard ANS 5.1 Get Different Results	Analytical Support Group found that calculations for decay heat, using various industry methods, may vary up to 25 percent. An information notice was proposed to inform the nuclear industry.
* LTB = Containment Systems and Severe Accident Branch					
M86925	BL	JWShapaker	4/30/96 T	BL 93-02 Supp: Generic/BWR Strainer Clogging	PART OF A TASK ACTION PLAN -- Final resolution of this issue, requesting licensee action.
* LTB = Electrical Engineering Branch					
M94841	IN	ENFields	5/4/96 T	IN: Loss of Offsite Power and Reactor Trip with One of Two EDGs Unavailable at Catawba Unit 2	Develop IN to discuss loss of offsite power and reactor trip with one of two EDGs unavailable at Catawba Unit 2.
* LTB = Plant Systems Branch					
M80296	LT	TAGreene	9/30/96 T	Generic Communications - Assessment of Turbine Failure at Vandellos 1	Development of staff NUREG or other publication to document turbine building fire issues for U.S. plants in light of Vandellos fire.

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Open Generic Communication and Compliance Activities
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TAC	Type	Contact	LA Comp	Title	Description
M91323	LT	NKHunemuller	5/31/96 T	Reactor Water Cleanup (RWCU) Study in Response to ACRS Concern	Review of the effects of an unisolated RWCU break at several BWR's. Result of ACRS concerns during the review of the ABWR
M92636	LT	TJCarter	6/30/96 T	Terry Turbine Dependability	Opened 6/28/95 to address a broadened look at Terry turbine dependability based on concerns from related TAC M92407, which has been closed. (TAC M92407 only addressed overspeeding due to governor valve stem binding.)
M93335	LT	ENFields	10/31/96 T	Main Control Room Envelope Unfiltered Inleakage	Use improved methodology to verify the effects of potential inleakage rates on compliance with radiation and toxic gas exposure limits inside the main control room.
M94045	IN	JRTappert	4/15/96 T	IN: Recent [illegible]ems with Overhead Cranes	Trojan experienced failure of overhead crane rail and Prairie Island experienced premature actuation of load limit device.
M94088	IN	EYWang	8/30/96 T	Removing Refueling Floor Shielding Plugs Prior to And Soon After Shutdown	Discusses the potential of being in an unanalyzed condition by removing refueling floor shield plugs prior to plant shutdown.
M94594	IN	JRTappert	4/30/96 T	IN: Wolf Creek Reactor Trip with One Train Essential Service Water System Inoperable	Evaluate generic implications of events relating to service water availability due to frazzle ice.

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Open Generic Communication and Compliance Activities
Sorted by Lead Technical Division and Branch

TAC	Type	Contact	LA Comp	Title	Description
M94912	BL	EYWang	5/24/96 T	BL: Movement of Dry Storage Casks Over Spent Fuel, Fuel in The Reactor Core, or Safety-Related Equipment	Discusses the issue involving the movement of heavy loads over SFP, over fuel in the reactor core, or over safety-related equipment.
* LTB = Reactor Systems Branch					
M80326	LT	SSKoenick	4/13/96 T	Accumulation of Volume Control Tank Cover Gass in ECCS Piping Connected to the Charging System.	Not a new issue, there have been several generic communications already issued. SRXB would like to close this out by memo.
M87297	LT	EJBenner	6/30/96 T	Generic Model For Probability of Operation With a Mis-Oriented Fuel Bundle	Model for non-detection of a mis-positioned fuel bundle during operation.
M91447	GL	JWShapaker	4/19/96 T	GL: Borafle- egradation in Spent Fuel Pool S ge Racks	Problems with previously unidentified high rate of Boraflex degradation, criticality concern.
M91599	GL	JWShapaker	4/19/96 T	GL 83-11 Supp: Licensee Qualification For Performing Safety Analyses in Support of Licensing Actions	PART OF A TASK ACTION PLAN -- Provides alternative means of licnesee qualification for performing sanalyses using generically approved methods.
M92635	GL	JWShapaker	5/15/96 T	GL: Reactor Coolant Inventory Loss and Potential Loss of Emergency Mitigation Functions While Shutdo	Loss of ECCS function due to steam voiding in RWST line to suction of ECCS pumps due to loss of RCS inventory in Mode 4 (Wolf Cree' .

