

**SEMIANNUAL STATUS REPORT ON THE DISPOSITION OF
PROPOSED GENERIC ISSUES**

OFFICE OF NUCLEAR REGULATORY RESEARCH

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INTRODUCTION

The Generic Issues Program addresses potential safety or security issues that do not fit clearly into other programs at the U.S. Nuclear Regulatory Commission (NRC). The Generic Issues Program includes five stages: (1) identification, (2) acceptance review, (3) screening, (4) safety/risk assessment, and (5) regulatory assessment. Throughout the five stages of the Generic Issues Program, the Program staff track and communicate generic issue status. The Semiannual Status Report on the Disposition of Proposed Generic Issues (GIs) contains status descriptions of issues that are active or have been recently closed in one of the first three stages of the Program.

An issue remains a proposed GI (pre-GI) until and unless the issue passes the screening stage and either becomes a GI or is dropped and exits the Program. During each stage of the Generic Issues Program, staff determines if more information is needed, if the issue should proceed to the next stage, or if the issue should exit the Program. As described in Management Directive 6.4, "Generic Issues Program," dated November 17, 2009, the Program will address only those issues that meet the Program's seven screening criteria. Proposed GIs that fail to meet any of these criteria at any time will exit the program and will not become GIs. When a proposed GI exits the Program, the possible outcomes include no action, further research, referral to appropriate regulatory program(s), or a voluntary industry initiative.

Proposed GIs that warrant further assessment in the Program and pass the screening stage will proceed to safety/risk assessment stage as a GI. Status of active GIs is documented separately in Quarterly Generic Issue Management Control System reports. Moreover, NUREG-0933, "Resolution of Generic Safety Issues," contains a description of the process and results of the resolution of each GI resolved under the Generic Issues Program.

Semiannual Status Report on the Disposition of Proposed GIs provides a brief description of proposed GIs, their identification and latest status in the Generic Issues Program. In addition, this report provides information regarding the acceptance review and screening stages for those proposed GIs that have been processed in those stages.

Issue Number: PRE-0001

Date Received: 2/18/2008

Issue Title: Multi-Unit Core Damage Events

Assigned GI Number:

Overall Status: Screening Review

Name of Submitters: State-of-the-Art Reactor Consequence Analyses Project staff, RES/DRA/PRAB

Acceptance Review Information

Date: 8/26/2008

POC: John Kauffman

Status: Accepted

Screening Review Information

Status: Review in Progress

Date:

POC: John Kauffman

Working Group:

Panel: Dube, Donald; Stutzke, Martin; Weerakkody, Sunil

Brief Description

Accident sequences have been identified that potentially result in core damage occurring in multiple units of a multiple-unit site. Multiple-unit core damage scenarios generally result from an initiator that can fail similar equipment in each unit (e.g., a seismic event or an internal flooding event) or are a result of a high degree of sharing of systems among the units, such as sharing all diesel generators.

The State-of-the-Art Reactor Consequence Analyses (SOARCA) project is under development and the results are preliminary. During the sequence identification and selection process for the SOARCA project for Surry Power Station and Peach Bottom Atomic Power Station, the SOARCA staff identified potential scenarios in which both units at each plant would be expected to experience accident sequence progression pathways leading to core damage as a result of the initiating event. These scenarios have estimated frequencies that are above the screening threshold for inclusion into the SOARCA project (in fact, they are among the highest frequency scenarios for each plant). Dual-unit events could impact not only the potential source terms but also the potential effectiveness and success of mitigative measures, because the plant staff must deal simultaneously with accidents in both units.

Postulated multiple-unit station blackout (SBO) would challenge the ability of the plant operating staff to respond. Current available plant resources (technical staff and equipment) are based on the assumption that only a single unit experiences an SBO and other units have not lost emergency alternating current power. Thus, it is assumed that other units can proceed to safe shutdown. A multiple-unit SBO may require more equipment, such as diesel-driven pumps and portable direct current generators, than what is currently available at most sites. Multi-unit SBOs that lead to core damage in two or more units will potentially increase the radionuclide releases and offsite consequences.