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TAC	Type	Contact	LA Comp	Title	Description
M93751	IN	RABenedict	4/10/96 T	IN: Closed Head Vent Causes Inaccurate Level Indication During Reduced Inventory	Improper venting of reactor coolant system permitted water level changes in reactor vessel to go undetected during reduced inventory operations.
M94565	BL	DLSkeen	9/30/96 T	BL: Slow Scram Solenoid Pilot Valves Caused by Viton Diaphragms	Scram solenoid pilot valves with viton diaphragms showing degraded scram times within 6-8 months. Currently tracking licensee response to RRG recommendations.
M94808	IN	EJBenner	4/12/96 T	IN: Potential Clogging of HPSI Throttle Valves During Containment Sump Recirculation Phase	Develop IN to discuss HPSI clogging phenomenon discovered at Diablo Canyon & Millstone

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Generic Communication and Compliance Activities Added
Since the Last Public Report (January 1996)

TAC	Type	Contact	Tech Branch	LA Comp	Status	Title	REASON ADDED
M94468	IN	ENFields	Instrumentations and Controls Branch	4/15/96	T	IN: Improper Equipment Setting Due to the Use Compensated Test Equipment	1/17/96 - Events Assessment Panel authorized development of IN at it 1/16/96 meeting
M94470	IN	EYWang	Events Assessment and Generic Communications Branch	8/23/96	T	IN: Overdrawal of TIP Probe	1/1796 - EAP authorized development of the IN at its 1/16/96 meeting
M94480	LT	DLSkeen	Events Assessment and Communications Branch	5/1/96	T	PM Survey: Spent Fuel Pool Cooling	AE Chaffee authorized long-term follow up on 01/17/96
M94565	BL	DLSkeen	Reactor System..	9/30/96	T	BL: Slow Scram Solenoid Pilot Valves Caused by Viton Diaphragms	1/26/96 - Acting Branch Chief EFGoodwin authorized development of this IN/BL..IN 96-07 issued 1/26/96 - Bulletin under consideration.
M94594	IN	JRTappert	Plant Systems Branch	4/30/96	T	IN: Wolf Creek Reactor Trip with One Train System Inoperable Essential Service Water	2/1/96: AEChaffee authorized development of IN. EAP endorsed this approach at its 2/6/96 meeting.

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TAC	TYPE	Contact	Tech Branch	LA Comp	Status	Title	Reason Added
M94615	IN	JLBirmingham	Analytical Support Group	5/17/96	T	IN: Users of Decay Heat Standard ANS 5.1 Get Different Results	EAP authorized development of IN at its 2/6/96 meeting.
M94794	IN	ENFields	Special Inspections Branch	6/1/96	T	IN: Deficiencies in Material Dedication and Procurement Practices and Vendor Audits	2/20/96: AEChaffee authorized development of IN. 3/19/96: The EAP Panel endorsed the development of IN.
M94840	IN	EJBenner	Reactor Systems Branch	4/12/96	T	IN: Potential Clogging of HPSI Throttle Valves During Containment Sump Recirculation Phase	2/27/96 - EAP authorized development of IN.
M94840	GL	JWShapaker	Operator Licensing Branch			GL: Changes in The Operator Licensing Program and Issuance of Rev. 9 of NUREG-1021	2/27/96 - EAP authorized development of GL.
M94841	IN	ENFields	Electrical Engineering Branch	5/4/96	T	IN: Loss of Offsite Power and Reactor Trip with One of Two EDGs Unavailable at Catawba Unit 2	2/27/96 - AEChaffee authorized development of IN. 3/26/96 - The EAP panel endorsed the development of IN.

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TAC	TYPE	Contact	Tech Branch	LA Comp	Status	Title	Reason Added
M94861	IN	RABenedict	Civil Engineering and Geosciences Branch	7/1/96	T	IN: Liner Plate Corrosion in Concrete Containment	EAP authorized development of IN at its 3/5/96 meeting.
M94862	IN	EJBenner	Materials and Chemical Engineering Branch	9/1/96	T	IN: Steam Generator Tube Inspection Results	EAP authorized development of IN at its 3/5/96 meeting.
M94912	BL	EYWang	Plant Systems Branch	5/24/96	T	BL: Movement of Dry Storage Casks Over Spent Fuel in the Reactor Core, or Safety-Related Equipment	The EAP authorized development of BL at its 3/19 meeting.
M95074	IN	DLSkeen	Special Inspections Branch	6/[illegible]6	T	IN: Problems with Westinghouse DHP Circuit Breaker Levering-In Device	The EAP authorized development of IN at its 3/26/96 meeting.

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TAC	Type	Contact	Tech Branch	LA Comp	Status	Title	Reason Closed
M82072	GL	JWShapaker	Mechanical Engineering Branch	1/26/96	C	GL 89-10, Supp 7: Consideration of Position Changeable Valves	Management decision to change the definition of a GCCA to include generic communications, even those all which are part of a task action plan.
M90863	GL	JWShapaker	Instrumentation and Controls Branch	2/27/96	C	GL: Inadequate Testing of Safety Related Logic Circuits	GL 96-01 issued on 1/10/96.
M91749	GL	JWShapaker	Technical Specifications Branch	1/31/96	C	GL: Relocation of RCS Pressure/Temperature Limits	GL 96-03 issued on 1/31//96.
M91896	GL	JWShapaker	Safeguards Branch	2/13/96	C	GL: Reconsideration of Plant Security Requirements	GL 96-02 issued 2/13/96.
M92601	IN	TJCarter	Reactor Systems Branch	3/19/96	C	IN: BWR Stability With Flow Slightly Less Than Natural Circulation Flow	IN 96-10 issued 2/13/96.
M93360	IN	EJBenner	Containment Systems and Severe Accident Branch	2/13/96	C	IN: Blockage of Untested ECCs Piping	IN 96-10 issued 2/13/96.

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TAC	TYPE	Contact	Tech Branch	LA Comp	Status	Title	Reason Closed
M93400	IN	EJBenner	Mechanical Engineering Branch	1/5/96	C	IN: Inoperability Masked by Downstream Indications During Testing	The Events Assessment Panel authorized development of the IN at its 9/5/95 meeting.
M93568	IN	ENFields	Reactor Systems Branch	3/25/96	C	IN 95-03, Supp: Loss of RC Inventory and Potential Loss Emer Mitigation Functions While in a Shut	Management Decision to provide separate TACs for TAP, GL, and IN.
M93568	IN	ENFields	Materials and Chemical Engineering Branch	3/25/96	C	IN: Control Rod Drive Mechanism Penetration Cracking	IN 96-11 issued 2/14/96.
M93752	IN	CVHodge	Reactor Systems Branch	1/ '96	C	IN: Shutdown Cooling Flow Bypassing Core Results in Temperature and Pressure Increases	The Events Assessment Panel authorized development of the IN at its 10/3/95 meeting.
M93753	IN	JRTappert	Containment Systems and Severe Accident Branch	2/26/96	C	IN: Potential Containment Leak Path Through Hydrogen Analyzer	IN 96-13 issued on 2/26/96.

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TAC	TYPE	Contact	Tech Branch	LA Comp	Status	Title	Reason Closed
M93754	IN	TKoshy	Plant Systems Branch	1/26/96	C	IN: Inadequate Testing and Design of Tornado Dampers	The Events Assessment Panel authorized development of the IN at its 10/3/95 meeting
M93842	LT	EJBenner	Reactor Systems Branch	2/2/96	C	Assessment of Corrosion of B&W Fuel Used in 2 Year Fuel Cycles	This TAC is closed based on the memo from RCJones to AEChaffee 2/22/96. The staff has concluded that there is a need to inspect the B&W fuel facility.
M94004	IN	JRTappert	Mechanical Engineering Branch	1/15/96	C	IN: Environmental Effects on Main Steam Safety Valve Set Point	IN 96-03, "Main Steam Safety Valve Setpoint Variation as a Result of Thermal Effects," was issued on 1/5/96.
M94044	IN	NKHunemuller	Events Assessment and Generic Communicati ons Branch	3/13/96	C	IN: Inadvertent Draining of Reactor Vessel and Isolation of Shutdown Cooling System	IN 96-15 issued on 3/8/96.
M94189	IN	TJCarter	Mechanical Engineering Branch	2/5/96	C	IN: Damage to Valve Internals Caused by Thermally - Induced Pressure Locking	Events Assessment Panel authorized development of the IN at its 12/5/95 meeting.
M94254	IN	EJBenner	Materials and Chemical Engineering Branch	2/12/96	C	IN: Damage in Foreign Steam Generator Internals	IN 96-09 issued 2/12/96

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M94370	LT	EMMcKenna	Operator Licensing Branch	2/5/96	C	Interface between Operators and Nuclear Engineers during Tests and Startups	Based on the memo from SARichards (HOLB) to AEChaffee, dated 2/5/96, this TAC is closed.
M94469	IN	NKHunemuller	Probabilistic Safety Assessment Branch	3/4/96	C	IN: Use of Individual Plant Examinations (IPEs) for Regulatory Decision Making	2/14/96 - Originator cancelled, letter to NEI to be issued.
M94494	IN	SSKoenick	Reactor Systems Branch	2/15/96	C	IN: South Texas Stuck Rod Event Following Reactor Trip	02/15/96 - IN 96-12 issued.
M94521	IN	EYWang	Events Assessment and Generic Communications Branch	3/4/96	C	Radwaste Facility Equipment Degradation at Millstone Unit 1	IN 96-14 issued on 3/1/96.
M94608	IN	JRTappert	Reactor Systems Branch	3/13/96	C	BL: Stuck Control Rod Problems	BL 96-01 issued on 03/18/96.
M94768	IN	JLBirmingham	Emergency Preparedness and Radiation Protection Branch	4/2/96	C	IN: Failure of Tone Alert Radio to Activate When Receiving a Shortened Activation Signal	IN 96-19 issued on 04/02/96

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M94778	IN	DLSkeen	Reactor Systems Branch	1/26/96	C	IN: Slow Five Percent Scram Insertion Times Cause by Viton Diaphragms Pilot Valves	1/26/96 - Acting Branch Chief EFGoodwin authorized development of IN. IN 96-07 was issued on the same day.
M94911	IN	TJCarter	Events Assessment and Generic Communicati ons	3/18/96	C	IN: Reactor Operation Believed To Be Inconsistent with That Described in The FSAR	IN 96-17 issued on 03/18/96.