Status Notes

The screening panel has met. The screening panel memorandum with recommendations is in concurrence review.

Issue Number: PRE-0002

Date Received: 3/11/2008

Issue Title: Assessment of Debris Accumulation on Boiling-Water Reactor Sump Performance

Assigned GI Number:

Overall Status: Not Accepted as a Generic Issue

Name of Submitters: John P. Burke, RES/DE/MEEB

Acceptance Review Information

Date: 7/6/2010

POC: John Kauffman

Status: Not Accepted

Screening Review Information

Status: Not Applicable

Date:

POC:

Working Group:

Panel:

Brief Description

Debris accumulation on the emergency core cooling system (ECCS) sump screen in boiling-water reactors (BWRs) may result in a loss of the net positive suction head (NPSH) margin for ECCS pumps. Loss of NPSH margin could impede or prevent the flow of water from the sump, which is necessary to meet the criteria of Title 10 of the Code of Federal Regulations (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

Status Notes

Before performing the acceptance review, the Office of Nuclear Regulatory Research (RES) performed a scoping study of the issue. The results are published in NUREG/CR-7011, "Evaluation of Treatment of Effects of Debris in Coolant on ECCS and CSS Performance in Pressurized Water Reactors and Boiling Water Reactors," issued May 2010. Based on the results and recommendations of the study, the Generic Issues Program staff concluded that the issue is being addressed by other existing programs and processes. Accordingly, the issue does not meet Generic Issues Program criterion "The issue cannot be readily addressed through other regulatory programs and processes; existing regulations, policies, or guidance; or voluntary industry initiatives". The basis for concluding that the Program criterion is not met includes:

- NUREG/CR-7011 states that "NRC has prepared a draft revision to RG 1.82 (DG 1234) that adopts the recommendations discussed in this report ... This revision to the regulatory guide incorporates lessons learned from resolution of Generic Letter 2004-02 (GL 04-02) and from staff guidance contained in safety evaluations prepared for industry topical reports for GL 04-02. Upon publication of this revision, the regulatory guidance will be consistent for BWRs and [pressurized-water reactors] PWRs except for [certain] guidance that is unique to each specific reactor type."

- NUREG/CR-7011 also states that the Boiling Water Reactor Owners Group (BWROG) has begun to address the recommendations in NUREG/CR-7011, and has discussed its resolution strategies and associated schedules with the NRC staff in a series of public meetings that started in June 2008. NUREG/CR-7011 recommendations include the need to conduct plant-specific walkdowns to determine debris types, conduct chemical effects testing, evaluate downstream effects, and assess existing head loss test data for the suction strainers.

Because this issue is being addressed by other existing programs and processes as explained above, the Generic Issues Program did not accept this issue as a generic issue.

Issue Number: PRE-0003

Date Received: 7/29/2008

Issue Title: Review Guidance for Storage Pool Criticality Safety

Assigned GI Number:

Overall Status: Not Accepted as a Generic Issue

Name of Submitters: Donald E. Carlson, NRO/DNRL/DDIP

Acceptance Review Information

Date: 2/27/2009

POC: John Kauffman

Status: Not Accepted

Screening Review Information

Status: Not Applicable

Date:

POC:

Working Group:

Panel:

Brief Description

Apparent technical deficiencies have been identified in the staff guidance used in reviewing the criticality analyses that demonstrate compliance with General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities," and with the requirements under 10 CFR 50.68, "Criticality Accident Requirements," for nuclear criticality safety in fuel storage pools. The review guidance in question appears primarily in Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3, issued March 2007, of NUGEG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," and the staff's memo from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 19, 1998, which is widely cited in licensee submittals and staff safety evaluations. Licensing safety analyses performed and approved in accordance with the established review guidance and practices are likely to contain various nonconservative errors and omissions in calculating the fuel pool's effective neutron multiplication factor, keff, under the conditions evaluated for compliance. Some of the analysis areas with apparent or potential review deficiencies include: burnup credit validation, fission product modeling, spent fuel record accuracy, absorber plate granularity, bundles with removed pins, actual fuel pin conditions, Monte Carlo undersampling, and potential pool wall effects.

Status Notes

The Generic Issues Program staff concluded that this proposed issue should not be accepted into the Generic Issues Program because the issue could be effectively and efficiently addressed by the program offices. After the Generic Issues Program issued this finding, NRR contacted the Nuclear Energy Institute (NEI) via letter to request a meeting (ADAMS Accession No. ML083100004). This meeting resulted in a collaborative effort to address this issue.

Issue Number: PRE-0004

Date Received: 8/13/2008

Issue Title: Loss-of-Coolant Accident with Delayed Loss of Offsite Power

Assigned GI Number:

Overall Status: Withdrawn

Name of Submitters: Vijay Goel and Sheila Ray, NRR/ADES/DE/EEEB

Acceptance Review Information

Date:

POC:

Status:

Screening Review Information

Status: Not Applicable

Date:

POC:

Working Group:

Panel:

Brief Description

Currently, nuclear plants are designed to handle only the simultaneous loss-of-coolant accident (LOCA) with delayed loss of offsite power (LOOP). Based on responses received to Question 3 of GL 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 1, 2006, only a few plants, such as San Onofre Nuclear Generating Station, Palo Verde Nuclear Generating Station, and Waterford Steam Electric Station addressed the response of safety-related equipment (e.g., emergency diesel generators or safety-related motors) when subjected to LOCA with delayed LOOP (double sequencing of equipment).

Status Notes

Upon review, the Generic Issues Program staff identified GI-171, "ESF Failure from LOOP Subsequent to a LOCA," that previously addressed the issue. Specifically, for GI-171, the NRC conducted a study demonstrating that the contribution to core damage frequency from the sequence of events was far less than originally anticipated. The results of the study are published in NUREG/CR-6538, "Evaluation of LOCA with Delayed LOOP and LOOP with Delayed LOCA Accident Scenarios," issued July 1997, and GI-171 was deemed resolved. When informed of this information, the submitters withdrew their proposed generic issue.

Issue Number: PRE-0005

Date Received: 9/5/2008

Issue Title: Electromagnetic Pulse Attack Threat

Assigned GI Number:

Overall Status: Not Accepted as a Generic Issue

Name of Submitters: Tom Farnholtz, OEDO/TRPS

Acceptance Review Information

Date: 11/19/2008

POC: Jack Foster

Status: Not Accepted

Screening Review Information

Status: Not Applicable

Date:

POC:

Working Group:

Panel:

Brief Description

The concern with electromagnetic pulse (EMP) attack was discussed in the report by the EMP Commission. The EMP Commission assessed the threat to the United States from an EMP attack. The postulated scenario of most interest is that a high-altitude nuclear blast with the appropriate characteristics is used to attack the United States. The resulting EMP damages electrical and electronic components, such that generating stations and control systems are damaged or knocked offline, resulting in loss of the grid to a large area. The grid outage could last several months or longer due to damage to the generating units, damage to large transformers, and the limited worldwide manufacturing capability to make large replacement transformers. In addition, the extended blackout would impact transportation systems needed to move replacement transformers. For nuclear plants, the stations would experience long-term LOOPs and would be reliant on emergency diesel generators, with limited fuel supply and limited means of refueling. The power plants could also experience other damage due to EMP. The extended blackout could also impact other important functions, such as emergency planning and the transportation and storage of nuclear materials.

Status Notes

The Generic Issues Program staff concluded that this proposed issue should not be accepted into the Generic Issues Program because the phenomenon and associated consequences, given an attack, needed further research to understand and quantify. The RES Division of Engineering further evaluated digital systems vulnerability to the EMP threat via the project titled, "Security Assessments of EM Vulnerabilities," which is part of the current RES Digital System Research Plan (ADAMS Accession No. ML061150050). The results of the additional research did not support the need for a generic issue.

Issue Number: PRE-0006

Date Received: 9/25/2008

Issue Title: Boron Precipitation Following a Loss-of-Coolant Accident in Pressurized-Water Reactors

Assigned GI Number:

Overall Status: Not Accepted as a Generic Issue

Name of Submitters: John Lamb, NRR/DORL/LPL1-2

Acceptance Review Information

Date: 4/30/2009

POC: Asimios Malliakos

Status: Not Accepted

Screening Review Information

Status: Not Applicable

Date:

POC:

Working Group:

Panel:

Brief Description

The proposed generic issue is the potential for boron concentration and precipitation to interfere with long-term core cooling following a LOCA in PWRs.

Status Notes

The Generic Issues Program staff's assessment of potential boron precipitation following a LOCA in PWRs is that this proposed generic issue is already addressed by existing regulations and guidance. The governing regulation is 10 CFR 50.46 and guidance is included in NUREG-0800, Section 6.3, "Emergency Core Cooling System." Section 6.3, under Subsection III, "Review Procedures," states that "The criteria, supporting analyses, plant design provisions, and operator actions that will be taken are reviewed to ensure that there will not be unacceptably high concentrations of boric acid in the core region (resulting in precipitation of a solid phase) during the long term cooling phase following a postulated LOCA." Because this proposed issue is addressed by 10 CFR 50.46 and the Standard Review Plan, the Generic Issues Program did not accept this issue as a generic issue.

Issue Number: PRE-0007

Date Received: 9/25/2008

Issue Title: Core Uncovery Following a Discharge Leg Loss-of-Coolant Accident

Assigned GI Number:

Overall Status: Not Accepted as a Generic Issue

Name of Submitters: John Lamb, NRR/DORL/LPL1-2

Acceptance Review Information

Date: 6/22/2009

POC: Asimios Malliakos

Status: Not Accepted

Screening Review Information

Status: Not Applicable

Date:

POC:

Working Group:

Panel:

Brief Description

The proposed generic issue is the potential of a long-term core uncovery following a LOCA induced by ruptures on the top of the reactor coolant pump discharge leg in PWRs. When the primary system break is located on the top of the cold-leg piping, the loop seal region will eventually fill with emergency core cooling injection. The coolant will fill the cold legs, allowing the liquid trapped in the vertical section of the suction-leg piping adjacent to the reactor coolant pumps, to form a large resistance to the flow of core-decay-heat-generated steam through the external-loop piping to the break. As a consequence of the water trapped in the loop seals, the pressure in the reactor vessel upper plenum will increase significantly in order to drive the steam through this region to the break. The pressure increase is large enough to depress the two-phase level into the core causing heatup of the fuel cladding. Although temperatures are expected to remain below 2,200 degrees Fahrenheit, oxidation limits are expected to be exceeded.

Status Notes

The Generic Issues Program staff's assessment of the potential threat of the core uncovery following a LOCA in PWRs is that this proposed generic issue is already addressed by existing regulations and guidance. The governing regulation is 10 CFR 50.46 and guidance is included in NUREG-0800, Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary." Section 15.6.5, under Subsection I, "Areas of Review," states that "A spectrum of both large and small break LOCAs are to be evaluated and the limiting break identified through sufficient analyses to determine the worst break peak clad temperature (PCT), the worst local clad oxidation, and the highest core wide oxidation percentage." The same subsection also states "The analyses must also be carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature." Also, "Break locations should include the side and top of the discharge leg to assure that the suction leg piping that fails to clear of liquid does not

result in depression of the two-phase mixture level into the core and result in the worst case PCT. If operator action is required to maintain conditions within 10 CFR 50.46 limits, then the equipment and operator action times to achieve a successful core cooling condition should also be identified.” Because this proposed issue is addressed by 10 CFR 50.46 and the Standard Review Plan, the Generic Issues Program did not accept this issue as a generic issue.

Issue Number: PRE-0008

Date Received: 1/5/2010

Issue Title: Water Hammer in the Residual Heat Removal System Following LOOP Coincident with a LOCA Signal

Assigned GI Number:

Overall Status: Not Accepted as a Generic Issue

Name of Submitters: Benjamin Parks, NRC/NRR/DSS

Acceptance Review Information

Date: 3/23/2010

POC: April Smith

Status: Not Accepted

Screening Review Information

Status: Not Applicable

Date:

POC:

Working Group:

Panel:

Brief Description

The proposed generic issue involves the potential for water hammer in the BWR residual heat removal (RHR) system following a LOOP coincident with a LOCA signal if the RHR system is aligned in the suppression pool cooling (SPC) mode of operation.

Operating experience shows that licensees have been known to use continuous RHR operation in the SPC mode to address a degraded or nonconforming condition such as leaking safety relief valve (SRVs). The NRC has the regulatory framework to review these activities and take regulatory action.

Status Notes

The Generic Issues Program did not accept this issue because it is addressed through other regulatory programs and processes; existing regulations, policies, or guidance; or voluntary industry initiatives. Specifically, 10 CFR 50.59, "Changes, Tests, and Experiments"; Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR 50; and the Reactor Oversight Process (ROP) address this proposed generic issue. 10 CFR 50.59 is cited in Information Notices 87-10 and 87-10 Supplement 1, which informed BWR licensees of this issue in 1987 stating, "When operating in the SPC mode, the RHR system is more likely to undergo a water hammer event if there is a loss of station power. Since the probability of a water hammer event increases as the amount of time the system is operated in the SPC mode increases, and the likelihood of damage to the system increases with the frequency of water hammer events, operating in the SPC mode more often than assumed in the FSAR may be an un-reviewed safety question as defined in 10 CFR 50.59(a)(2)(i). In addition, a significant increase in the amount of time the RHR system is operated may affect the amount and types of preventive maintenance and monitoring activities that are required to ensure that it is capable of performing its intended function."

The regulatory framework includes, and is not limited to:

- The ROP through safety system design and 10 CFR 50.59 inspections;
- The ROP per Part 9900, Technical Guidance, "Resolution of Degraded and Nonconforming Conditions," of the NRC Inspection Manual;
- 10 CFR Part 50, Appendix B to ensure design control; appropriate instructions, procedures, and drawings, and corrective action is taken to restore the degraded plant condition back to its previous condition as described in the updated final safety analysis report (e.g. fix or replace leaking SRVs).

Because the proposed issue is addressed through existing regulatory programs and processes, the Generic Issues Program did not accept this issue as a generic issue.

Issue Number: PRE-0009

Date Received: 7/19/2010

Issue Title: Flooding of Nuclear Power Plant Sites Following Upstream Dam Failures

Assigned GI Number:

Overall Status: Screening Review

Name of Submitters: George Wilson; Meena Khanna; Lois James, NRR/DRA/APOB

Acceptance Review Information

Date: 8/9/2010

POC: Richard Perkins

Status: Accepted

Screening Review Information

Status: Review in Progress

Date:

POC: Richard Perkins

Working Group: Sancaktar, Selim; Philip, Jacob; Bensi, Michelle

Panel:

Brief Description

During a review of a regulatory action associated with an operating nuclear power plant, the NRC staff identified a higher than expected potential for both the external flooding hazard due to a potential upstream dam failure and its associated consequences to the public health and safety and the environment.

Status Notes

The screening review is complete. The screening panel memorandum is in concurrence review.